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Safety of Existing Nuclear Installations

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ASAMPSA2
BEST-PRACTICES GUIDELINES
FOR LEVEL 2 PSA DEVELOPMENT AND APPLICATIONS

Volume 2 - Best practices for the Gen II PWR, Gen II BWR L2PSAs. Extension to Gen III reactors

Reference ASAMPSA2

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Main Authors	E. Raimond (IRSN), T. Durin (IRSN), N. Rahni (IRSN), R. Meignen (IRSN), M. Cranga (IRSN), H. Löffler (GRS), O. Mildenberger (GRS), G. Lajtha (NUBIKI), C.S. Santamaria (IBERINCO), J. Dienstbier (UJV), A. Rydl (UJV), J.E. Holmberg (VTT), I. Lindholm (VTT), I. Männistö (VTT), E-M. Pauli (AREVA GmbH), G. Dirksen (AREVA GmbH), L. Grindon (AMEC NNC), K. Peers (AMEC NNC), G. Hulqvist (FKA), F. Parozzi (ERSE), F. Polidoro (ERSE), E. Cazzoli (CCA), J. Vitazkova (CCA), L. Burgazzi (ENEA), , L. Oury (TRACTEBEL), C. Ngatchou (TRACTEBEL), S. Siltanen (Sipola) (FORTUM), I. Niemela (STUK), T. Routamo (STUK), , P. Helström (SCANDPOWER)
Other Contributors	F. Pichereau (IRSN), A. Bentaib (IRSN), Y. Guigueno (IRSN), C. Bassi (CEA), H. Brinkman (NRG), A. Seidel (VGB), B. Schubert (VGB / VATTENFALL), R. Wohlstein (VGB / EON), S. Guentay (PSI), L. Vinçon (AREVA SAS), JP. Van Dorsselaere (IRSN – SARNET coordinator)
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Summary :

The Volume 2 the ASAMPSA2 guideline presents specific recommendations regarding L2PSA development for Gen II and III NPPs.

Visa grid			
	Main author(s) :	**Verification**	**Approval (Coordinator)**
Name (s)	All Partners (see Above)	All Partners (see Above)	E. Raimond
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MODIFICATIONS OF THE DOCUMENT

Version	Date	Authors	Pages or paragraphs modified	Description or comments
1	03-12-2010	See above		Draft submitted for external review.
2	01-31-2013	See above	All	Additional text has been included to take into account some remarks of the external review.
3	04-30-2013	See above	Minor editorial modifications.	Version 1 of the ASAMPSA2 guidelines.

LIST OF DIFFUSION

Name	Organisation
All WP1, WP2, WP3, WP4 ASAMPSA2 Partners. Specific list of organizations concerned by L2PSA development and applications for NPP. The guidelines is a public document	In particular, organizations in countries members of OECD/NEA/CSNI and observers. www.asampsa2.eu www.irsn.fr

ASAMPSA2 PROJECT SUMMARY

The objective of the ASAMPSA2 project was to develop best practice guidelines for the performance and application of Level 2 probabilistic safety assessment (L2PSA), for internal initiating events, with a view to achieve harmonisation at EU level and to allow a meaningful and practical uncertainty evaluation in a L2PSA. The project has been supported and funded by the European Commission in the 7th Framework Programme.

Specific relationships with communities in charge of nuclear reactor safety (utilities, safety authorities, vendors, and research or services companies) have been established in order to define the current needs in terms of guidelines for L2PSA development and application. An international workshop was organised in Hamburg, with the support of VATTENFALL, in November 2008.

The L2PSA experts from ASAMPSA2 project partners have proposed some guidance for the development and application of L2PSA based on their experience, open literature, and on information available from international cooperation (EC Severe Accident network of Excellence – SARNET, IAEA standards, OECD-NEA publications and workshop).
At the end of the ASAMPSA2 project, the guidelines have been submitted to an international external review open to European nuclear stakeholders and organizations associated to the OECD-CSNI working groups on risk and accident management. A second international workshop was organized in Espoo, in Finland, hosted by FORTUM, from 7 to 9th of March 2011 to discuss the conclusions of the external review. This final step for the ASAMPSA2 project occurred just before the Fukushima Daïchi disaster (11th of March 2011). All lessons from the Fukushima accident, in a severe accident risk analysis perspective, could not be developed in detail in this version of the ASAMPSA2 guideline.

The first version of the guidelines includes 3 volumes:

- *Volume 1 - General considerations on L2PSA.*
- *Volume 2 - Technical recommendations for Gen II and III reactors.*
- *Volume 3 - Specific considerations for future reactors (Gen IV).*

The recommendations formulated in these 3 volumes are intended to support L2PSA developers in achieving high quality studies and focussing time and resources on the factors that are most important for safety.

L2 PSA reviewers are another target group that will benefit from the state-of-the art information provided.

This first version of the guidelines is more a set of acceptable existing solutions to perform a L2PSA than a precise step-by-step procedure to perform a L2PSA. One important quality of this document is that is has been judged acceptable by organizations having different responsibilities in the nuclear safety activities (utilities, safety authorities or associated TSO, research organization, designer, nuclear service company …).
Hopefully it can contribute to the harmonization of the quality of risk assessments.

Most activities related to the development of the guidelines were performed before the Fukushima Daïchi accident. Some complementary guidance for the assessment of severe accident risks induced by extreme events will be developed in a follow-up European project (ASAMPSA_E).

ASAMPSA2 PARTNERS

The following table provides the list of the 21 ASAMPSA2 partners.

1	Institute for Radiological Protection and Nuclear Safety	IRSN	France
2	Gesellschaft für Anlagen- und Reaktorsicherheit mbH	GRS	Germany
3	NUBIKI Nuclear Safety Research institute Ltd.	NUBIKI	Hungary
4	TRACTEBEL ENGINEERING S.A	TRACTEBEL	Belgium
5	IBERDROLA Ingeniería y Construcción S.A.U	IBERINCO	Spain
6	Nuclear Research Institute Rez pl	UJV	Czech
7	Technical Research Centre of Finland	VTT	Finland
8	ENEA - Ricerca sul Sistema Elettrico SpA	ERSE SpA	Italy
9	AREVA NP GmbH	AREVA NP GmbH	Germany
10	AMEC NNC Limited	AMEC NNC	United-Kingdom
11	Commissariat à l'Energie Atomique	CEA	France
12	Forsmark Kraftgrupp AB	FKA	Sweden
13	Cazzoli consulting	CCA	Switzerland
14	National Agency for New Technologies, Energy and the Environment	ENEA	Italy
15	Nuclear Research and consultancy Group	NRG	Nederland
16	VGB PowerTech e.V.	VGB	Germany
17	Paul Scherrer Institut	PSI	Switzerland
18	Fortum Nuclear Services Ltd	FORTUM	Finland
19	Radiation and Nuclear Safety Authority	STUK	Finland
20	AREVA NP SAS France	AREVA NP SAS	France
21	SCANDPOWER AB	SCANDPOWER	Sweden

ASAMPSA2 CONCEPT AND PROJECT OBJECTIVE(S)

Members of the European community who are responsible for fission reactor safety (i.e.plant operators, plant designers, Technical Safety Organisations (TSO), and Safety Authorities) have repeatedly expressed a need to develop best practice guidelines for the L2PSA methodology which would have the aim of both efficiently fulfilling the requirements of safety authorities, and also promoting harmonisation of practices in European countries so that results from L2PSAs can be used with greater confidence..

Existing guidelines, like those developed by the IAEA, propose a general stepwise procedural methodology, mainly based on US NUREG 1150 and high level requirements (for example on assessment of uncertainties). While it is clear that such a framework is necessary, comparisons of existing L2PSA which have been performed and discussed in (6th EC FP) SARNET L2PSA work packages, have shown that the detailed criteria and methodologies of current L2PSAs strongly differ from each other in some respects. In Europe the integration of probabilistic findings and insights into the overall safety assessment of Nuclear Power Plants (NPPs) is currently understood and implemented quite differently.

Within this general context, the project objectives were not to share L2PSA tools and resources among the partners, but to highlight common best practices, develop the appropriate scope and criteria for different L2PSA applications, and to promote optimal use of the available resources. Such a commonly used assessment framework should support a harmonised view on nuclear safety, and help formalise the role of Probabilistic Safety Assessment.

A common assessment framework requires that some underlying issues are clearly understood and well developed. Some important issues are:

- *the PSA tool should be fit for purpose in terms of the quality of models and input data;*
- *the scope should be appropriate to the life stage (e.g. preliminary safety report, pre-operational safety report, living PSA) and plant states (e.g. full power, shutdown, maintenance) considered;*
- *the objectives, assessment criteria, and presentation of results should facilitate the regulatory decision making process.*

The main feature of this coordination action was to bring together the different stakeholders (plant operators, plant designers, TSO, Safety Authorities, PSA developers), irrespective of their role in safety demonstration and analysis. This variety of skills should promote a common definition of the different types of L2PSA and so help develop common views.

The aim of the coordination action is to build a consensus on the L2PSA scope and on detailed methods deemed to be acceptable according to different potential applications. In any methodology, especially one developed from a wide range of contributing perspectives, there will be a range of outcomes that are considered acceptable. To represent this range, the project has initially considered a 'limited-scope' and a 'full-scope' methodology, based on what is currently technically achievable in the performance of a L2PSA. In *this respect it should be noted that what is technically achievable may not be cost effective, but for the purpose of this project it was taken to represent the upper bound of what may be considered 'reasonable'.*

- *'Limited-scope' methodology*

  A limited description of the main reactor systems, associated with standard data on the reactor materials, severe accident phenomenology and human actions reliability will lead to a simplified L2PSA. This 'limited-scope' PSA would include some indication of the main accident sequences that contribute to the risk of atmospheric releases due to a severe accident. For example, 'limited-scope' methods could apply to a L2PSA performed with a limited number of top events in the event-tree and mainly dedicated to identification of accident sequences which contribute to the Large Early Release Frequency (LERF). However such a L2PSA can include very detailed and complex supporting studies for the quantification of these top events. Engineering judgement may also help in the quantification of the top events of a limited scope L2PSA but the justification of this engineering judgement is considered as a key issue.

- *'Full-scope' methodology*

  This method can utilise sophisticated methods that consider the full range of reactor initial states and possible accidents together with detailed physical phenomena modelling and uncertainty analysis. As a consequence these L2PSAs allow identification of the most sensible sequences with their probabilities of occurrence (annual frequencies) and associated fission product release to the environment. These L2PSAs also allow identification of the uncertainty range of the results, weak points in the reactor system and operation, and the accident phenomena which would need further assessment to improve the relevance of the results. In such a wide ranging L2PSA, the quantification of sequences leading to large early release is not the only objective.

In reality, most current L2PSAs are at an intermediate level between these two approaches. However this representation was recognised as a pragmatic way to organise the coordination action because it allowed discussion on both simple and elaborated methodologies. It should be assumed that the need for application of an advanced method is established from the results obtained by an earlier simplified study in regard to specific requirements of the national safety authorities.

Evidently the second type of approach is time consuming and supposes a qualified dedicated team. Some applications do not warrant this level of detail and additionally some small stakeholders (especially utilities) cannot afford this level of commitment. The scope should be appropriate to the application and life stage under consideration and the detailed methods should represent an acceptable balance between best practice and available resources. L2PSA results obtained using differing approaches or for differing scopes should not be directly compared.

When developing the guideline it was found by the partners <u>that a clear distinction between limited-scope and full-scope was very difficult to formalize</u> and it has been decided to present in the report, for each issue, some recommendations that may refer to simplified or detailed approaches. The guidelines users are then supposed to develop themselves a strategy to build a consistent set of L2 PSA event trees and supporting analysis.

ASAMPSA2 CONTRIBUTION TO THE COORDINATION OF HIGH QUALITY RESEARCH

As explained above, in spite of the availability of existing L2PSA guidelines, the recent comparisons of existing L2PSA, performed and discussed in SARNET L2PSA work packages and also in CSNI workshops (Koln 2004, Petten 2004, Aix en Provence 2005), have shown large differences in practical implementation of L2PSAs and integration of probabilistic conclusions into the overall safety assessment of Nuclear Power Plants (NPPs).

The main contribution of the project should be the reduction of the lack of consistency between existing practices on L2PSA in the European countries.

The project had strong links with SARNET (Severe Accident Network of Excellence) and took into account all harmonization activities performed in other framework (IAEA,OECD-CSNI, WENRA, EUR, ANS, ASME …).

ASAMPSA2 COORDINATION MECHANISMS

The ASAMPSA2 organisation of the coordination action was based on three working groups:

- *A transverse group of End-Users, consisting of representatives of plant operators, plant designers,TSOs, safety authorities, R&D organisations, and L2PSA developers. The objectives of this group were:*
  - *to define and/or validate the initial needs for practical L2PSA guidelines for both 'limited' and 'full-scope' methods according to the different potential applications and specific End-User needs at the beginning of the coordinated action;*
  - *to provide a continuous oversight of the work of the Technical Group;*
  - *to verify that any proposed L2PSA guidelines can fulfil the initial and evolving End-User needs if required at the end of the coordination action;*
  - *to propose any follow-up actions in collaboration with the Technical Group.*

  This group was coordinated by PSI and includes representatives from IRSN, NUBIKI, TRACTEBEL, IBERINCO, VTT, AREVA GmbH, AMEC-NNC, FKA, CCA, VGB, FORTUM, and STUK.

- *A technical Group in charge for the development of a L2PSA guideline for Gen II and III reactors ;*

  This group was coordinated by IRSN and includes representatives from GRS, NUBIKI, TRACTEBEL, IBERINCO, UJV, VTT, ERSE, AREVA GmbH, AMEC-NNC, FKA, CCA, FORTUM, AREVA-SAS, and SCANDPOWER.

- *A technical Group in charge of the development of a L2PSA guideline (or prospective considerations) for some specific Gen IV reactors.*

  This group was coordinated by CEA and includes representatives from IRSN, AREVA GmbH, ERSE, ENEA, AMEC-NNC, NRG, and AREVA SAS.

The overall coordination of the ASAMPSA2 project was assumed by IRSN, including all administrative tasks and relationship with EC services.

SOME LIMITS OF THE ASAMPSA2 PROJECT

The number of issues that were addressed in the ASAMPSA2 project and discussed in the guidelines is very large. Nevertheless, these best practice guidelines have to be considered as a set of acceptable existing solutions to perform a L2PSA and not as a precise step-by-step procedure to perform a L2PSA.

The reader should be aware that issues such as external events, fire hazard, and ageing are not in the scope of this first version of the guideline, consistently with the Grant Agreement with the European Commission. For these topics, it was identified a needed for further harmonization activities during the End-Users final review. The Fukushima accident has then further highlighted their importance. Additional developments are expected to be included in any future updates of these guidelines.

VOLUME 2 CONTENT

GLOSSARY

AGR	Advanced Gas-cooled Reactor
AICC	Adiabatic Isochoric Complete Combustion
ALARP	As Low As Reasonably Practicable
APET	Accident Progression Event Tree
ASAMPSA2	Advanced Safety Assessment Methodologies: PSA Level 2
ASAMPSA_E	Advanced Safety Assessment Methodologies: Extended PSA
ASEP	Accident Sequence Evaluation Program
ATHEANA	A Technique for Human Event ANAlysis
ATWS	Anticipated Transient Without Scram
BDA	Boron Dilution Accident
BMMT	Base Mat Melt-Through
BSL	Basic Safety Limit
BSO	Basic Safety Objective
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAHR	Connectionism Assessment of Human Reliability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CESA	Commission Errors Search and Assessment
CET	Containment Event Tree
CHF	Critical Heat Flux
CHRS	Containment Heat Removal System
CICA	Important configurations of accident operation
CLI	Criteria for Limiting Impact
CMFD	Complex Multidimensional Fluid Dynamics
CPC	Common Performance Conditions
CREAM	Cognitive Reliability Error Analysis Method
CST	Condensate Storage Tank
DCH	Direct Containment Heating
DF	Decontamination Factor
DFC	Diagnostic Flow Chart
DNBR	Departure from Nucleate Boiling Ratio
EAM	Early Accident Management
ECCS	Emergency Core Cooling System
EFC	Error Forcing Context

EOC	Error of Commission
EOO	Error of Omission
EOP	Emergency Operating Procedure
ERO	Emergency Response Organisation
ESF	Engineered Safety Feature
ERMSAR	European Review Meeting on Severe Accident Research
FCI	Fuel Coolant Interaction
FEM	Finite Element Model
FLI	Failure Likelihood Index
FLIM	Failure Likelihood Index Methodology
FP	Fission Product
FSAR	Final Safety Analysis Report
HAEA (NSD)	Hungarian Atomic Energy Authority (Nuclear Safety Department)
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HFE	Human Failure Event
HHSI	High Head Safety Injection
HORAAM	Human and Organisational Reliability Analysis in Accident Management
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
I&C	Instrumentation and Control
IE	Initiating Event
ILRT	Integrated Leak Rate Test
INSAG	International Nuclear Safety Advisory Group
IPSART	International Probabilistic Safety Assessment Review Team
IRWST	In-containment Refuelling Water Storage Tank
IVMR	In-Vessel Melt Retention
IVR	In-Vessel Retention
L1PSA	L1PSA
L2PSA	L2PSA
LAM	Late Accident Management
LBLOCA	Large Break Loss of Coolant Accident
LDW	Lower Drywell
LER	Large Early Release
LERF	Large Early Release Frequency
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LHSI	Low Head Safety Injection

LMP	Larsson Miller Parameter
LRF	Large Release Frequency
LWR	Light Water Reactor
MCCI	Molten Core Concrete Interaction
MCS	Minimal Cut set
MDEP	Multinational Design Evaluation Program
MERMOS	Methode d'Evaluation de la Réalisation des Missions Opérateur pour la Sureté
NEA	Nuclear Energy Agency
NKS	Nordic Nuclear Safety Research
NPP	Nuclear Power Plant
NPSAG	Nordic PSA Group
NSC	Nuclear Safety Codes
OECD	Organisation for Economic Co-operation and Development
PAR	Passive Autocatalytic Recombiner
PDF	Probability Density Function
PORV	Power Operated Relief Valve
PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PSG	Probabilistic Safety Goal
PWR	Pressurised Water Reactor
QA	Quality Assurance
R&D	Research and Development
RAMP	Review of Accident Management
RCS	Reactor Coolant System
RHWG	Reactor Harmonization Working Group (within WENRA)
ROAAM	Risk Oriented Accident Analysis Methodology
RPV	Reactor Pressure Vessel
RST	Reference Source Term
RWST	Refuelling Water Storage Tank
SA	Severe Accident
SAD	Strategy, Action, Diagnosis
SAG	Severe Accident Guideline
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SAR	Safety Analysis Report
SASS	Severe Accident Safe State
SBO	Station Blackout

SBLOCA	Small Break Loss of Coolant Accident
SCG	Severe Challenge Guideline
SCST	Severe Challenge Status Tree
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SHARP	Systematic Human-Action-Reliability Procedure
SLI	Success Likelihood Index
SLIM-MAUD	Success Likelihood Index Method using the Multi-Attribute Utility Decomposition
SOARCA	State-of-the Art Reactor Consequence Analysis
SPAR-H	Standardised Plant Analysis Risk - HRA
SRV	Safety Relief Valve
TH	Thermo hydraulic
THERP	Technique for Human Error Rate Prediction
TSC	Technical Support Centre
TSO	Technical Support Organisation
UA	Unsafe Action
USNRC	United States Nuclear Regulatory Commission
VEAM	Very Early Accident Management
VF	Vessel Failure
WOG	Westinghouse Owners Group

1 INTRODUCTION

The objective of the present guidelines is to identify some best-practices regarding Level 2 Probabilistic Safety Assessment (L2PSA) development and applications. These guidelines propose a set of acceptable existing solutions to perform a L2PSA instead of a precise step-by-step procedure.

It has been established through a collaborative effort of 21 European organisations and funded by the European Commission in a perspective of harmonisation. At the beginning of the ASAMPSA2 project a survey and a workshop were organised to identify the L2PSA End-Users needs in terms of guidance. The conclusions [2] have been summarised in Volume 1 Appendix 9.5.

The present document takes into account some of the recommendations proposed during the external review and the workshop organized at the end of the project ([3], [4]).

1.1 THE 3 LEVELS OF PROBABILISTIC SAFETY ASSESSMENT

A definition of the 3 levels of Probabilistic Safety Assessment can be found in *IAEA* Safety Standard SSG-4 [1].

"PSA provides a methodological approach to identifying accident sequences that can follow from a broad range of initiating events and it includes a systematic and realistic determination of accident frequencies and consequences. In international practice, three levels of PSA are generally recognised:

> *(1) In Level 1 PSA, the design and operation of the plant are analysed in order to identify the sequences of events that can lead to core damage and the core damage frequency is estimated. Level 1 PSA provides insights into the strengths and weaknesses of the safety related systems and procedures in place or envisaged as preventing core damage.*
>
> *(2) In Level 2 PSA, the chronological progression of core damage sequences identified in Level 1 PSA is evaluated, including a quantitative assessment of phenomena arising from severe damage to reactor fuel. Level 2 PSA identifies ways in which associated releases of radioactive material from fuel can result in releases to the environment. It also estimates the frequency, magnitude and other relevant characteristics of the release of radioactive material to the environment. This analysis provides additional insights into the relative importance of accident prevention and mitigation measures and the physical barriers to the release of radioactive material to the environment (e.g. a containment building).*
>
> *(3) In Level 3 PSA, public health and other societal consequences are estimated, such as the contamination of land or food from the accident sequences that lead to a release of radioactive material to the environment.*

PSAs are also classified according to the range of initiating events (internal and/or external to the plant) and plant operating modes that are to be considered."

1.2 HOW TO USE THE ASAMPSA2 GUIDELINE

The guideline includes considerations and technical recommendations on most topics that should be addressed in a L2PSA. The technical recommendations are based on the authors experience (or open literature).They are supposed to help the L2PSA developers or reviewers to improve the quality of the L2PSA they consider.

The ASAMPSA2 guidelines have to be considered as a technical complement of the other existing "high level" guidelines like those of IAEA [1] or certain national guides. It proposes practical solutions and tries to define what could / should be done to obtain a state-of-the-art study. It was not the intention of authors to define any quantitative or qualitative safety requirement. This activity is the responsibility of the National Safety Authorities.

A wide group of institutions and authors has contributed to this document. The working modus of the project has been to assign the drafting of individual sections to those partners which had particular knowledge in the respective issue. This process naturally led to a compendium which tends to provide detailed elaborations and practical examples on each issue rather than giving practical examples of a complete L2PSA, where an in-depth investigation of each and every detail is neither necessary nor possible. Therefore, each section in this document to some extent represents state-of-the art considerations, but it is not likely that there is a single L2PSA existing which covers all issues in such detail.

The content of the guideline encompasses the very large number of issues that have to be examined in a L2PSA depending on:

- the number of initiators and core damage sequences from the L1PSA,
- the plant design and it's link with the physical phenomena that need to be considered,
- the L2PSA final application.

All issues may have not been discussed but the authors have tried to address as many topics as possible.

L2PSAs may support some important decisions regarding plant safety and management, for example:

- How far should reactors in operation (Gen II) be improved regarding the protection of population and environment (accident prevention, accident consequences limitations), especially in relationship with plant life extension decisions?
- Are the safety goals that have been assigned to a reactor been met?

In that context, the ASAMPSA2 partners have deemed it necessary to highlight discussions on the L2PSA applications. This explains why the guideline distinguishes between general considerations regarding L2PSA (including applications) and all technical issues.

All these considerations have been conducted by the ASAMPSA2 partners to separate the guidelines into 3 volumes:

Volume 1 - General considerations on L2PSA

This volume provides some general views on the management of a L2PSA, the existing background in many countries or international organisations and discusses the link between L2PSA results and their final application.

Volume 2 -Technical considerations for Gen II and III reactors

This volume provides information regarding specific methods to be used in a L2PSA (L1/L2PSA interface, accident progression event trees, release categories, human reliability analysis, etc) and recommendations on studies that need to be performed to support a L2PSA (physical phenomena, system behaviour, source term assessment).

Volume 3 - Specific considerations for future reactor (Gen IV)

This volume is more prospective but provides some interesting views on the applicability of existing L2PSA approaches for BWR and PWR to four Gen IV concepts.

Many variations are possible in the precise way of developing and use of L2PSA and the authors hope that this guideline will be useful either to efficiently develop new L2PSA or to improve existing ones.

The authors are aware that knowledge and methodologies may evolve in the near future but one should also consider that more than 30 years of research on severe accident are now available for severe accident risk assessment.

Robust L2PSA regarding decision-making should now be the norm and hopefully this guideline will contribute to this objective.

When using this guideline, the authors recommend successively examining the following points:

- What are the final applications of the L2PSA under consideration?
- Taking into account the final application and the plant design, what should the general features of the study be? Considerations:
    - Scope and level of detail,
    - Structure of the study: number of Plant Damage States, number of Release Categories, type of probabilistic tools to be used, etc,
    - Realism of the study: are conservative assumptions acceptable or not? Is the assessment of uncertainties needed or not?
- What should the precise content of the study be? Considerations:
    - List of physical phenomena that should be addressed,
    - List of systems that should be modelled,
    - List of human actions that should be modelled.
- How should each event be modelled? Considerations:
    - Do the assumptions reflect the state-of-the-art knowledge?
    - Are the dependencies between events correctly addressed?
- How relevant are the final conclusions of the study? Considerations:
    - What would be the best methodology for presentation of final results for the considered application?
    - How robust are the results regarding uncertainties and simplifications (if any)?
    - What emphasis should be placed on the L2PSA results, taking into account some imperfections?

The guideline is intended to provide useful information on all these issues for both L2PSA developers and L2PSA reviewers.

1.3 REFERENCES

[1] Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Specific Safety Guide N° SSG-4.

[2] ASAMPSA2/WP1/13/2008-13 PSI/TM-42-08-1 ASAMPSA2 – Results and Synthesis of Responses from the End-Users to the Survey on End-Users Needs for Limited and Full Scope PSA L2 14/77

[3] ASAMPSA2/WP1/D 1.4/2011-32 – IRSN-DSR-SAGR-11-249 - Minutes of the 7-9th March Helsinki workshop – E. Raimond

[4] ASAMPSA2/WP1/D 1.4/2011-31 - PSI TM-42-11-03 - ASAMPSA2 – Synthesis of the L2PSA End-Users Evaluations of the “Best-Practices Guidelines for L2PSA Development and Applications” S. Güntay

2 PROBABILISTIC ACCIDENT PROGRESSION MODELLING, QUANTIFICATION AND RESULTS PRESENTATION

2.1 OVERVIEW

The development of a L2PSA is an integrated process involving the following tasks:

1. Definition of plant damage states (PDS), the interface to the L1PSA.
2. Definition of the release categories that are the end states in the severe accident scenarios.
3. Development of the severe accident scenarios from onset of core damage, where the plant damage states can be seen as the initiating events and the end states are a set of defined release categories (RC).
4. Quantification of frequencies for the different accident scenarios and release categories.

The same structure of a L2PSA is now in-use for most studies as represented in the following figure:

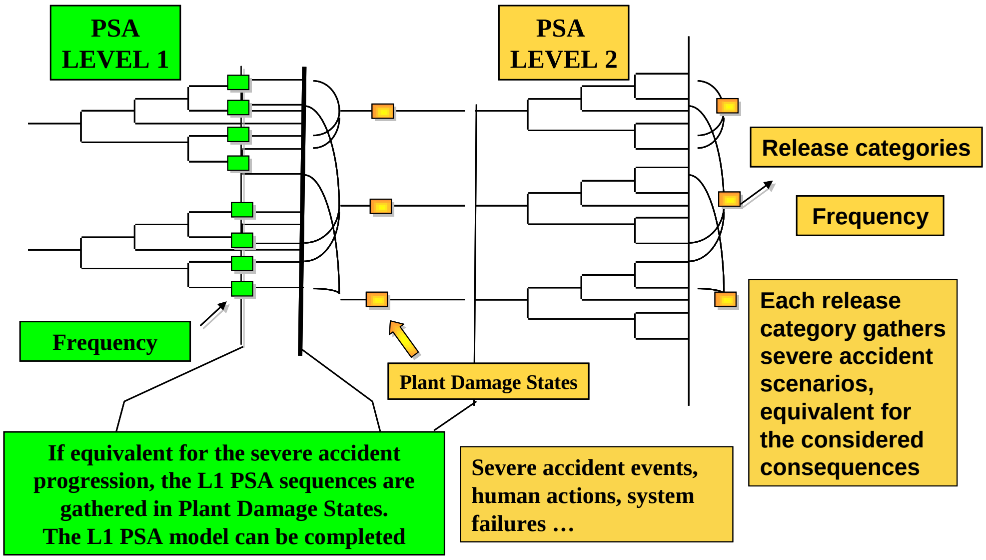


Fig. 1 Typical organisation of a L2PSA event tree modelling

The objectives and scope of the L2PSA project are of key importance to define the level of details needed in the PDS and RC definitions and in the accident progression modelling and analysis.

The event trees used to model the plants behaviour during severe accidents are called either Accident Progression Event Trees (APET) or Containment Event Trees (CET). Note that APET and CET are often used interchangeably with the same meaning. However CET was originally used for smaller level 2 event trees which focus on the containment failure modes, and APET represented event trees with a larger degree of modelling with specifics in phenomena, systems behaviour, and containment failure modes. The level 2 analysis is not restricted to the containment, but also addresses phenomena, operator actions etc. that are part of the accident progression. Therefore, the term Accident Progression Event Tree is used in this guidance document.

The probabilistic analysis is reliant on an information exchange with the analysis of phenomena, the containment structural analysis, the analysis of operator interactions (human reliability analysis – HRA), the systems analysis and the source term assessment.

The L2PSA probabilistic model is constructed based on the accident progression analysis results and form interrelated event trees – from level 1 to level 2 - and fault trees with basic events. The basic events represent function and system failures, operator action failures, and phenomenological events. The basic events and the logic between them describe the severe accident progression, release paths, and eventually characterise the source term for each release category (frequency, timing, dynamics, mixture and amount of various radioactive nuclides released to the atmosphere).

Probabilistic results are evaluated with the model and these results and relating uncertainties and sensitivity cases are defined in accordance with the objectives and scope of the study being undertaken. Usually, the model is developed and checked with both deterministic and probabilistic calculations. This may lead to changes in PDS/RC definitions and the sequences assigned to each PDS and RC. Several iterations may be needed before a final version is defined.

The next sections cover the following issues:

1. Development of the level 1 and level 2 interface,
2. Development of the severe accident progression scenarios,
3. Definition and characterisation of the release categories.

The presentation of results is mainly addressed in Volume1; chapter 5 and 6 of the guideline.

It is important to note that several technical solutions exist for the development of a severe accident event tree depending on the probabilistic and deterministic tools (and their capabilities) that are used. In particular,

- The L1PSA and L2PSA models can be “integrated” or “separated”,
- The different events can be modelled by different ways in the APET (split fractions, end-user functions, fault trees),
- The time dependency of the events can be modelled with different levels of details (specific methods as the Dynamic Reliability Methods (DRM) may be useful),
- The uncertainties can be assessed by very different techniques,
- The source term assessment can be integrated (or not) in the event tree,
- A common severe accident event tree can be used for all PDS or several event trees can be developed (the events are adapted to some PDS).

The following chapters are based on the different techniques in use among ASAMPSA Partners. L2PSA developers or reviewers have to make a clear distinction between:

- What should be done to obtain a L2PSA modelling without any imperfection,
- What can be done with the real functionality of the simulation (probabilistic and deterministic) tools but also the computational capacity and the ease to maintain the L2PSA modelization.

The reference tools for this chapter are RiskSpectrum, EVNTRE, SPSA and KANT (used by the ASAMPSA2 partners). Short descriptions of these tools are available in Volume 1, Appendix 9.2.

2.2 DEVELOPMENT OF THE LEVEL 1 AND LEVEL 2 INTERFACE

The following sections consider several aspects of the transition from L1PSA to L2PSA. Some of the issues show up in more than one section because they influence more than one aspect. In order to support the reader, three basic principles are listed which have to be observed when building the interface from level 1 to level 2:

- Consistency: the endpoint of L1PSA and the initial point of L2PSA must be identical. There must be no gap between the two levels. (e.g. no undefined section of the accident sequence should exist between the RPV coolant level falling below a certain limit, and melting of fuel.)
- Completeness: the endpoint of L1PSA must contain all information needed for L2PSA. (e.g. issues belonging to the level 1 phase should be analysed completely within level 1, and not be analysed within the level 2 tasks.)
- Unbiased: the set of information defining the endpoint of L1PSA should not be biased. (e.g. the information should not be "pessimistic", or "conservative". It should rather reflect best estimate, including uncertainties if needed. This may sometimes be difficult to achieve since the L1PSA often contains inherent caution, like pessimistic minimal systems requirements for avoiding core melt) ; this principle is especially important when L2PSA is used for risk ranking approaches.

2.2.1 Interface development overview

An interface between a L1 and L2PSA is needed to establish the connection between the L1PSA event tree model and the L2PSA event tree model. The end states in the level 1 model must have at least the degree of resolution that is required by the entry states to the level 2 event tree model. The interface is defined by plant damage states (PDS) which are sometimes also called accident classes. The PDSs are the equivalent of initiating events in the level 1 event trees, i.e. they are the initiating events in the level 2 event trees.

Accident sequences from L1PSA are grouped together into PDSs in such a manner that all accidents within a given PDS can be treated in the same way. Each PDS represents a group of level 1 accident sequences that have similar characteristics of importance for severe accident scenarios, e.g. accident timelines and generation of loads on the containment, thereby resulting in similar severe event progression and radiological source terms. Attributes of the accident progression that will influence accident chronology, the containment response, or the release of radioactive material to the environment, should be identified. These attributes of the PDSs provide conditions for the performance of severe accident analysis.

The plant damage states have to characterise the parameters that are needed to describe the sequence in the L2PSA analysis and those that influence the accident progression and source term. Additional parameters may be defined that are of interest to track back from the level 2 results through the level 2 and 1 models to identify important elements in the level 1 domain contributing to interesting level 2 results.

The development of the interface between the Level 1 and L2PSA parts includes the following tasks:

- Familiarisation with any existing L1PSA,
- Definition of plant damage states,
- Extensions, remodelling of level 1 part (e.g. revisiting initiating events analysis for possible better characterization of initiating events or inclusion of events screened out from a level 1 perspective. different requirements in terms of capacity, time available etc. means that other or different

functions may need consideration – more time available for operator actions, one instead of two trains, use of a function that was not available in a level 1 perspective, or need for adding branches that provide information on the status of containment systems mitigating the effects of severe accidents that is required for assignment of level 1 sequences to appropriate plant damage states),

- Grouping, assignment of the defined plant damage states to the level 1 event tree sequences,
- Human reliability assessment related to PDSs if needed for the extensions mentioned above.

The work is dependent on the availability of an existing L1PSA or not. An existing L1PSA needs to comply with requirements for L2PSA (scope, objectives). It is also necessary for a L2PSA team to familiarise themselves with the existing L1PSA to develop a L1-L2 interface able to support treatment of dependencies between the L1PSA accident sequence analysis and the L2PSA Accident progression analysis.

Fig. 1 gives an idea of the overall model items from initiating events, through the level 1 event trees to the plant damage states, to the accident progression event trees, and ending in defined release categories.

Note that the work in developing the interface is an integrated activity which is together with the work in developing the level 2 event trees (APET) and defining the release categories. Several iterations may be needed before the three tasks are completed. This is similar to the level 1 work in performing the accident sequence analysis comprising identification and grouping of initiating events, and developing the level 1 accident sequence progression.

Note that if L2PSA is performed without the need for producing explicit level 1 results, an interface between the different levels may not need to be explicitly defined. The model will consist of event trees where the entry states are the initiating events and the end states are release categories. However, even in this case it may be useful to structure each logical event tree model into a set of linked sub trees.

2.2.2 Definition of plant damage states

The general definition of core damage is the heatup of the reactor core (nuclear fuel) to the point where prolonged oxidation and severe fuel damage occur. But the exact definitions used for core damage vary between different PSAs (or depending of accident sequences in one PSA), one PSA may define onset of core damage as the time in point when the fuel is uncovered, whereas another uses a cladding temperature at 1204°C as the criteria for core damage. The definition is used in calculating success criteria (time available, water feeding rates and amounts needed etc.) to avoid core damage. It is important to have a consistency between level 1 and level 2 with regard to the core damage definition and it is also important that the core damage definition reflects the objectives and use of the PSA. Examples of Core Damage criteria are:

- When core is uncovered,
- When core exit temperature reaches a specified temperature,
- When cladding fails,
- When radioactive releases from the core start,
- When core material becomes molten.

L1PSA sequences are in many cases basically divided into one success state (OK) and one failure state (core damage - CD). There are also examples where more differentiated core damage states definitions are used in a L1PSA, and where the core damage states brings some further information, as in the following example of core damage states used in many Swedish L1PSAs:

- OK,
- CD1 – Sequences with reactivity control failure,
- CD2 – Sequences with injection/feeding failure,
- CD3 – Sequences with long term cooling failure.

However, even the more detailed core damage states above are not enough for the L2PSA where more information is needed to define the interface towards the level 2 event trees properly.

The L1PSA has a large number of sequences with core damage. These sequences need to be grouped into specific Plant Damage States (PDS) to make the subsequent accident progression analysis manageable. This will then be the link to the level 2 Accident Progression Event Trees.

Modelling of the level 2 part requires that the Plant Damage States (PDS) are defined in terms of the attributes that would influence the way that the accident progresses to challenge the containment integrity and the release of radioactive material to the environment. The attributes shall bring enough information to act as initiating events to the level 2 event trees and support both the development of the accident scenarios and eventually the use of a relevant set of release categories. It will also allow for tracking of certain information about initial states and functions/systems states from the level 1 part through to the release categories. It is thus necessary to identify these characteristics and define the appropriate set of plant damage states to be assigned to the sequences.

Given core damage, the core damage states usually are grouped according to some major criteria. One major criterion is the containment status, e.g. a split into one group where radioactive material is released from the reactor coolant system to the containment, and another group where the containment is either bypassed or ineffective. It is therefore necessary for the plant damage states to identify the containment status (e.g. intact and isolated, intact and not isolated, failed or bypassed) and, for bypass, the type and size of the bypass (e.g. interfacing system loss of coolant accident (LOCA), steam generator tube rupture). If the reactor building/secondary containment is likely to have a major influence on the source term, then its status is also defined by the PDS.

Other attributes that need to be considered in characterising the PDSs are plant failures in the L1PSA that could influence either the containment challenge or the release of radioactive material:

- The type of initiating event can, for example, affect the rate of discharge of fluid to the containment, progression of the core melt and hydrogen generation, and the timing of the release of radioactive material,
- The failure mode of the core cooling function affecting the timing of core melt,
- The extent of fuel damage, e.g. the damaged core is arrested (recovered) in the reactor pressure vessel without melt through. Note that recovery which avoids a PDS can be modelled in the level 1 part only, while recovery which does not avoid a PDS will have to be addressed in the level 2 part as one of the first questions in the accident progression model,
- The primary system pressure at the onset of core damage and the status of safety/relief valves and other components that could change RPV pressure before failure of the RPV lower head. RPV pressure at the time of lower head failure is important as it may influence the mode of debris discharge to containment. This, in turn, could present a challenge to containment integrity, for instance, regarding phenomena like high pressure melt ejection (HPME) and direct containment heating (DCH). RPV

pressure after the onset of core damage also influences the possibility of temperature and pressure induced failures of the RCS (e.g. creep rupture of piping and steam generator tubes, or thermal seizure of a safety/relief valve in the open position). The pressure will be influenced by the initiating event and the functionality of any depressurisation system,

- Determining the response of containment, it is also very important to know the status of the containment's engineered safety features. This may have an impact on containment cooling, the removal of radioactive material, the mixing of combustible gases present, etc.

Table 1 provides examples of plant damage state attributes. The list is based on [20] with some modifications.

Table 1 Example of Plant Damage State Attributes

Initiating event:	Large pipe LOCA Small pipe LOCA Stuck-open safety/relief valve Transient/LOCA (LOCA may increase temperature and amount of air-borne radioactivity in the containment or drywell) ATWS Bypass event (Interfacing systems LOCA, or steam generator tube rupture)
Physical Properties of the primary system and the containment:	RCS pressure at core damage -High (relief valves are challenged) -Medium (above low-pressure coolant injection head) -Low (including method of depressurisation) Availability of water below RPV Containment atmosphere (air filled / inerted) For BWR, also PS function and condensation pool temperature
Status of safety systems (characteristics of the sequence history):	All injection fails to start (no injection, early damage) Coolant injection initially successful, but recirculation cooling fails (later core damage) Steam generator cooling availability Support systems
Status of containment's engineered safety features:	Sprays (if any): - Operate at all times - Fail on demand - Initially operate, but fail later -not used switchover to recirculation cooling Suppression pool (if any): - Effective at all times - Ineffective (pool drained or bypassed early) - Bypassed late Fan coolers (if any): - Operate at all times

	- Fail on demand - Fail late Venting /Filter systems - Operate at all times - Fail on demand - Fail late Hydrogen mixing/ Status of recombiners/ igniters
Containment status at time of core damage	Intact and isolated at the onset of core damage Intact, but not isolated at the onset of core damage Structural failure or enhanced leakage (with indication of size and location of leakage) Bypass (e.g. V-LOCA or SGTR)
Status of secondary containment (reactor or enclosure building):	Intact and isolated at the onset of core damage Intact, but not isolated at the onset of core damage Structural failure or enhanced leakage
Time of core damage	This is related to the onset of release. It is important that early and late release sequences can be identified in the APET. It is also important to consider nominal leak contributions when describing the release categories, see chapter 7.

The original L1PSA is not likely to directly support assignment of plant damage states to the existing sequences. The resolution is too low for L2PSA. The end states may sometimes also be too conservative and lacking some mitigating systems or interactions that may prevent core damage, or at least prevent large core damage.

The final set of defined plant damage may therefore require additional modelling in the level 1 event trees, in the level 2 event trees or in bridge trees. The first approach means that the level 1 event trees are extended, e.g. to include top events addressing the availability of the containment systems. Another way is to model all the systems (containment and other mitigative systems) in the accident progression Event Trees, although care is then needed to ensure that correlations with the Level 1 sequences, such as dependencies on common support systems, are maintained. Yet another way is to use bridge trees that essentially act as extensions to the Level 1 trees.

2.2.3 Discussions of specific areas

The subsections below discuss the following specific areas:

- Primary system cooling recovery,
- Plant damage states with containment bypass,
- Extension to other initiating events,
- Extension to other power operating states,
- Final selection of plant damage states,
- Plant damage states for an existing L1PSA,
- Screening of initiating events, sequences and plant damage states.

2.2.3.1 Primary system cooling recovery

Cooling of the fuel to avoid melt through of the RPV, but not necessarily to avoid core damage, may be possible. This can be the case if cooling is recovered or if there is a possibility to find success criteria for certain functions and systems that, when successful, lead to a situation where a damaged core is arrested in the RPV. This kind of recovery can be treated by additional modelling in the level 1 event trees or bridge trees that allow this information to be part of the PDS definitions. It can also be part of the APET questions asking if cooling can be recovered to limit the degree of fuel damage and thus avoid melt through of the reactor pressure vessel. The most common approach is to check for cooling recovery in the accident progression event trees.

2.2.3.2 Plant damage states with containment bypass

PDS with containment bypass have the potential for large releases into the environment because the containment is the last remaining barrier when core damage is going on. First of all, the PDS has to identify the mode of containment bypass. Common examples are steam generator tube rupture in PWRs or failure to isolate the reactor coolant loop in BWRs. Additional examples of attributes of importance for containment bypass cases are those that account for attenuation of radioactive material concentrations along the release pathway or affect the time of release, such as the initiating event type, the status of the emergency core cooling system (including failure time) and whether the leak pathway is isolable after a period of time or if it passes through water (e.g. steam generator inventory or flooded building). For leaks into the auxiliary building or an equivalent one, the status of emergency exhaust filtration systems with heating, ventilation and air conditioning and whether or not the leak is submerged could be significant and should be taken into account.

2.2.3.3 Extension to other initiating events

A PSA may originally have been developed for a less than full scope set of initiating events.

Extending the scope of initiating events addressed in the PSA, e.g. to include internal and external hazards, of course requires that the impact of such events on systems needed for severe accident mitigation be considered, including those supporting operator actions, as well as the impact on containment integrity need to be taken into account. This may lead to the definition of new PDSs, e.g. in the case of earthquakes with potential for inducing containment failure. Extension into more initiating events requires the PSA analysis to consider the need of introducing new PDSs when existing PDSs cannot be used. See also below about revisiting the level 1 initiating events analysis to investigate if there is a need to add events that were screened out from further consideration in the level 1 study.

2.2.3.4 Extension to other power operating states

Initiating events occurring during different operating modes differs with regard to inventory, status of primary circuit and containment, e.g. open or closed RPV and containment. Additional PDSs thus need to be defined for low power and shutdown states if the differences have a major impact on plant behaviour in severe accidents and the resulting source term in the level 2 event tree end states. One example is to consider mid-loop operation in a PWR when the primary circuit inventory is low, or cases when the primary circuit is open (e.g. during RPV head removal or during refuelling).

2.2.3.5 Final selection of plant damage states

The number of plant damage states need to be manageable. There are examples of L2PSAs with hundreds of PDS and others with 20 PDS. It is up to the PSA project to define the accuracy needed in the specific case. A large number of PDS resulting from a preliminary set of attributes can be reduced. One approach is to combine similar PDSs and perform a bounding analysis selecting a representative sequence to characterise the PDS for the purpose of the Level 2 analysis. Another approach is to use a frequency cut-off as a means of screening out less important PDSs. A careful screening is required prior to introducing a frequency cut-off criterion at the PDS level. This is especially the case when dealing with PDSs that could involve large and early releases of radio nuclides to the environment. In any case, the selection process should take into account the degree of variability and uncertainty introduced in the L2PSA by the PDS grouping and how this affects the specific objectives of the PSA.

2.2.3.6 Plant damage states for an existing L1PSA

The L2PSA may be an extension of a L1PSA performed without plans for a Level 2 or 3 analysis. Aspects relevant to definition of PDSs are then unlikely to have been considered in the Level 1 study. For example, L1PSA may not address the status of containment systems or other systems which do not directly affect the determination of core damage (i.e. they do not contribute to the success criteria for preventing core damage). In such cases, the L1PSA needs to be expanded to account for the missing aspects in the definition of PDSs. One method for doing this is to update the level 1 event trees by adding the required functions and sequences making the sequences matching the PDS definitions. Another method is to develop separate 'bridge event trees' which link to Level 1 system models. These bridge trees will make sure that every sequence will match a defined PDS. Both methods may be applied in parallel.

2.2.3.7 Screening of initiating events, sequences and plant damage states

Screening may be applied for initiating events, sequences and plant damage states.

Initiating events may have been screened out in a L1PSA if they had frequencies low enough to be insignificant for the level 1 results. There is a need to revisit a level 1 initiating events analysis and check if any screening was applied. Initiators that were screened out from a level 1 perspective, but are important to consider for level 2, are then identified and added to the list of initiators to be considered in the L2PSA.

Screening may also be applied in the interface between level 1 and 2. Sequences and plant damage states can be screened out from further consideration in the analysis. This may be especially relevant in the case of a separated modelling approach (see chapter 2.4). It will then give focus to the dominating contributors to be used as input to the level 2 event trees. The screening threshold of the core damage sequence or cut set frequencies should be so low that the dismissed sequences constitute only a minor fraction of the sequences taken into account.

Screening of sequences and PDS can also be applied when using the integrated modelling approach and may reduce calculation times. It is ensured that screened out sequences are not linked to any level 2 event tree.

However, the integrated approach (see chapter 2.4) with event tree linking allows explicit consideration of all dependencies in the integrated model during the quantification of the release category frequencies.

It should be especially noted that screening criteria need to be checked, and the screening may be revisited when area events (e.g. internal flooding and internal fire) and external events are added to the PSA. The reason for this is that area events and external events are likely to affect the conditional failure probabilities of functions and systems because they increase the failure probability of components affected by them.

2.2.4 Assignment of PDS to the L1PSA sequences

The plant damage states are assigned to the sequences in the Level 1 trees and any bridge trees.

This activity can be supported by the use of a decision tree representing rules for the assignment of PDS to sequences. There may be practical difficulties in setting up and applying such rules because the L1PSA may not always be able to deliver sufficient information. For example, a L1PSA may find that the failure of at least three out of four systems leads to core damage. However, it is normally not further elaborated and quantified whether three out of four or four out of four systems have failed. For L2PSA accident progression it may be important to know whether one out of four or zero out of four systems are still available. This issue is further addressed in the section “Level 1 Extensions” below. A practical solution (however not always satisfying) is to follow a pessimistic approach (i.e. assuming the failure of all four systems) which is justified for highly redundant (but not diverse) systems.

Fig. 2 and Fig. 3 show an example of such a decision tree; a simplified Level 1 event tree which only includes major systems and initiating events. Every sequence is checked against the rules and the corresponding PDS is identified.

Fig. 2 shows the decision part with the headings representing every question that need to be answered to identify the PDS to be assigned to each specific sequence. Fig. 3 shows the resulting characteristics of each specific PDS matching the answers. This process makes it traceable to evaluate and assign PDS and also shows which combinations of attributes are of interest.

Accident Scenario													PDS
IE	PS function	ATWS	Overpressure of vessel	Injection to vesssel	Forced de-pressu-risation	323	Containment cooling	Residual heat removal 321	Residual heat removal 244	FILTRA activated	Contain-ment intact	Contain-ment at-mosphere	
Transient		no	no	yes	no	no	yes	N/A	N/A	no	intact		OK
							no	yes	N/A	no	intact		OK
								no	N/A	activated	intact	steam	OK-S1-A
										no	no	N/A	OK-S1-B
				no	yes	yes	yes	N/A	N/A	no	intact		OK-323-T
							no	yes	N/A	no	intact		OK-323-T
								no	yes	activated	intact	steam	ST824
										no	no	N/A	ST834
									no	activated	intact	steam	ST623
										no	no	N/A	ST633
						no	N/A	N/A	N/A	no	intact	nitrogen	ST211
												air	ST212
					no	no	N/A	N/A	N/A	no	intact	nitrogen	ST111
												air	ST112
			yes	no	no	no	N/A	N/A	N/A	activated	intact	steam	ST121
										no	no	N/A	ST131
		yes	no	yes <10%	yes	N/A	N/A	N/A	N/A	activated	intact	steam	ST224
										no	no	N/A	ST233
					no	no	N/A	N/A	N/A	activated	intact	steam	ST123
										no	no	N/A	ST133
				yes>10%	yes	N/A	N/A	N/A	N/A	activated	no	steam	ST233
					no	no	N/A	N/A	N/A	activated	no	steam	ST133
				no	yes	N/A	N/A	N/A	N/A	no	intact	nitrogen	ST211
												air	ST212
					no	no	N/A	N/A	N/A	no	intact	nitrogen	ST111
												air	ST112
			yes	no	no	no	N/A	N/A	N/A	activated	intact	steam	ST121
										no	no	N/A	ST131
LOCA	OK	no	no	yes (top)	N/A	N/A	yes	N/A	N/A	no	intact		OK
		yes (botten)	yes (other)				no	yes	N/A	no	intact		OK
								no	N/A	activated	intact	steam	OK-S3-A
										no	no	N/A	OK-S3-B
				no (top)	N/A	yes	yes	N/A	N/A	no	intact		OK-323-T
							no	yes	N/A	no	intact		OK-323-T
								no	yes	activated	intact	steam	ST824

Fig. 2 Decision tree part 1 for assignment of plant damage states to accident sequences

PDS		PDS Characteristics									
	Note	Time-point	Vessel pressure	Contain-ment pressure	Contain-ment cooling 322	Contain-ment spray 322	Alterna-tive cooling 244	FILTRA	Contain-ment	Atmos-phere	316
OK											
OK											
OK-S1-A											
OK-S1-B											
OK-323-T											
OK-323-T											
ST824		MS	L	L	N	N	yes	A	OK		V
ST834		MS	L	L	N	N	yes	N	B		V
ST623		S	L	L	N	N	N	A	OK		V
ST633		S	L	L	N	N	N	N	B		V
ST211		T	L	L	X	X	X	N	OK		K
ST212		T	L	L	X	X	X	N	OK	L	K
ST111	1	T	H	L	X	X	X	N	OK		K
ST112	1	T	H	L	X	X	X	N	OK	L	K
ST121	1, 3	T	H	L	X	X	X	A	OK		K
ST131	1, 3	T	H	L	X	X	X	N	B		K
ST224	1	T	L	L	N	N	X	A	OK		V
ST233	1	T	L	L	N	N	X	N	B		V
ST123	1	T	H	L	N	N	X	A	OK		V
ST133	1	T	H	L	N	N	X	N	B		
ST233	1	T	L	L	N	N	X	A	B		V
ST133	1	T	H	L	N	N	X				
								A	B		V
ST121	1, 2	T	L	L	X	X	X	N	OK		K
ST212	1, 2	T	L	L	X	X	X	N	OK	L	K
ST111	1	T	H	L	X	X	X	N	OK		K
ST112	1	T	H	L	X	X	X	N	OK	L	K
ST121	1, 3	T	H	L	X	X	X	A	OK		K
ST131	1, 3	T	H	L	X	X	X	N	B		K
OK											
OK											
OK-S3-A											
OK-S3-B											
OK-323-T											
OK-323-T											
ST824		MS	L	L	N	N	yes	A	OK		V

Fig. 3 Decision tree part 2 for assignment of plant damage states to accident sequences

Thermal hydraulic calculations are made in support of the PDS assignment and the level 2 accident progression modelling (event tree logic and release category definitions).

Integrated thermal hydraulic codes, used for L2PSA like MAAP, ASTEC and MELCOR, that include thermal hydraulic models, are all simplified with respect to best-estimate or mechanistic codes. They can be used for these thermal-hydraulic calculations but also some codes like RELAP, ATHLET, CATHARE or some plant simulators (for example the simulator SIPA, which includes CATHARE2 and models of most NPP systems, was used by IRSN for the 900 MWe PWR L2PSA) can be used. These thermal-hydraulics calculations performed for L2PSA may also be useful to improve the L1PSA modelling and strong interaction between L1PSA and L2PSA teams for this stage is highly recommended.

Usually a limited number of representative sequences are chosen for each PDS; a deterministic model is developed and calculated. It is possible to choose one of the dominant sequences to represent the PDS in the accident progression analysis. There might be several sequences in the same order of frequency. In these cases, the most conservative (if clear) of the dominant sequences may be chosen as representative of the PDS, especially if the main objective of the study is to demonstrate that some criteria like Large early Release Frequency (LERF) is met. The choice of representative sequences is described in detail in chapter 4.

In the case where there are several equivalent (in frequency) sequences in one PDS associated to different thermal-hydraulics sequences, and if the objective of the L2PSA is to provide some realistic results, it may be sensible to create different PDS.

2.2.5 Integrated or separated probabilistic model

There are two approaches in developing the probabilistic model of a L2PSA:

- Integrated model,
- Separated model.

One key difference between the two approaches is the way how the interface transfer information from L1PSA to L2PSA. The necessary degree and precision of information transfer is defined by the needs of L2PSA analysis, and it has to be provided whatever the approach.

With regard to the interface between level 1 and level 2 and quantifications made for the plant damage state, the main difference between the two approaches is the mode of documentation and data transfer from level 1 to level 2.

There are different tools with different functionalities that can be used when applying integrated or separated modeling approaches.

In principle, any L1PSA tool can be used for integrated analysis. The tool must then in some way support linked event trees, i.e. transfer of scenario information at the end states in one event tree to a linked event tree where the end states in the first tree are the initiating events. RiskSpectrum is one tool that is used for development and analysis of integrated models.

For separated modelling, there are several special L2PSA codes like EVNTREE and KANT that are specifically designed to take care of the modelling of the level 2 part (accident progression after core damage). These codes requires that the plant damage states in the PDS, are quantified separately and the PDS frequencies are then combined with the conditional probabilities derived with the level 2 specific code to get the final release category frequencies. The current situation is that the special level 2 codes have special features, e.g. user defined functions that calculate the severe accident event probabilities. The different codes are described in some more detail in appendices to volume 1.

Note that development of the codes continues and functionalities needed to support specific modelling features, handling of dependencies, uncertainty analysis, simplification of input and review etc. may be added or improved in newer versions of the different codes.

First L2PSA were mostly concentrating on phenomenological issues (like hydrogen, core concrete interaction, etc.) which can be addressed without consideration of system restoration. In this case there is less need for integrated approach tools in handling dependencies between level 1 and level 2. However, if accident management or recovery within the L2PSA becomes an important subject, the knowledge about failed or intact systems (and causes of failure) is crucial and then there is more need for integrated approach tools or more developed interface that include information on systems.

2.2.5.1 Integrated model approach

The same computer tool is applied for L1PSA and level 2 and the model database contain all level 1 and level 2 information. The quantification of the end points (release categories frequencies) consider all information from initiating events in level 1 through level 1 event trees and fault trees, level 2 event trees and fault trees. Therefore, the level 2 event tree analysis can use all information available in the level 1 part of the analysis. This is helpful when the state of systems and components (e.g. depressurisation valves, or availability of electric power) is needed in the level 1 and level 2 part of the model. Some properties related to the integrated modelling approach and tools are listed below:

- L1PSA and L2PSA are combined in the same model database and have an explicit link – event tree linking is applied
- The L1PSA event trees have the plant damage states as the end points and these PDS also acts as ”initiating events” to the linked level 2 event trees. The number of PDS can be limited.
- The severe accident progression after core damage is modeled in the linked event trees using the same tool as is used for the level 1 part.
- The L2PSA part models the conditional events in the severe accident progression with a split up in different branches (severe accident events) depending on the plant damage state,
- The release category frequency quantification considers the complete scenario from the original initiating events in level 1 through the level 1 event trees, further through the APETs and to the RCs.
- Dependencies between L1 and L2PSA is considered in the calculation (MCS analysis and quantification), both success events and failed events in level 1 are considered ; one of the positive effects with the linked approach is that dependencies between L1 and L2PSA can be considered in an integrated analysis with a release category as the top event. The code will automatically (shall automatically) take care of situations with an already failed component in level 1 so that no credit is given again in the level 2 part and working equipment can be given credit for by letting the code remove cut sets containing failure of working equipment.

- Special attention has to be given to the use of minimal cut set quantification techniques in both point estimate and uncertainty analysis since especially the level 2 event trees may contain mutually exclusive events and events with large probabilities,
- It is important that the probabilistic code used when applying the integrated model approach considers success events (mutually exclusive events), so that the cut sets are checked with regard to any success functions in the sequences, and it removes cut sets that are invalid, i.e. if an event appears both as success and as failure in the same cut set ; there are limitations with the approach due to this requirement on independence between basic events. However, this is a very standard requirement when performing a PSA and the analysts are aware of this.

2.2.5.2 The separated model approach

The separated model has the level 1 and 2 parts of the full model portions in separate data bases and has no explicit link.

A specific interface has to be defined where the level 1 tool provides the necessary information which is needed by level 2. In practice this interface is a set of different PDSs (typically less than 100), each with characteristic properties (e.g. high pressure or low pressure sequences; with or without failure of containment isolation; etc.). The frequency or the distribution of frequencies is given for each PDS by the level 1 tool. In principle, this approach could also transfer any set of information about systems and components, operators, but with presently existing tools one might encounter practical difficulties in handling the related data in an exhaustive way (individual status of all NPP components cannot be transferred by PDS attributes). The L2PSA developer must define an appropriate limited set of information on systems through PDS attributes. Since the separated model applies a specific tool for L2PSA like EVNTREE or KANT, this tool can be tailored specifically to the needs of level 2. A particular topic is the possibility to calculate branch probabilities by user-defined subroutines.

The link between is still the set of PDS, but the quantification is made in steps. The level 1 part is calculated separately and the PDS are probabilistically represented by PDS frequencies and a list of PDS attributes. Some of the features of the separated approach are listed below.

- L1PSA has its own database consisting of the initiating events and the corresponding level 1 event trees
- The plant damage states are assigned to the end points in the level 1 sequences or bridge trees,
- The severe accident progression after core damage is modeled separately with a special L2PSA tool.
- The L2PSA tool input is the plant damage states including their characteristics that supports the modeling of the severe accident scenarios,
- The plant damage states in the level 1 part are quantified separately and the results are multiplied with the conditional probabilities in the level 2 part to calculate the release category frequencies.
- The level 2 sequence calculations are performed as multiplication of branch probabilities (with deterministic calculation of the plant state evolution after each event) and are not based on minimal cut set theory. All difficulties of the integrated approach using a L1PSA tool where the quantification is made with minimal cut sets can be avoided,

- User-defined functions can be used flexibly for complex quantifications (e.g. to calculate the probability of hydrogen combustion in function of containment atmosphere evolution),
- It is easy to set up many branching points, also with multiple branches,
- Sampling is straightforward in the case of Monte-Carlo studies.

Without direct linking of L1PSA event trees and L2PSA event trees in a single model, consideration of dependencies between the level 1 and level 2 parts need to be made in the context of the level 1 / level 2 interface. When using EVNTREE or KANT, it is part of the definition of the parameters that define the plant damage states. This may be more or less complicated than using the integrated approach: if the L2PSA includes a precise description of the NPP systems, all information on the systems status has to be provided through the PDS attributes and it may conduct to a long list of PDS.

Also, the aspect of actually looking at the dominating event combinations, from initiating event to release category using the separated model approach, means that complete information about the specific event combinations making up the different cut sets may be more difficult to obtain.

2.2.6 Treatment of dependencies between L1PSA1 and L2PSA

Certain functions or events in the level 2 part of the model have dependencies to the level 1 part of the model. A single event or combination of basic events representing or contributing to a failure in the level 2 part may already have occurred in the level 1 part. This has to be properly addressed in the level 2 part.

Human actions may have been modelled for level 1 purpose and the same or related type of actions is used in the level 2 part. Again, this information has to be carried over to the level 2 part, and be properly taken into account in the human reliability analysis of the L2PSA.

Dependencies may exist in the components represented by the functions in the level 1 and 2 event trees. Many dependencies of this type are considered by the PDS definitions and then the layout of the level 2 event trees given each specific PDS. However, since L2PSA starts per definition with a degrading core, most safety related systems did fail in the level 1 (except in the case where human failure explains the accident), so that only a very limited number of systems may be available at the interface from level 1 to level 2. This simplifies the modelling of the interface in general and of dependencies in particular.

The issue gets more complex if recovery of systems in the level 2 part of the analysis is to be considered. In that case it is necessary to transfer all required information to the level 2, such as the mode of component failures, repair times, availability of resources.

2.2.7 Mission times in L1PSA and L2PSA

Mission times have to be defined for the probabilistic system failure analysis. The conditional probability of system failure is higher for a system operated during a long duration. The present paragraph explains the difficulties that may be encountered.

Mission times to consider in a L2PSA are usually longer than in a L1PSA. Mitigating systems may need to be in operation for a long time period to limit the source term.

Mission times for each individual system should vary depending on each accidental sequence and each success criteria. It is important to use a correct mission time for each item of equipment, both for supporting Level 1 and 2 PSA quantifications and results presentations.

The modelling of systems mission times can become extremely complex if the mission time of each item of equipment is adapted to each accident sequence. By simplification, the mission times in the L1PSA are often defined globally as a maximised delay before obtaining a stable end state without core damage: a common mission time is often defined for all systems and all accidents (24 hours is generally used). A L2PSA has another perspective.

If the scope of the study is limited to containment failure modes, the analysis has to be carried on until all phenomena challenging the containment are over. In practice, most L2PSAs end with the issue of containment challenge due to core concrete interaction which may pressurise the containment or melt through the containment bottom within typical time scales of one to several day(s). For the cases where the containment is not challenged by these phenomena, it may be of interest to examine if long term degradation can occur, e.g. due to corrosion of metallic components, or due to a combined chemical and thermal and radiation attack on containment components, such as penetrations. However, this type of analysis is not yet state-of-the-art in L2PSA.

If the scope of L2PSA is to define radioactive releases to the environment (source terms), the mission time has to be extended after the containment failure until the release rate has become insignificant. In practice, this may be easy to define for sequences when the containment failed under pressure, creating a high peak release. However, if a gradual release occurs, e.g. during repeated containment venting cycles, it is not obvious when the analysis should be finalised.

The behaviour of some specific system may be questionable in severe accident conditions. For example, the long term operability of the sump recirculation, in severe accident conditions can be difficult to establish for Gen II reactors. In that case, conditional failure probabilities, possibly depending on the mission times can be suitable. In that case, the absence of severe accident conditions qualification will certainly be a dominant issue in comparison with mission time definition.

In conclusion of this chapter, the following recommendations can be made:

- A precise definition of mission time for each system (depending on the accident sequence, but although on the PSA part (level 1 or level 2)) is desirable but may be quite difficult to obtain in practice,
- Some simplifications can be established (in L1PSA and L2PSA),
- The impact of these simplifications on the final results and conclusions should be commented in the L2PSA documentation.

2.2.8 L1PSA modelling extension for L2PSA

The L1PSA extension includes the following activities:

- Revisiting the initiating events analysis,
- Identification of cases where functions have different requirements in terms of capacity, time available etc.

Revisiting the initiating events analysis is important for identification of events possibly screened out from further consideration in the L1PSA. Such events, especially if they lead to large release (e.g. of events that may have not been considered in L1PSA on a probabilistic criteria) may be of interest in a level 2 perspective and shall in such a case be added to the list of initiating events and an appropriate event tree model be developed. Revisiting the initiating events analysis is also important for the possible need of more detailed information about the initiating event. It might in L2PSA be important to know the location of a leak (e.g. the section of the containment where the leak is) while in L1PSA it is sufficient to know that there is a leak at all without further interest in the exact location.

More detailed requirements for L2PSA in comparison with L1PSA can have an impact on the fault tree logic and time available for operators.

For example, twos trains are required to avoid core degradation according to deterministic rules, but one train is sufficient to avoid large core degradation and melt through of the RPV according to realistic assessment. To keep the level 1 model intact, there is a need to define specific level 2 functions. These functions need to be administered properly making sure that a L1PSA analysis uses the level 1 success criteria and the L2PSA analysis uses the level 2 success criteria. This can be accomplished by the use of house events or similar solutions available in the tools applied for the analysis.

Credit may be given to systems that were not used in the L1PSA, because from L2PSA perspective, time available is longer and allows more time for manual action. Such cases can also be handled by defining new functions and an appropriate use of house events.

Being able to run analysis cases from both a L1 and L2PSA perspective with the same model (in case of the integrated model approach) requires that the different cases are managed by appropriate naming and analysis control.

Adding branches that provide information to appropriate plant damage states may be needed. This is dependent on the needs resulting from the attributes used in the definition of the plant damage states. It might be necessary to ask extra questions to assign the defined PDS to each sequence.

Having defined the plant damage states and the needs for adjustments in the L1PSA model, the fault trees and event trees are updated with respect to:

- Changes in success criteria (number of trains, credit for new systems, time for recovery operator actions, operating time of individual components in a L1PSA compared to a L2PSA perspective),
- Assignment of the plant damage states to the level 1 sequences,
- Inclusion of bridge event trees,
- Treatment of dependencies.

The modelling extensions are dependent on the properties of the probabilistic code used, e.g. if the code supports the separated and integrated modelling approaches.

Note that often there is a choice where different characteristics shall be considered, either in the level 1 part which is extensions of original level 1 model, or in bridge trees, or in the APET. However, for reasons of consistency it is recommended not to shift the distinction point between L1 and L2PSA. For example, human reliability analysis related to mitigation of core damage should not be performed in L1PSA, and considerations on the failure of containment systems occurring before core damage should not be performed in L2PSA.

In practice, L2PSA deterministic analyses may show that success criteria for the technical systems are less demanding than traditionally assumed in L1PSA. It is recommended to adjust the L1PSA accordingly. However, if this is not practical, L2PSA should separately identify such sequences which have been provided by L1PSA as core damage sequences, but where level 2 did not identify core damage. Observe that less demanding success criteria in one part of the model may change the success criteria in another parts of the model.

The extended model is likely to be larger, with more sequences than the original level 1 model.

It is recommended to use modelling techniques that makes it as easy as possible to quantify both level 1 and level 2 results for the overall level 1 and level 2 model, whether it is an integrated or separated model approach that is used. Comparison of the core damage frequency with the plant damage frequencies and release category frequencies is one of the methods for checking the model accuracy.

2.3 DEVELOPMENT OF THE ACCIDENT PROGRESSION EVENT TREES

2.3.1 Introduction

In L2PSAs, event trees are used to delineate the sequence of events and severe accident phenomena after the onset of core damage that challenge the successive barriers to radioactive material release. They provide a structured approach for the systematic evaluation of the capability of a nuclear plant to cope with core damage accidents.

The sequence analysis identifies the development of the accident after reaching a plant damage state and is the basis for the structure of the accident progression event trees and the related function/system fault tree models.

The events considered in the APET are of different nature:

- Physical phenomena,
- Systems behaviour (as primary, secondary, safety, severe accident mitigation ...)
- Operator actions,
- Containment behaviour / failure modes (leakages, structural response, increased leakage, failure).

The APET allows the description of severe accidents through L2PSA sequences, from a plant damage state (PDS), which consists of a grouping of L1PSA sequences (see 2.2) up to the plant final state after the accident, including possible containment failure. L2PSA sequences are grouped in turn into release categories (RC) which represent the most characteristic L2PSA result.

For each PDS, several accident scenarios are possible depending on the occurrence of the different events. Uncertainties can highly influence the relative probability of possible accident scenario paths. This possibility of multiple consequences analysis is the main interest of the L2PSA modelling in comparison with deterministic analysis that mainly focuses on a single accident path.

A sound knowledge of severe accident issues in general and on the plant specific accident progressions in particular is needed for setting up an APET. While general knowledge can be obtained by studying appropriate publications, plant specific information has to be acquired by particular analyses. The centre of such analyses is the calculation of a set of reference sequences with state-of-the-art integral thermal hydraulics codes. This issue is addressed specifically in the section on definition and calculation of representative sequences, see chapter 4.

2.3.2 Initiation of the development of L2PSA model

The modelling of the level 2 sequences highly depends on the plant characteristics and on the scope and objective of the PSA.

At the beginning of the development, the APET/CET structure can be based on a generic skeleton of event tree, depending on the plant type and encompassing as many events as may possibly contribute, and later be adapted according to the considered plant specific design.

It can be noted that understanding of the severe accident progression can be improved by the use of simplified diagrams as an intermediate step before creation of the level 2 event trees., (see the example below). Success block diagrams have been identified as useful for communicating the scenarios with plant personnel during the development of these scenarios.

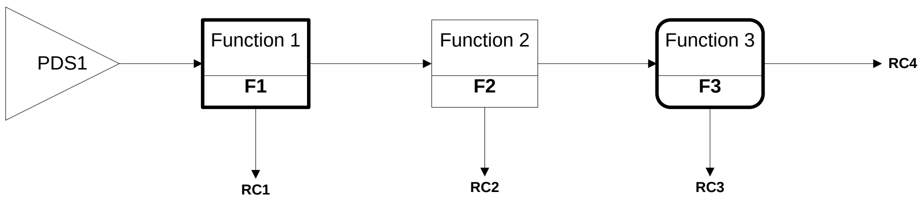


Fig. 4 Simplified diagram example - Success block diagram

The APET modelling approach may be influenced by the functionality of the probabilistic code used for the analysis (availability of multiple branch functions, event tree linking, run times for different kinds of modelling etc.) and also by the choice of separated or integrated modelling approach. But in any case, it must be checked that the modelling is suitable to the scope and objective of the PSA.

2.3.3 Global L2PSA event trees or adapted event trees to particular PDS

The same structure of a L2PSA is in-use for most studies as represented in the Fig. 1 above.

Nevertheless, some variations in the precise structure of the event trees are possible. In particular, to simplify the content of the event tree modelling, it can be useful to develop event trees adapted to some specific PDS.

For example:

- Nominal power and shut-down states of reactor can conduct to different event tree,
- PDS with high pressure in the vessel at beginning of core degradation can be treated by a specific event tree including depressurisation events.

Whatever the solution used to develop the event tree, the final consistency of the result has to be checked, especially when uncertainties analysis is introduced in the APET modelling (different event trees should conduct to the same RC and the final frequencies [mean, 5%, 50%, 95% percentiles] of each RC have to take into account all individual contributions.)

2.3.4 Organisation of APET in separated phases

The severe accident progression is generally separated into 3 phases:

- In-vessel core degradation phase,
- Vessel rupture phase,
- Ex-vessel phase.

The Fig. 5 below presents an example of an APET developed for a French PWR (IRSN).

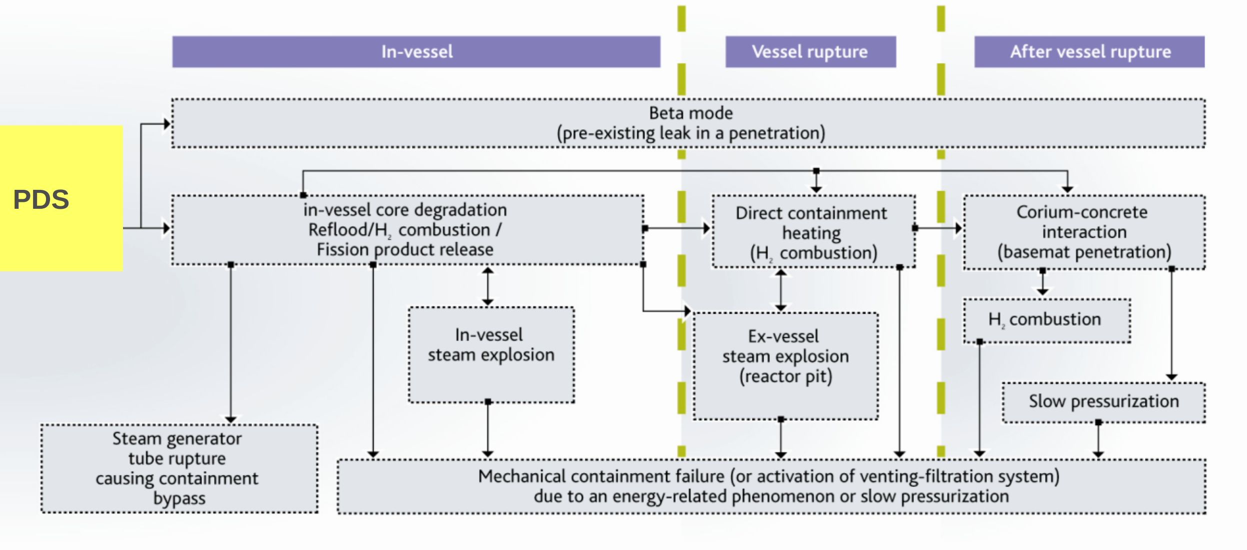


Fig. 5 Example of APET structure for a French PWR L2PSA (from IRSN)

2.3.5 APET events to be considered

As mentioned above, the events considered in the APET are of different nature:

- Operator actions,
- Containment behaviour,
- System behaviour,
- Physical phenomena.

The success or failures of operator actions are usually important for the L1PSA results. Both level 1 operator actions before core damage that relates to avoiding core damage, and level 2 actions relating to consequence mitigation functions, need consideration in estimating their weight on L2PSA final results.

L2PSA operator actions that are considered in the analysis correspond to the actions recommended in the Severe Accident Management Guide (SAMG), and other guides (e.g. guides regarding status of containment).

In particular, potential adverse effects of severe accident management actions should also be considered as part of the event tree logic. For instance, injection of water into a degraded core may be able to arrest the progression of a severe accident; however, there is also the potential for an energetic fuel-coolant interaction, fuel shattering and additional releases of steam, hydrogen and radioactive material [5]. There is also a risk that actuation of the containment spray system will lead to hydrogen explosion, since the hydrogen volume fraction will be increased by condensation of the steam in the atmosphere.

Operator actions and human error probabilities are addressed in detail in chapter 3.

The containment structural response (beyond the initial design) and its tightness is a key issue. Containment leakages can be pre-existing or due to severe accident loads. This issue is addressed in details in chapter 5.

The effect of the environmental conditions resulting from a severe accident on the survivability of components and systems credited within the Level 2 model should be assessed and, as appropriate, accounted for. Environmental impacts may include temperature, pressure, humidity and radiation conditions, as well as effects derived from energetic events (for example, short term temperature and pressure spikes or impulse loadings from detonations or steam explosions). The behaviour of safety systems in severe accident conditions has to be specifically addressed in L2PSA, since these systems may not have been designed to withstand such conditions. This issue is further addressed in chapter 6.

A list of the phenomena of importance that typically may be considered in a L2PSA is given below. The importance of these phenomena may depend on the plant type. A list of phenomena and events of importance in each phase of the accident progression is listed below. The phenomenological events are addressed in detail in Chapter 4.

IN-VESSEL CORE DEGRADATION

- Core degradation and fission product release from the core,
- Induced-RCS rupture including Induced-SGTR,
- Hydrogen production,
- Restoration of systems (for example: core-cooling, spray),
- Vessel cooling from outside,
- Consequences of in-vessel water injection (coolability, hydrogen production, RCS pressurisation, recriticality, …),
- Containment atmosphere composition (recombiners/igniter effect) and containment pressurisation,
- Containment venting,
- Hydrogen distribution/combustion (deflagration, detonation) and consequences due to pressure and temperature load (containment leak, failure or system degradation),
- Corium criticality,
- Relocation of core material into lower plenum,
- In-vessel steam explosion and consequences (leak in the RCS, vessel rupture, containment rupture),
- Vessel rupture (delay, break size, bottom penetrations, …).

VESSEL RUPTURE PHASE

- Direct Containment Heating, including H2 combustion and vessel uplift,
- Ex-vessel steam explosion,
- Corium criticality below vessel,
- Distribution of corium in lower containment rooms,
- Corium impact on containment penetrations.

EX-VESSEL PHASE

- Molten Core Concrete Interaction (MCCI)
- Corium coolability,
- Basemat lateral and axial erosion,
- Impact of water injection,
- Production of steam and non-condensable gases and fission product,
- H_2/CO combustion,

- Evolution of containment atmosphere composition and long term pressurisation,
- Containment venting,
- Corium attack on steel containment structures,
- Pool scrubbing,
- Melt propagation into ducts and channels,
- Temperature (local and global) load and consequences.

2.3.6 Dependencies between APET events

The probability of an event may depend on the values of physical variables. This probability will thus change if another event, occurring previously, changes the values of these physical variables.

The APET development must be optimised to tackle dependencies between events. For example, SARNET [10] recommended taking care in the APET organisation since the different potential consequences of the melt release into the reactor cavity (vessel rocketing, melt dispersal and direct containment heating, cavity failure, ex-vessel steam explosion, containment failure) which are strongly interdependent may only be assessed separately and successively.

The precise modelling of dependencies may be highly dependent of the level of details in the modelling of events:

- It is possible to model only dependencies between some macro-events by logical rules, for example, it can be considered that “in-pressure vessel rupture and DCH” is impossible if the event “RCS depressurisation by safety valve” is successful,
- It is also possible to model the dependencies between events by using some variables describing the plant systems status, for example the vessel pressure at vessel rupture can be calculated in the event-tree modelling and used in macro-event “containment failure by DCH at vessel rupture”. This approach is used for example by IRSN with KANT: the variables describing the systems status are updated after each event.

It is highly recommended to document how all dependencies are taken into account in the model. This should be of major interest to a reviewer when looking at the quality of the modelling.

2.4 REALISATION OF THE ACCIDENT PROGRESSION EVENT TREES

2.4.1 APET structure

As mentioned above, the APET need to consider the different phenomena and containment failure modes in each phase along with potential mitigating functions and operator actions, including dependencies between phases, e.g. hydrogen burning in an early phase will lower the risk for hydrogen explosion in succeeding phases.

Some PSAs develop a separate event tree for each phase and link them together to form the entire APET structure for a given plant damage state.

The way that this is done is very much dependent on the way that either approach is supported by the PSA tool being used.

There are special level 2 event tree codes that have various degrees of flexibility to define user-specific functions in the branches. These functions calculate the branch point probabilities taking into account the sequence specific information about the scenario that characterises the starting point of the Level 2 scenario, the PDS.

The decomposition into sub-issues makes the branch probability more traceable. It is important that the rationale used for the different branch probabilities are carefully documented. The assessments may be carried out separately and reported in support documentation with the results being used in the APET nodal questions or may be an integral part of the APET in the form of the decomposition event trees and fault trees mentioned above.

2.4.2 APET linked with source term calculation

Depending on the probabilistic tools in-use, it may be possible to introduce the source term calculation in the APET. Such an approach is possible with tools like EVNTREE, SPSA and KANT. In that case, fast-running source terms codes are used as End-User function in the event tree. Some details and examples are provided in chapter 7 and also in appendices in the volume 1 of the guideline.

It can be mentioned that the introduction of source term does not really modify the methodology used to develop the APET but only the post-processing of results (source terms results have to be managed and not only the frequencies of release categories).

2.4.3 How to model the different events?

2.4.3.1 Modelling overview

The modelling and quantification tasks are very much interrelated and dependent on the software tools used and the choice of an integrated or separated modelling approach. Note that there are tools that may better support one of the approaches, so the choice of software tools is closely related to the choice of a non-integrated or separated modelling approach.

The input to the actual creation of the event tree and fault tree model is the result of the interface definition, the accident sequence progression and the defined release categories. This means the update / extension of the level 1 model that may be needed as part of assigning plant damage states to the level 1 sequences, the creation of the level 2 event trees (including definition of event tree functions), links to system top event fault trees, entering of basic events representing the modelling of phenomena, containment failure modes, new fault trees for accident consequence mitigation systems, and eventually the assignment of the set of release categories to the level 2 event tree sequence end states.

Both integrated and separated approaches need to apply specific modelling techniques and use of functionalities in software to arrive at a correct treatment of the dependencies that allows the tracking of the relevant information from level 1 through level 2 in support of the presentation and interpretation of results.

2.4.3.2 Mission times

To be able to quantify the probability failure of a mitigation system or to define the time for which a physical phenomenon has to be considered, mission times have to be defined.

This choice of a mission time may be particularly important for a mitigation system, the design of which relies on redundancies. In the case of a twofold redundancy without a predominant common cause failure, the probability of the system failure is proportional to the square of the mission time.

Therefore, to be as realistic as possible, the mission time should be accident dependent and should correspond to a final end state for L2PSA (stable plant state, no more significant releases). The discussion on the definition of a final end state in various L2PSA is in the first part of the SARNET report [5]. The difficulties of such definition can be seen also from the discussion in section 2.2.7 in connection with the L1–L2PSA interface.

2.4.3.3 Use of fault trees to model containment failure modes

There are options in modelling of the containment failure modes. One approach is to use fault trees combining the conditions that are needed for a certain failure mode, with an event representing the containment failure mode probability given that the conditions are satisfied. The conditions may be represented by basic events with probabilities, e.g. the probability of a certain amount of hydrogen combined with the probability of a certain amount of oxygen, a certain amount of steam. This is illustrated in Fig. 6 and Fig. 7.

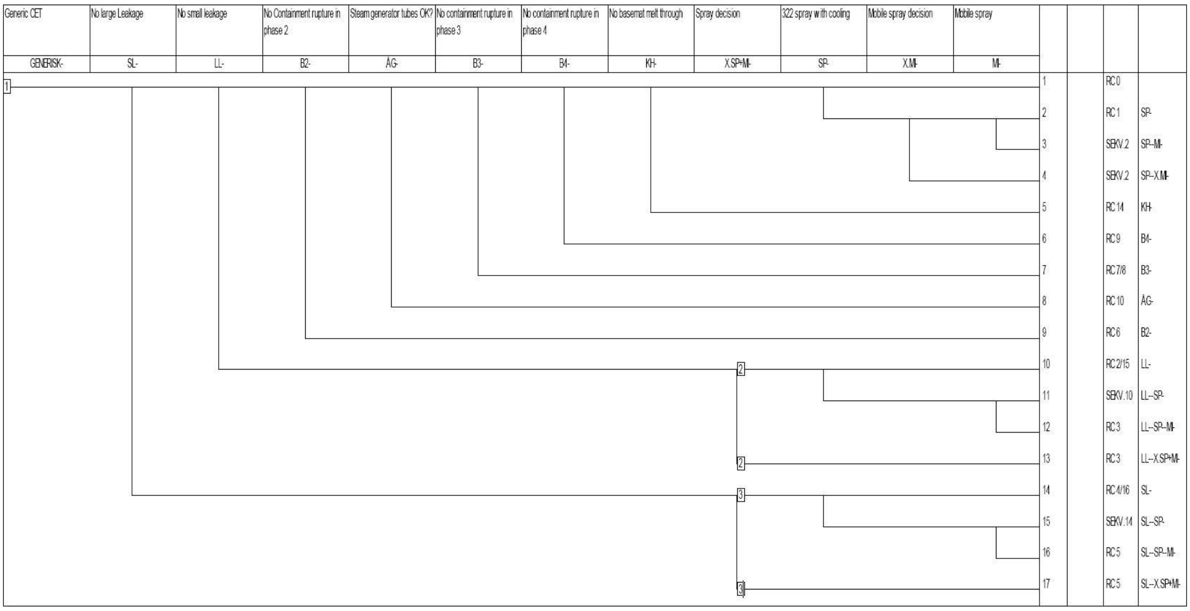


Fig. 6 Generic event tree with functions and containment failure modes.

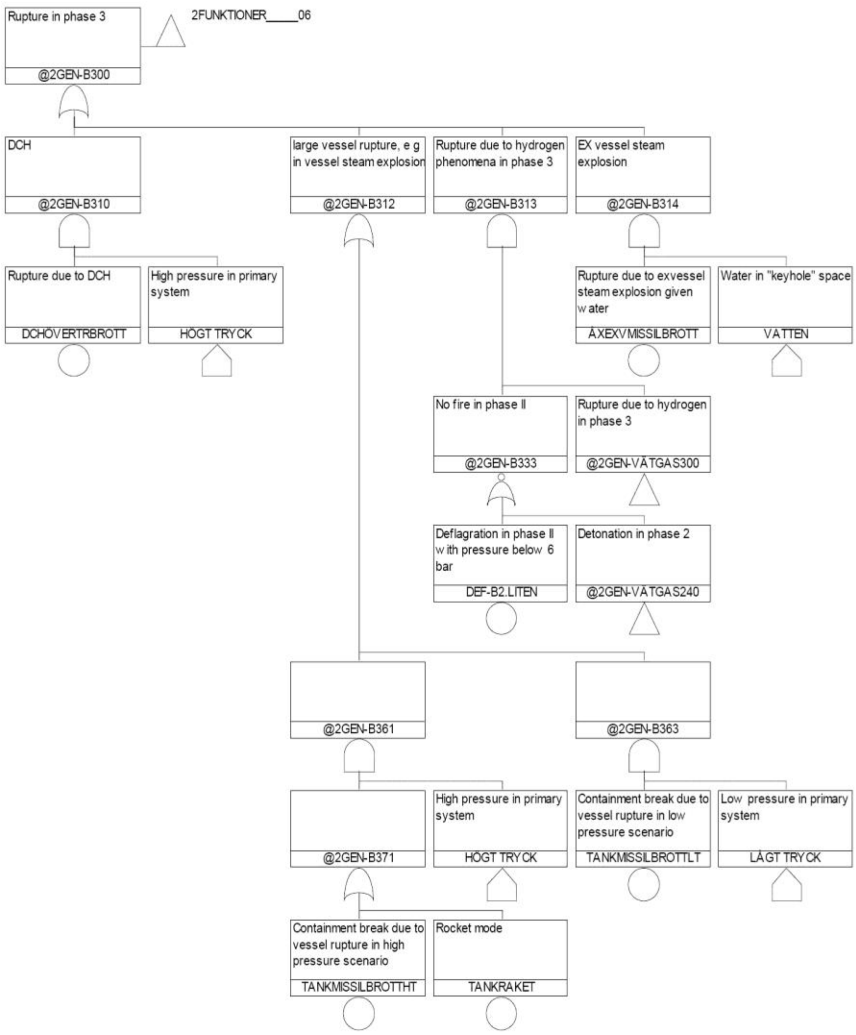


Fig. 7 Fault tree for function B3 in previous figure 6.

L1PSA codes like Risk-Spectrum provide a practical means to develop fault trees.

Similar logic can be developed by the use of decomposition event trees or user defined functions with L2 codes like EVNTREE or KANT.

2.4.3.4 Event Tree branching nodes

Another approach is to use event tree branches to model the same problem. In this approach the issues which are addressed by fault tree elements in the example above are covered by successive branching points in an event tree. There would e.g. be branching points for the primary pressure, for hydrogen issues and for RPV failure modes, followed by the branching point which determines the containment failure, depending on the previous branches.

2.4.3.5 Split Fractions in L2PSA event trees

The split fraction assignment is normally referred to as "degree of belief" of occurrence of an event (some people refer to expert judgment or subjective judgment) and it represents an analyst's quantitative measure of the likelihood of an event given the available supporting information. It is the process which was adopted in the NUREG-1150 study and is still used in some L2PSAs. In the quantification process, the outcome of the main

physical phenomena should now preferably be quantified by validated code calculations analysis including some sensitivity studies. For some cases, quantification can be left to engineering judgment.
Generic split fractions can be used but attention should nevertheless be paid to plant specific issues.

2.4.3.6 Decomposition Event Tree (DET)

The logic used to develop an appropriate branch probability can sometimes be made more traceable by decomposing the problem (e.g. does the containment fail due to hydrogen combustion) into a number of sub-issues (e.g. hydrogen concentration inside the containment and availability of ignition sources) according to the governing phenomena .
Such assessments may be carried out separately and reported in support documentation with the results being used in the APET nodal questions or may be an integral part of the APET in the form of decomposition event trees which are linked to the APET headings [5].
The codes EVNTREE and KANT have this feature of decomposition event trees.
This approach may also refer to the ROAAM method which tries to overcome the level of subjectivity by decomposing complex issues and making reference to the state of knowledge for sub-issues.

2.4.3.7 "User-functions" nodes or modelling

For some events, it may be necessary to develop models which are more complex than only branching nodes with conditional probability. In that case, the development of user functions provides enough flexibility to allow any modelling whatever its complexity.
User-functions can for example be developed to introduce some precise information on the physical evolution of some key parameters during the accident progression (for example, the in-vessel pressure, the containment gas composition and pressure evolution). These models shall be qualified by comparison with appropriate experiments or with sophisticated computer code results. They should:

- Give a "best-estimate" evaluation of a physical phenomenon and of its consequences,
- Take into account uncertainties,
- Be very fast running,
- Replace the sophisticated codes used for severe accident with appropriate accuracy.

This provides many possibilities to correctly handle the dependencies between the events. In terms of principle, the User-Function allows a "deterministic" calculation of the plant evolution on each accident progression path. This increases the confidence in the final result and may reduce the approximation in the modelling in comparison with a model obtained only by simple branching nodes.
The inconvenience of this approach can be the complexity of the overall model.
Codes like EVNTRE or KANT provide flexibility in the development of User-Functions.

2.4.3.8 Response surface method

The response surface method uses the design of experiments to minimise the number of calculations to be performed with some sophisticated thermal hydraulics code to develop some simple deterministic models (User functions) for the APET as described in the previous chapter. The following approach has been used by IRSN for the French 900 MWe L2PSA [6].

First step: choice and hierarchy of upstream variables

- Firstly, experts have to propose a list of upstream variables (presumed to be influential on the phenomenon to be quantified); for each variable, a possible interval of variation is defined,
- A first design of experiments (list of calculations) is defined; each variable can only take the extreme values of its variation interval,
- For each combination of upstream variables values obtained in design of experiments, a calculation of downstream variables (main results regarding the phenomenon to be quantified) is realised with the sophisticated code (the number of runs with a sophisticated code depends on its execution speed),
- Statistical analysis is achieved for each downstream variable: a hierarchy between the upstream variables is established in function of their influence on the result,
- The upstream variables that have only a small influence may be eliminated,
- At the end of this step, the initial choice of upstream variables is adjusted.

Second step: elaboration of response surfaces for each downstream variable

- A second experience plan is defined with more possible values for each upstream variable,
- For each combination of variables values obtained in the experience plan, a calculation of downstream variables is realised with the sophisticated code,
- For each downstream variable, the best mathematical function (response surface) of upstream variable is constructed with statistical analysis (minimal regression). For the construction of the response surface a physical and a statistical approach must be associated,
- The statistical uncertainties of the surface response are estimated.

Third step: check the response surfaces accuracy

- Other calculations with sophisticated codes are made with new combinations of upstream variable values,
- For each combination and each downstream variable, result of response surfaces is compared to the sophisticated code result,
- The first and second step may be completed if the accuracy of the response surface is not sufficient.

2.4.3.9 Example for hydrogen issue in APET

APETs consist of many branching points, and each branching point can have more than two branches. Therefore large APETs practically cannot be displayed completely in diagrams or reports. This is similar to the difficulty in L1PSA to present a complete fault tree.

In order to present at least partly a section of a large APET, an example is provided below from a PSA. It relates to the hydrogen issue in the in-vessel phase. The question to be answered is the probability that flame acceleration and deflagration-detonation-transition (DDT) will occur.

The plant is equipped with passive autocatalytic recombiners (PARs). A few (about 15) complete accident progression analyses with MELCOR 1.8.6 have been performed. The following section of the APET shows how uncertainties of the MELCOR analyses, and issues which do not exist in MELCOR (ignition sources) are combined to a complete assessment.

The section of the APET consists of six branching points. For each branching point, the branches and the rules for determining the branch probabilities are explained. The probabilistic tool which has been used is EVNTRE. The EVNTRE feature for defining and applying user defined functions has been used in the example below. The example reflects the issue in the form of text and comments. An EVNTRE specific syntax exists which translates the text into EVNTRE data input.

Branching point CI2:

Question: Is there an ignition source available in containment between CDS and core relocation?

There are 3 branches:

IGSe2 Ignition source (early) exists between CDS and core relocation

IGSl2 Ignition source (late) exists between CDS and core relocation

IGSn2 No ignition source exists between CDS and core relocation

Firstly, according to PAR manufacturer and to latest experiments, ignition by PARs shall be assumed, if at a certain PAR location the hydrogen volume fraction reaches 10 % and the steam volume fraction is <= 50 %. This type of ignition is called "early". If an early ignition occurs, the combustion will not undergo flame acceleration phenomena, and detonations are not possible. However, it is not a design requirement for the PARs to act as igniters, and they have not been qualified accordingly. A residual probability for no ignition by PARs around 10 % is estimated. Secondly, according to German PSA L2 "Datenband" [37] potentially detonative containment atmosphere is ignited below 12 vol.% H_2 with probability 0.8 to 1.0, and above 12 vol.% H_2 with the complement (i.e between 0.2 and 0.0). These statements have been combined as follows:

1. No ignition source at all is incredible, and it would be optimistic to dismiss ignition because no ignition means no combustion and no hydrogen risk. Therefore, probability for no ignition source (branch IGSn3) is set to zero.

2. Ignition with hydrogen vol. fraction <= 10 % (early ignition, branch IGSe2) occurs with probability 0.9 to 1.0 (uniform distribution).

3. Ignition with hydrogen vol. fraction above 10 % (late ignition, branch IGSl2) occurs with complementary probability

Branching point H2GME2:

Question: What is the MELCOR hydrogen generation in in-vessel phase?

The “branching point” has just a single branch which is used for definition and input of parameters to be used in downstream branching points.

In this branching point, a user defined parameter H2GME2 is defined and quantified for different cases. This parameter is the MELCOR hydrogen mass generated in the in-vessel phase. The parameter will be applied in subsequent branching points. In the present branching point only the direct MELCOR hydrogen masses are considered. The uncertainty of the MELCOR results is taken into account in following branching points.

Case 1: If a transient occurs as initiating event: A MELCOR run exists for a typical sequence within this case. This run calculated 800 kg of hydrogen.

Therefore: H2GME2 = 800.0 kg

<u>Case 2:</u> If case 1 is not true (i.e. if the initiating event is a LOCA):

Several MELCOR runs exist with Hydrogen generation between 600 kg and 700 kg.

Therefore: H2GME2 = 600.0 – 700.0 kg (uniform distribution)

(Remark: cases with reflooding of a partly damaged core are investigated in a separate part of the APET)

Branching point H22:

Question: Provide Hydrogen in-vessel data

The “branching point” has just a single branch which is used for definition and input of parameters to be used in downstream branching points.

These parameters are defined:

H2GSS2	Minimum hydrogen mass generated in in-vessel phase (kg) according to state of-the art
H2GSH2	Maximum hydrogen mass generated in in-vessel phase (kg) according to state of the art
RANDH2	Random number for hydrogen volume fraction assessment
H2VFA	Lower hydrogen volume fraction limit for flame acceleration and DDT (vol. %)

The parameters will be applied in a subsequent branching point for evaluating the distribution of hydrogen volume fractions in comparison to the uncertain limits for flame acceleration and DDT.

A NEA state-of-the-art report /NEA 01/ shows that the degree of final cladding oxidation can be in the range of 25 % - 90 % for a typical PWR. Characteristic values are between 30 % - 50 % depending on the sequence. The final cladding oxidation is lower for a fast than for a slow sequence. Water flooding increases the final cladding oxidation.

The reactor under consideration has slow core degradation because there is much water available during core degradation. Therefore, it can be expected that the hydrogen generation be in the upper part of the range. Consequently, the upper limit of generation is assumed to be 90 % of the theoretical maximum. The lower limit for hydrogen generation is set to 30 % of the theoretical maximum (see characteristic NEA data above).

The total Zr mass in core could theoretically generate 1240 kg of hydrogen. Therefore, the upper limit (90 %) is 1116 kg, and the lower limit (30 %) is 372 kg.

The distribution of the hydrogen volume fraction is assumed to be triangular with the MELCOR volume fraction (see next branching point) as peak value. Parameter RANDH2 is needed to define the distribution shape.

The minimal hydrogen volume fraction necessary for flame acceleration plus deflagration-to-detonation-transition (DDT) to be possible is depending on several parameters like steam content or geometry which are not available in this context. In addition, the corresponding dependencies are complicated. Therefore, an expectation value of 15 vol.% and a range between 10 vol.% and 20 vol.% was estimated from Fig.1.1-1 in the NEA SOA report (curve: detonation limit lambda=0.5m /NEA 01/).

H2GSS2 = 1116.0 kg

H2GSH2 = 372.0 kg

RANDH2 = 0.0 – 2.0 (symmetric triangular distribution)

H2VFA = 10.0 – 20.0 (uniform distribution)

Branching point H2VMX2:

Question: Maximum MELCOR hydrogen volume fraction w/o ignition

The "branching point" has just a single branch which is used for definition and input of parameters to be used in downstream branching points.

In this branching point, a user defined parameter H2VMX2 is defined and quantified for different cases. H2VMX2 is the maximum hydrogen volume fraction [%] which could occur in the containment if no ignition occurs.

The standard assumption in the MELCOR runs is that ignition due to PARs occurs when a hydrogen volume fraction of 10 % is reached and oxygen and steam concentrations (<50 vol.%) are in the ranges allowing combustion. In the event tree, however, the ignition could also be delayed and consequently a maximum hydrogen volume fraction > 10 % could be reached (cf. branching point CI2). MELCOR analyses indicated two different sets of sequences with different hydrogen volume fractions.

Case 1: Case 1 consists of sequences where MELCOR runs showed a deflagration in the containment, i.e. 10 % hydrogen volume fraction has been reached. To estimate the maximum possible concentration without ignition a variation of a MELCOR run for a certain sequence was made with disabled ignition and a maximum concentration of about 16 vol.% was reached. From these observations, it is concluded that rather high maximum concentrations are possible if there is no (early) ignition. Due to general uncertainties, a maximum concentration of 17 vol.% (instead of the 16 vol.% observed in a MELCOR run) was chosen for upper limit of the distribution. The lower limit was put at 9 % and a homogeneous distribution was assumed.

Therefore: H2VMX2 = 9.0 – 17.0 (uniform distribution)

Case 2: All remaining cases (i. e. cases where MELCOR did not show ignition)

In all other MELCOR runs no deflagration was calculated before RPV failure, i.e. the maximum hydrogen concentration was always below 10 %. Because no detailed analysis was made, an average value of 9 vol.% is assumed for the maximum hydrogen concentration in these cases. Due to general uncertainties, a possible range of +/- 3 % is assumed with a uniform distribution.

Therefore: H2VMX2 = 6.0 – 12.0 (uniform distribution)

Branching point H2FA2:

Question: Does Hydrogen potentially reach limit for flame acceleration + DDT

This branching point asks for the possibility to reach critical hydrogen volume fractions, assuming no early ignition. Whether an ignition actually occurs in the critical regime depends on assumptions about ignition, to be evaluated in the following branching point.

There are 2 branches:

H2FAy2 Hydrogen flame acceleration + DDT limit is potentially exceeded

H2FAn2 Hydrogen flame acceleration + DDT limit cannot be exceeded.

The following issues are taken into account in quantifying H2FA2

1. The total hydrogen mass generated by MELCOR in the in-vessel phase (see question H2GME2)
2. Hydrogen mass generated in the in-vessel phase according to SOA, providing an uncertainty range around the MELCOR hydrogen mass (see question H22)
3. Maximum hydrogen volume fraction, which would be reached in any position inside the containment if no combustion occurred, derived from MELCOR results (see question H2VMX2).
4. An uncertainty band for the maximum hydrogen volume fraction based on uncertainty of in-vessel hydrogen generation (see above item 2) and nominal MELCOR hydrogen volume fractions (see item 3 above). (Calculated by user function using information provided above)
5. The probability that conditions for flame acceleration + DDT exist (Calculated by user function using information provided above).

For the plant under consideration the probability for branch H2FAy2 has been determined as 5.497E-02 while H2FAn2 is the complement (9.451E-01). H2Fay2 only indicates that the conditions for flame acceleration and DDT are reached if no early ignition occurs. It does not yet provide final probability for flame acceleration and DDT. Therefore the next branching point is needed for final assessment.

Branching point COFA2:

Question: Does flame acceleration + DDT combustion occur in containment atmosphere?

There are 2 branches:

COFAy2 Flame acceleration + DDT combustion occurs between CDS and relocation

COFAn2 No flame acceleration + DDT combustion occurs between CDS and relocation

Case 1: Flame acceleration + DDT combustion occurs with probability 1.0 if maximum hydrogen volume fraction in containment could reach flame acceleration limits (see question H2FA2 above) and if ignition sources do not ignite below 10 vol.% hydrogen (see question CI2).

In the reference plant the calculated fraction of sequences belonging to this case is 2.700E-03

Case 2: All remaining sequences cannot not reach flame acceleration conditions or they ignite below 10 vol.% hydrogen. No flame acceleration occurs. In the reference plant the calculated fraction of sequences belonging to this case is 9.973E-01.

According to the introduction of this section, the question to be answered by this part of the APET is the probability that flame acceleration and deflagration-detonation-transition (DDT) will occur. The answer as given by this APET example is 2.700E-03.

2.4.4 Dynamic reliability methods (DRM)

2.4.4.1 Introduction

An event tree can easily identify different orders of phenomena – e.g. a branching point in an event tree could have the two branches “A occurs before B” or “B occurs before A”. But even with a particular effort to take into account all the possible dependencies between the events, a global L2PSA exhibits certain limitations, particularly with regard to explicitly time-related aspects of the accidents. Again, classical L2PSA tools like EVNTRE and KANT can in principle deal with this issue by applying time dependent user-defined functions calculating branching probabilities. However, less suitable tools show a related methodological handicap which has long been identified. Various international studies have been carried out to overcome it. Different methods, called Dynamic Reliability Methods (DRM), have been developed to better take into account time-related aspects of the accidents. This subject has been studied within the SARNET European excellence network on severe accidents [14],[15].

The DRM can be classified in two categories:

- DRM using classical PSA tools,
- DRM using new specific tools.

The DRM using classical PSA tools have to deal with the limitation of the existing tools. A solution is to develop a tree of event-trees (called macro-event tree) with a classical tool, as shown on the following example: we consider two time related and dependent events, A and B. A can occur before B, or B can occur before A. The following figure shows how we can deal with this problem with a classical approach and the macro-event method:

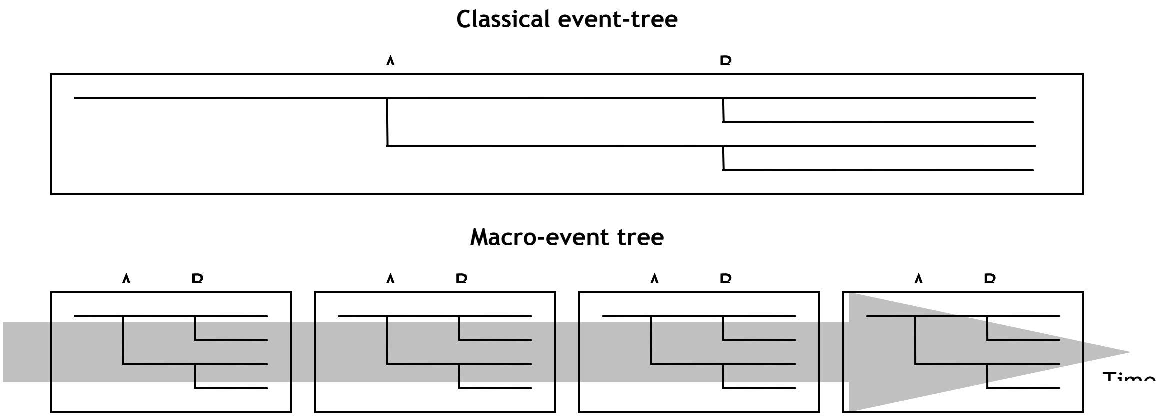


Fig. 8 DRM with duplication of macro-events at each time step (from IRSN [16])

The event tree deals with time and is divided into time intervals (each macro-event deals with one time interval, the number depends on the required precision and the computation limits). This solution shows that classical tools (KANT was taken as example) can be used if they are flexible enough (possibility of implementation of physical modelling in the event-tree nodes). The use of a macro-event allows taking into account time-dependent stochastic events. Such a solution is quite simple, especially when events are modelled by user functions and do not need complex methodology. Severe accidents codes can also be considered as users-function to quantify the events. The example with KANT was extended to a coupling with an ASTEC however in that particular case a specific algorithm (management of ASTEC calculations restarts) was implemented to reduce CPU time induced by ASTEC as much as possible.

The second category is based on the development of new tools dedicated to the dynamic reliability problems. Two of these dedicated new methods are the Monte-Carlo Dynamic Event-Trees (MCDET), and the Stimulus-

Driven Theory of Probabilistic Dynamics (SDTPD). The asset of these methods is that they are faster than methods using classical tools.

The probabilistic dynamics method MCDET is a combination of Monte Carlo simulation and the discrete dynamic event tree (DDET) method. It was implemented as a module which can operate in tandem with any deterministic code simulating the system and process dynamics. MCDET was supplemented by a Crew-Module which permits simulation of the dynamics of human actions depending on but also acting on the system and process dynamics as modelled in the deterministic code and on stochastic influences as modelled in MCDET [17].

These different methodologies have been the object of a benchmark exercise relative to hydrogen combustion risk assessment in case of water injection during in vessel core degradation for a French 900 MWe PWR, in the framework of SARNET [15].

L2PSA always have to deal with time related events and dependencies between the events. Nevertheless, DRM are not efficient enough (in terms of computation times) to deal with complete L2PSA. That is why, for the moment, DRM are only applied for simplified studies on specific subjects.

2.4.4.2 Example of DRM application by IRSN

Dynamic reliability methods have been applied by IRSN on a specific subject: the quantification of hydrogen combustion risks in relation with severe accident measures, giving some interesting outcomes for the Severe Accident Management Guide review [16]. In fact, dynamic reliability methods can be complementary to global L2PSA for some specific applications, even if for the moment it cannot be used for a complete L2PSA.

The method used by IRSN consists of building a macro-event tree with KANT that models the occurrence of each event (here manual in-vessel water injection and its consequences on hydrogen production, manual spray system activation and its consequence on the containment gas composition, evolution of containment gas composition during the time-step, gas mixture flammability, ignition of a combustion partial or not, evolution of atmosphere composition due to combustion, containment failure due to combustion). The repetition of these macro-events associated to probabilistic function related to the stochastic events (manual actions, ignition) in a global event tree provide a simple solution to model all combinations of events whatever their chronological order.

The results obtained by such a study must of course be considered very carefully due to the uncertainties on the different phenomenological issues. The quality of the simplified models used cannot be compared to those included in the detailed deterministic severe accident codes, and all the details of the real physics are not represented. Nevertheless, it can provide useful information and help in the understanding of where the risks are. Examples results are presented on the following figure:

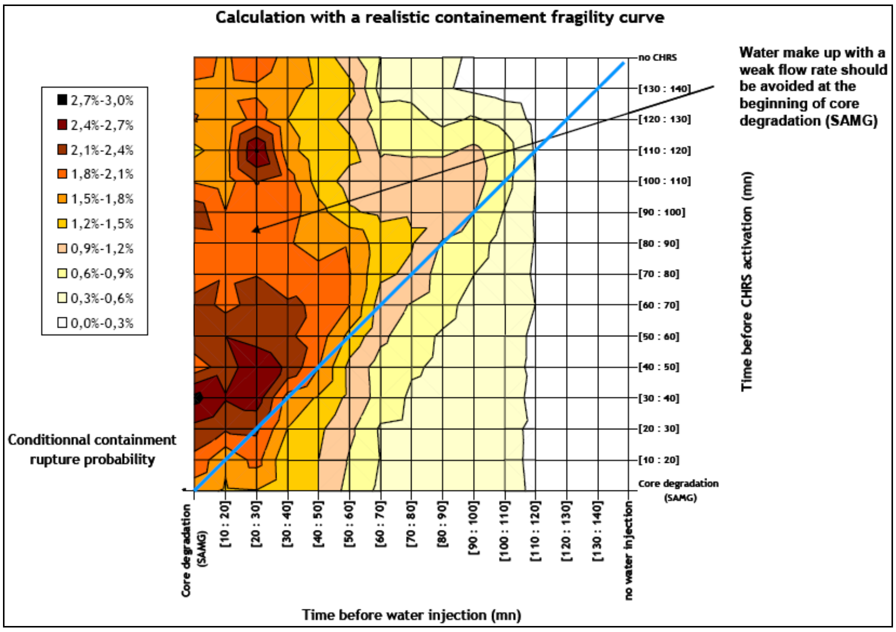


Fig. 9 Example of DRM application.

Conditional containment rupture by hydrogen combustion probability in function of safety system activation

(from IRSN L2PSA on a French PWR)

The study shows a positive role of the spray system whatever its time of activation (the depressurisation effect seems to be more beneficial than the increase of flammability; the spray system activation induces more hydrogen combustions but with low total pressure peaks that do not jeopardise the containment); this suggests that the interest in SAMG of an activation of the spray system should be examined as soon as possible without any limitation. The graph also shows that with no water injection and no spray system activation, there is no containment failure. This is effectively a result of the global L2PSA. In that case vessel rupture would be highly probable.

2.5 RELEASE CATEGORY DEFINITION

2.5.1 General approach

Most L2PSAs finally aim at providing information about the release of radionuclides into the environment. In principle each individual sequence of an APET would lead to a specific type of release. However, even if it were possible to perform this detailed analysis, it is better to group sequences together which have similar releases. Sequence groups which have similar releases are called release categories. This grouping process is addressed in the present section. In a subsequent step a specific source term of radionuclides is associated to each release category. This step is addressed in chapter 7 of the guideline. The link between release categories definition and the final presentation of L2PSA results is presented in the Volume 1 of the ASAMPSA2 guideline.

There are a number of factors that impact the way of defining the release categories, mainly related to the objectives and scope of the PSA.

The above impacts the details in timing, the details in tracing failed and working protection and consequence limiting functions in level 1 as well as level 2 parts of the PSA model, and details of containment failure modes.

A Release Category defines a grouping of APET end points, within which it is expected that the source term into the environment is similar because all of the following apply:

- Identical, or similar, initiating events and plant failures have occurred,
- Identical, or similar, severe accident phenomena have occurred,
- Identical, or similar, engineered features are available to mitigate the release of radioactivity to the off-site environment,
- Identical, or similar, containment response is expected.

This grouping brings together accident sequences with similar fission product release and transport mechanisms. Historically, this was partly done to reduce the number of deterministic source term calculations to be performed within the L2PSA. However, with modern computing capability, some approaches do not have this restriction.

Depending on the objectives of the individual L2PSA, additional attributes needed for subsequent calculation of offsite consequences may also be defined, such as release height and thermal energy of the release.

A generic table of attributes, which may be used to specify the set of Release Categories, is given in Table 2.

Table 2 Attributes that may be used to specify the set of Release Categories [9]

Release attributes	Variations	
Timeframe of the severe accident in which the release begins	At the onset of core damage (e.g. bypass of the containment) Early (during in-vessel core damage) Intermediate (immediately following breach of the reactor pressure vessel) Late (at least several hours after breach of the reactor pressure vessel)	Typical attributes of L2PSA
Pressure of reactor pressure vessel during core damage	High (near nominal) Low (depressurised)	
Modes or mechanisms of containment leakage	Design basis accident leakage Beyond design basis accident leakage Catastrophic rupture of containment Loss of coolant accident in interfacing system Steam generator tube rupture Open containment isolation valves Basemat penetration	
Active engineered features providing capture mechanisms for radioactive material	Sprays Fan coolers Filtered vents Others	
Passive engineered features providing capture mechanisms for radioactive material	Secondary containments Reactor buildings Suppression pools Overlying water pools Ice beds Tortuous release pathways Submerged release pathway	
Time relative to the start of the severe accident	Short (e.g. for pressurised water reactor typically less than 2 hours) Medium (e.g. for pressurised water reactor typically between 2 and 10 hours) Long (e.g. for pressurised water reactor typically greater than 10 hours)	Additional attributes for linking to Level 3 PSA
Location of release	Ground level Elevated (stack or roof)	
Energy of release	Low (minimal buoyancy in ex-plant atmosphere) Energetic (highly buoyant)	
Release rate	Rapid 'puff' release Slow continuous release Multiple plumes	

It can be seen that there are different possibilities for the categorisation of the release taking into account release attributes such as timing of the release or containment failure mode. When selecting the Release Category attributes, the emphasis should be in presenting the PSA results to support the high level objectives. Depending on these high level PSA objectives, some of these attributes can be combined.

For some applications, the release magnitude attribute alone may be sufficient, e.g. if the licensing limits set for the release are determined by a simple risk metric such as large release frequency (LRF). However, the time dependent behaviour of the releases and system dependency will then be lost.

From the containment system behaviour point of view, it may be beneficial to categorise the release categories based on containment failure modes, as the PSA results directly give the information on the failing systems. From these results it is easier to find out the weaknesses of the severe accident management approach and to identify possible enhancements on the plant that should be made to achieve better containment performance.

Categorisation based on timing of release alone does not give adequate information without the magnitude of the release. For supporting plant external emergency preparedness planning it gives valuable information on what kind of actions are to be concentrated on. If only timing is to be used as a risk metric, a threshold release should be selected, above which the release is considered significant. Sometimes this licensing limit is applied, i.e. a large early release frequency (LERF), which involves determination of the combination of both large and early release.

A single categorisation scheme might not be adequate if all the different potential outcomes of the L2PSA study are to be considered. In such a case it may be beneficial to apply several different categorisation schemes (filters) when processing the results of the L2PSA.

If there are plans to carry out a level 3 PSA for the plant, extensive combination of the individual sequences is not recommended, unless the release characteristics (e.g. time behaviour and the elevation / location / energy) of the releases in the combined sequences are adequately similar. Adequacy here depends on the accuracy of the model used in the subsequent evaluation of the environmental consequences in the Level 3 PSA. In general, the release information in individual sequences should be maintained as far as possible, as combinations could lead to loss of dependencies within the sequences and the subsequent environmental consequence analysis could give misleading results.

A simple example for a group of release categories is provided below.

Table 3 Simple release category example

RC-0	Design leak
RC-1	Filtered vent
RC-2	Slow pressurisation, containment failure (CF)
RC-3	Base-mat penetration
RC-4	Late ex-vessel phenomena, CF
RC-5	CF at RPV failure, low RPV press.
RC-6	CF at RPV failure, high RPV press.
etc	Containment leak, bypass, V-LOCA, recovery, containment spray

If a release analysis is performed for each individual sequence of the APET, providing a specific source term for each sequence, the definition of release categories as described above is not really needed although it may be helpful in interpreting PSA results.

Source term based release categories can be defined according to the released amount of radionuclides as follows:

- Sequence grouping based on the released amount of every element or chemical form included in the definition of the RC
    - The group intervals determine the resolution of the RC.
- Additional attributes, like time of release, may be included
    - RC span a N+M-dimensional space with N different elements and chemical forms and M additional attributes.
- The number of elements and attributes should be low and concentrate on the radiologically most relevant ones,
- A common measure (e.g. total release of activity in Bq, or the integrated health hazard based on radiotoxicity of the released isotopes) could be applied for grouping.

2.5.2 Examples for categorisation

Many plant design features and accident phenomena have been shown to influence the magnitude and characteristics of source term for severe accidents. Some of them are fixed and others can vary from one accident sequence to another, changing the associated release category. Examples of these changeable factors are given in table 2 above.

In the following, examples for the categorisation of the releases are discussed based on the containment failure mode, and combined with the magnitude and timing of the release. For a particular plant the categorisation could be condensed (e.g. hydrogen is a minor issue in an inerted containment) or may have to be amended (e.g. melt through of penetrations at bottom of containment).

2.5.2.1 Example 1: Conventional - approximately 10 Release Categories (GRS)

In this scheme adopted in the PSA for a PWR, the Release Categories are based on the containment failure mode alone, which defines the available pathways for offsite release. It is restricted to full power operation. Further, different failure modes with comparable releases are put into the same release category. (E.g. the different sequences “high pressure RPV failure” and “bypass through uncovered SG tube leak” are grouped together because both have a very high and early release.) This is resulting in a relatively small scheme which can easily be documented.

Table 4 Frequency of Release Categories excerpt from [34]

RELEASE CATEGORIES*		
Name	Main reason for release 1)	frequency (10-7/a)
FKA	High pressure RPV failure or Bypass through uncovered SG tube leak	2.1
FKB	Failure to isolate containment ventilation	0.13
FKC	Bypass through covered SG tube leak	0.23
FKE	Failure of sump suction tubes or failure of containment venting	1.4
FKF	Failure of venting filter due to hydrogen combustion or leak in annulus due to hydrogen combustion	2.1
FKH	Ventilation duct failure at venting system due to hydrogen combustion (filter intact)	2.6
FKI	Containment venting as designed	8.8
FKJ	Containment function within design range	7.7

*all Release Categories except FKJ have additional ground contamination

2.5.2.2 Example 2: Conventional - approximately 20 Release Categories (STUK)

In the following example, from a PSA for a BWR, the categorisation of the releases is carried out based on the containment failure mode combined with additional attributes for the magnitude and timing of the environmental release. The objectives to be reached by this PSA are identification of containment failure modes and their frequencies and estimate of associated releases for emergency preparedness. The containment failure mode classification is comparable to the previous one, but without grouping different modes into the same release category, and with additional features for shut down states. The release magnitude is broadly grouped into.

- Reactivity induced accident (RIA) and vessel breach (VB),
- Open containment,
- Containment bypass,
- Large isolation failure,
- Failure to isolate the fuel transport channel penetrating the containment,
- Early containment failure,
- Late containment failure,
- Filtered venting,
- Small isolation failure,
- Leak,
- Intact containment,

- Fuel damage in the fuel pool.

Reactivity induced accident (RIA) and vessel breach (VB) are considered very energetic events for the type of plant under consideration, and it is very probable that containment failure occurs almost simultaneously with the initiating event. The releases start very rapidly, as the fuel is either fragmented or without coolant, and release magnitude will be very large. Vessel breach may even lead to recriticality of the core after its disintegration.

Open containment is usually considered during BWR refuelling outages. Many BWR designs do not have real containment above the operation deck, and thus the possible core damage occurring during refuelling outage would lead to more or less direct release path into the environment. As the reactor building itself is not designed to withstand substantial overpressure, it would have only minor retention capability. The releases would occur from the beginning of the core damage and the amount of released fission products would be very large.

Containment bypass involves a leak path directly from the RCS outside the containment. Typical bypass routes considered are failure to isolate the reactor coolant loop in case of a leak outside the containment. As retention in the containment is not effective, the releases are very large, depending on the leak diameter of the bypassing system. However, RCS depressurisation directs part of the leakage into the containment, and if the bypass route is very small, the sequence should be classified to some other category, e.g. small containment isolation failure.

Large isolation failure could typically take place when carrying out maintenance work. The large penetrations through the containment wall are not necessarily closed during maintenance, the material hatch is being used and the personnel locks may have special arrangement to allow easy access of the containment. Another possibility is failure to isolate the ventilation system. Especially during maintenance, a large capacity ventilation system is used, but the system used during power operation is large enough to cause large isolation failure, as well. The releases start from the beginning of the core damage, and the release fraction is very large. Distinguished from the early containment failures, the containment would not pressurise, and thus a similar burst would not take place, but the release is rather continuous.

Failure to isolate the fuel transport channel penetrating the containment during refuelling outages can also be considered as large isolation failure. Although the refuelling pool inside the containment and the fuel pool in the fuel building outside the containment are filled with water, the overpressure building up in the containment would eventually push the water out of the containment leading to air outflow of the containment. Outside the containment the leak path would pass through the fuel pool water, and thus some of the fission products would be trapped in the pool. The isolation failure in this case would result in somewhat different behaviour of the release than that in atmospheric release path from the beginning of the accident. Therefore the fuel transport channel isolation failure could be classified as late containment failure introduced later. The classification, however, may depend on the plant design and emergency operating procedures, and therefore any single category is not introduced here.

Early containment failure may be caused by an energetic event such as hydrogen burn or steam explosion. These may occur already during the in-vessel phase, although the in-vessel steam explosion is often ruled out as a very unlikely mechanism. Depending on the containment design, a global hydrogen burn could threaten the containment integrity, but hydrogen detonations are of concern to any containment design. In case of high-pressure melt ejection the reactor cavity integrity is challenged and melt would be spread around the containment causing direct containment heating (DCH) and rapid pressurisation. Early containment failure would result in large leakages, and thus the release fraction would be very large. The release is also characterised by the burst of the fission products present in the containment atmosphere at the time of failure.

Late containment failure may be due to slow overpressurisation or basemat melt-through. As the failure occurs rather late after the beginning of the core damage, the majority of the fission products released into the containment atmosphere are deposited. Slow overpressurisation would lead to containment failure at pressures substantially higher than the design pressure. Therefore the release fraction is much lower than in the case of the early failure, although the release is also characterised by the burst of the containment atmosphere from high pressure. The leak path might not be as large as caused by an energetic event, as the pressurisation takes place gradually. The basemat melt-through, however, would eventually lead to large failure, but the airborne releases would probably be lower than that due to other kind of large failures. Furthermore, in case of the steel-shell containment, even the slow overpressurisation could lead to catastrophic failure. The nature of the releases, e.g. from the released fission products point of view, somewhat differ between these two failure mechanisms, and therefore they can also be distinguished when selecting the Release Categories.

In general, large failures results in such high flow rates that the leakage collection systems outside the containment cannot cope with them. Therefore it is not realistic to assume that very large leakages could be directed to stack via filtration systems, but they rather should be treated as ground releases.

Filtered venting is applied in some designs to protect the containment from overpressurisation. The need for its usage would be realised rather late, although somewhat earlier than the failure due to slow overpressurisation, as its set point is usually the design pressure of the containment. Due to the late usage of the system, the fission products have been deposited from the containment atmosphere, and furthermore, the release is reduced by the filtering system and directed to the stack. If carbon filters are used, the possibility of the filter heat-up due to accumulating fission products may lead to burn of the filters, which significantly increases the release.

Small isolation failure results from unsuccessful isolation of system having small diameter penetrations. The leakage through the system affects containment pressurisation, but still overpressure will build up. Containment overpressurisation, however, may be impossible. The release is continuous, but of much lower rate than in case of large isolation failure. Deposition in the containment decreases the environmental releases. Especially if the spray system is available, the release fraction can be decreased significantly, as the spray washes fission products from the containment atmosphere and reduces containment pressure. In case containment overpressurisation is possible, the containment filtered venting or late containment failure will probably not affect the overall release fraction, unless the initial leak is directed to stack via filtering system.

Leak is considered as a leakage through a closed but untight isolation valve, through leaking penetration or due to damaged seal of large penetrations at a rate exceeding the containment design. Some very small isolation failures may be classified as a leak, as well. The leakage is low enough not to significantly affect the containment pressurisation, and the releases may be collected to be directed to the stack. The releases are relatively small, and possible filtration reduces them drastically. Also the availability of the spray system reduces the releases.

Intact containment is considered if the containment leakages remain below the specified design values. The releases remain very small, although spray system was not used or leakages were not collected to be released through stack via filtration. Of course, these engineered systems decrease the release fraction significantly.

Fuel damage in the fuel pool may, depending on the design, take place in a separate building or in the containment. The cases with the fuel damage inside the containment may be classified in the categories presented above, but those occurring in a separate fuel building are different, as there is no containment at all. The category of open containment is not necessarily applicable either, as the fuel may have cooled down for several years. The amount of fuel may be much larger than that present in the reactor core and long-lived radioisotopes, such as 137Cs, are still present within the fuel. Therefore this kind of fuel damage has the potential for very large releases.

When considering the containment behaviour above, it is assumed that the containment design is of low-leakage type; typically the design value is well below 1% per day. If the containment or confinement design leakage is higher than this, the Release Category classification should be modified accordingly when considering the cases of containment isolation failures, leaks and intact containment.

In case such a detail in release magnitude is not needed, small and very small categories, as well as very large and large categories, could be combined. The same could be applied also for very early and early release timing. The difference of early and late release timing in large release, however, should be distinguished.

An example of release categorisation based on the classification described above is shown in Table 5. The different initiators between the sequences may affect the absolute timing of the sequence and thus the time behaviour between LBLOCA and SBLOCA may have a difference of several hours. The difference, however, from the emergency preparedness point of view may not be very different, as the knowledge of the severity of the accident and thus the level of the countermeasures may appear only after proceeding over a certain level of system failures. The need for specific systems may appear later during more slowly proceeding sequences, and thus the time between discovering the severity of the accident and the core damage may not be of such widespread as the absolute timing would suggest. Therefore the differences in absolute timing in classification do not cause major inadequacy of the selected categorisation.

Table 5 STUK Example of Release Categories

Release	Containment failure mode	Release timing [a]	Elevation [b]
Very large	RIA or VB	Very early	Ground
	Open containment (no containment)	Very early	Ground
	Containment bypass: - SGTR - V-LOCA	Very early	Ground
	Large isolation failure: - Equipment hatch - Personnel lock - Ventilation system	Very early	Ground
	Early containment failure: - Hydrogen burn - Steam explosion - High-pressure melt ejection	Early	Ground
Large	Slow overpressurisation	Late	Ground
	Basemat melt-through	Late	Ground
	Filtered venting + filter burn	Late	Stack
	Small isolation failure	Early	Ground
Moderate	Filtered venting	Late	Stack
	Small isolation failure + spray	Intermediate	Ground
	Leak	Intermediate	Stack
Small	Leak + spray	Intermediate	Stack
	Small isolation failure + leak collection	Intermediate	Stack
Very small	Small isolation failure + spray + leak collection	Intermediate	Stack
	Leak + leak collection	Intermediate	Stack
	Intact containment	Intermediate	Stack

[a] Intermediate means a situation where a relatively small leakage is continuous and the release builds up during a long period. Other kinds of timings consider the majority of the release taking place in a shorted period.

[b] Energy content of the release is not considered here. Releases taking place at lower elevations may be elevated due to high temperature gases released at the same time as fission products.

2.5.2.3 Example 3: Large Number of Release Categories (IRSN)

In this scheme there is no condensation of APET endpoints. This approach is only practicable due to recent advances enabling rapid source term calculations for large numbers of accident sequences (see section 7.7)

Objectives of this PSA are to assess containment safety features, identify containment failure modes and their frequencies, and estimate the associated environmental releases for emergency preparedness. The classification of containment failure modes is similar to example 2 but additional attributes preserve information on the accident phenomenology and performance of containment safety features. In this case the release magnitude and characteristics are not grouped at all but, as a simplified source term methodology is used, the uncertainty in the environmental release magnitude and characteristics for any single sequence may be higher than in the traditional integrated codes approach. In terms of presentation of PSA results, it is necessary to post-process the results into a smaller number of re-grouped Release Categories.

In the French PWR 900 MWe L2PSA, L2 sequences have been gathered into Release Categories according to 37 specific parameters calculated through the Accident Progression Event Tree (APET).
These parameters are defined for each phase of accident (in vessel phase, vessel rupture phase, ex-vessel phase) and provide information about:

- Initial reactor state,
- Containment bypass (through containment penetration, interfacing LOCA, initial or induced SGTR),
- Accident phase kinetics (beginning of core degradation, vessel rupture, basemat penetration)
- Time of core reflooding,
- Dynamic phenomenon (hydrogen combustion, in-vessel steam explosion, ex-vessel steam explosion, direct containment heating),
- Venting system opening,
- Time of containment failure,
- Leakage section,
- Aerosol behaviour,
- State of spray system, ventilation/filtration systems, RCS safety valves.

Due to the large number of modalities for each parameter, several thousands of Release Categories are generated by the APET quantification, but a release calculation can be performed for each Release Category (a fast-running source term code is included in the APET and is quantified by KANT).
In this example, each end point of the APET is essentially its own Release Category. Within such a scheme it is necessary to make the final presentation of results in terms of grouped results, in this case termed 'Regrouped Release Categories (R-RCs)'. The R-RC scheme is based on containment failure modes and amplitude of consequences.

2.5.2.4 Example 4 : PSA with Full Level 3 (Sizewell B NPP, UK)

Within the UK regulatory framework there is an expectation that Level 1, Level 2 and Level 3 PSA will be performed.

The objective of this PSA was primarily to assess individual risk to members of the public. Within the L2PSA, the objectives were to assess containment safety features, identify containment failure modes and their frequencies and estimate the associated environmental releases for the Level 3 PSA. A subsidiary objective was to assess an uncontrolled release frequency, similar to the LRF concept described above.

The classification of containment failure modes is similar to example 2 but additional attributes preserve information on the accident phenomenology. Information on the performance of key containment safety features is included in the PDS definition. In this case the sequence characteristics are initially grouped but not extensively so. In terms of presentation of PSA results, the results were post-processed into a smaller number of L3 release categories, which formed the Level 2/3 interface.

The categorisation scheme has two stages. Firstly, sequences represented by the APET end states are grouped on the basis of similar accident phenomenology. This grouping was carried out using a source term logic tree, which is a condensed version of the APET addressing only those issues important to the source term. For each PDS, up to 38 intermediate source term categories are identified. Secondly, these intermediate source term categories are allocated into a set of pre-defined Release Categories on the basis of their off site consequences alone. These offsite consequences were based on estimations of the health effects resulting from a certain source term, an assessment which normally is outside the scope of a L2PSA.

The advantages of this two stage scheme are:

- The number of intermediate source term categories (i.e. combinations of potential accident phenomenology) allows a wide range of phenomena that are potentially important to the fission product release and transport, to be addressed,
- The Release Categories are pre-defined and well spaced in terms of their off site consequences,
- The site dependent Level 3 PSA is partially de-coupled from the L2PSA. Therefore, site dependent issues are only addressed in the Level 3 PSA. Updating of the Level 1/2 PSA does not feed forward into a requirement for updating the offsite consequence assessment in the Level 3 PSA - only the frequency allocated to individual Release Categories is changed.

The eventual Release Category attributes in the two step approach described above are:

1. Effective whole body dose at 80 m or 3 km in the first year following the accident, without implementation of early off site countermeasures, but excluding ingestion dose that would be averted by regulatory intervention to control contaminated foodstuffs,
2. Release duration,
3. Ratio of dose contributors from volatile to involatile nuclide groups,
4. Availability of warning time for evacuation within the Detailed Emergency Planning Zone prior to the main phase of release.

A total of 22 Release categories are defined [35]:

Table 6 Source Term Categorisation for the Sizewell B PSA

Cn						Sn
Short Duration				Long Duration		
Adequate Warning Time		Insufficient Warning Time		Insufficient Warning Time		
Dominated by volatile radionuclides	Dominated by involatile radionuclides	Dominated by volatile radionuclides	Dominated by involatile radionuclides	Dominated by volatile radionuclides	Dominated by involatile radionuclides	
C6SW		C6S				
C5VSW	C5ISW	C5VS	C5IS			
C4VSW	C4ISW	C4VS	C4IS			
		C3VS	C3IS	C3VL	C3IL	
		C2VS	C2IS	C2VL	C2IL	
						S3
						S2
						S1
						S0

Cn bounding dose of 10^n mSv at 3 km

Sn bounding dose of 10^n mSv at 80 m

It should be noted that the Release Category scheme, which ranges from a lower limit of 1 mSv whole body effective dose at 80 m up to doses of 1000 Gy at 3 km from the site, is applicable to design basis faults, beyond design basis faults and severe accidents. In calculating the dose, the pathways considered are: cloud exposure, inhalation and ground gamma doses calculated for the first year.

The Release categories used for core damage accidents are S2 and above.

2.5.2.5 Example 5 : Generic Scheme based on segregating accident sequences on the basis of the anticipated off-site response

There are a number of possible approaches, based on factors governing the magnitude and timing of the release, which are used to define Release Categories. One example of a scheme based on segregating accident sequences on the basis of the anticipated offsite response is given here (Ref. EUR 16502EN) and is suitable for categorisation in PSAs that have identification of LERF as a high level objective:

Below Threshold Release:

Releases associated with design basis faults or a release that is not likely to cause acute health effects in the vicinity of the plant or any long term restrictions on the use of extensive areas of land or water. One set of threshold values (expressed as a mass fraction of the total core inventory released offsite) used in previous EC Framework Programme projects is [36]:

Species / group	Threshold release
Noble gases, Iodine, Caesium, Tellurium	0.001
Molybdenum (Ruthenium), Barium, Strontium	0.0001
Cerium, Plutonium	0.00001

These thresholds can be seen as reasonably bounding environmental release fractions for accident scenarios where engineered safety features and SAM measures are successfully implemented, for modern reactor designs. These releases are, typically, applicable to the early phase of the accident (1 to 2 days) when early countermeasures are being considered and implemented as long as at least one barrier remains intact between fuel and environment. These threshold releases should not be applied to very late phase events such as late containment overpressure failure, basement melt-through etc.

Small Releases:

Releases where there may be a need to intervene to protect members of the public. The first countermeasure that may be required is the restriction of foodstuffs, albeit for a short period of time. The accident at TMI2 falls into this category.

Moderate Releases:

Releases which will require short term sheltering and the issue of stable iodine tablets to members of the public close to the site and may require the evacuation of small numbers of people and the control of foodstuffs in the wider population.

Sequences resulting in a moderate release are typically:

- Those where, after vessel failure, there will be a release to the environment via containment leakage with containment safeguard systems temporally unavailable,
- Those where, after vessel failure, the containment safeguard systems are unavailable and the containment fails due to penetration of the basement following a prolonged period of MCCI,
- Those where, after vessel failure, venting will be required due to the pressure increase within the containment by MCCI.

Large Releases:

Releases which may lead to a small number of early deaths and significant areas of ground contamination.

Sequences resulting in a large release are typically:

- Those where the major release occurs several tens of hours after vessel failure, by late over-pressurisation of the containment,
- SGTR with the secondary side filled with water,
- Interfacing system LOCA with the further retention of fission products in the auxiliary building or in the annulus.

Early Major Releases:

Releases which may lead to early (deterministic) health effects in the local population.

Sequences resulting in a major release are typically:

- SGTR without secondary side filled with water,
- Major pipe break in the auxiliary building (V sequence),
- Failure to isolate containment ventilation.

2.6 APET QUANTIFICATION AND RESULTS PRESENTATION

2.6.1 Overview of quantification methodology

The modelling and quantification are very much dependent on the objectives of the L2PSA, the use of separated or integrated approach, and the software tools available for the project.

The use of separated approach requires that plant damage states and the APET/CET release categories are calculated separately. The PDS frequencies are then in principle multiplied with the conditional release category probabilities to get the total release category frequencies.

The use of an integrated approach allows a direct quantification of the release category frequencies from the initiating events. PDS quantifications are in this case also needed to get the information on frequencies of individual plant damage states and their dominating contributors. Quantification may also be performed for individual sequences.

The amount of different types of probabilistic analyses for sequences, PDS,RCs, function and system top events depends on what is required to support results presentation and checking of model correctness.

2.6.2 Minimal cut set based quantification

Producing minimal cutsets based on event trees, fault trees and basic events, an important presumption is that the events are independent. This is of course necessary to consider when the study is designed, in L1PSA as well as in L2PSA. Note that: the events modelled have to be independent does not mean that the analyst cannot model situations with dependencies (e.g. the use of "split fractions").The solutions to do this is dependent on the possibilities that are offered by the software that is used.

A code like RiskSpectrum PSA needs to have a special modelling and treatment of branches in APETs (e.g. using different exclusive basic events for the different branches). It is important to know how the code chosen for the analysis perform different types of quantifications. One example is the use of Min Cut Upper bound for calculation of the top event frequency (Min Cut Upperbound is one of the methods used in RiskSpectrum, but also rare event approximation is calculated and second and third order approximations are also possible). The analyst needs to be careful about Min Cut upper bound results, because they may be non-conservative. The reason is that the min cut upper bound calculation treat events in the cutsets as independent even if they not necessarily are.

Looking from a sequence perspective this is not a problem. And since the sequences end up (in most cases) in different release categories this is also not a problem for calculation of release category frequencies either. A problem can occur when grouping of several release categories is performed (so that several branches appear in the same cutset list and these are treated as independent events. Of course, using only the first order approximation in the quantitative evaluation makes sure that the L2PSA part is correctly quantified. On the other hand, the L1PSA part may have difficulties with the first order approximation, especially if there are some independent events with large probabilities. The final impact on the result will be minor in most realistic cases, but has to be checked. Another problem with using a L1PSA code is when trying to perform an uncertainty analysis. The code does not necessarily know which basic events are dependent, and a parameter uncertainty analysis of all parameters in the L2PSA part may not be possible. In this case, the uncertainty has to be analysed with sensitivity cases. Note that a recent RiskSpectrum version has the possibility to use uncertainty distributions created outside of RiskSpectrum and these distributions can be designed to take into account any dependency between events in the model (see further the appendix in volume I describing RiskSpectrum).

In L1PSA, minimal cut sets, obtained by boolean minimisation, provide precise information regarding component failure modes and human errors leading to system failures and onset of core degradation. But unlike L1PSA, which is just interested in frequency of core degradation sequences, L2PSA has to address two dimensions, frequency and release associated to level 2 sequences (or to release categories).

Use of fault trees and minimal cut-sets in L2PSA

Sometimes L2PSA models are elaborated using level 1 software, which is based on event trees, fault trees and minimal cut sets. Minimal cut set theory contains some prerequisites that must be fulfilled by the system under analysis. In many cases it is possible to reconstruct the problem in such way that the results are credible. It is also possible to do different checks to identify if there are any cut set theory implications on the results, e.g. with regard to "success" branches, events with large probabilities, use of min cut upper bound in the results, impact on parameter uncertainty analysis results.

Differences between L1 and L2PSA event tree models

L1PSA can be regarded as an *additive* process, which takes each initiating event and adds its contribution to the frequency of relevant Plant Damage State (PDS). In contrast to level 1, L2PSA is a process, where the frequency from PDSs is divided into release categories. Hence, L2PSA preserves the total frequency that comes from level 1. To achieve this, L2PSA uses conditional probabilities (=mutually exclusive probabilities), which are rarely used in L1PSAs. An example of mutually exclusive probabilities in L1PSA is the use of summer time / winter time fractions. Usually in a L1PSA model, failure probabilities are small, and then the success probability is close to 1.0 and then success path probabilities do not need to be quantified exactly. It is enough to logically take into account success events and remove cut sets with mutually exclusive events. This is the standard procedure in L1PSA quantifications.

L1PSA consists of determination of frequencies of undesired consequence (Core Damage, CD). Most often this is done by modelling the undesired event with event trees and fault trees, from which minimal cut sets leading to CD are generated. This combination of fault trees and minimal cut sets set forward some assumptions, most important of which are the following:

1. Basic events are independent of each other, i.e. the probability of a basic event does not depend on the probability of any other basic event in any way,
2. The order of occurrence of events is not relevant.

These assumptions are fulfilled to a reasonable extent in most L1PSAs. Their validity in L2PSA is discussed below.

Independent events vs. mutually exclusive conditional probabilities

The independence assumption states that for two basic events A and B, the probability that they appear together is P(AB) = P(A)*P(B). However, if A and B are conditional probabilities of two alternatives, then P(B)=1-P(A) and *P(AB)=0*. Calculation with Mincut Upper Bound estimate assumes that P(AB) = P(A)*P(B), which is incorrect for conditional probabilities. Note that different modelling techniques can be used to make sure that impossible cut sets are removed before quantification. The ways this can be done is dependent on the specific features of the code being used. In any case, probabilities in L1PSA are in most cases so small that such error will be negligible. As regards L2PSA, the quantification error for MCS with mutually exclusive events may be somewhat more significant, since L2PSA often contains quite large conditional probabilities. This impact on the result can be checked by comparing the Mincut upper bound result with the result when applying the rare event approximation, which is a conservative estimate.

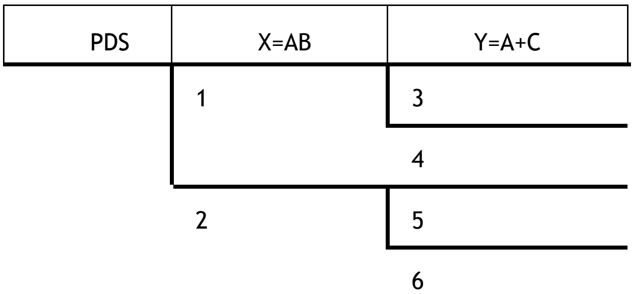
Conditional probabilities: sum for each branch point ≡ 1.0

The sum of probabilities for each branch point must be exactly 1. If this is not the case, level 2 model outputs more or less another frequency than defined by PDSs.

In some L2PSAs fault trees are used to quantify branch probabilities of Accident Progression Event Trees (APET). This is not problematic if basic events of fault trees used to calculate branch probabilities are independent from basic events in other questions and basic events in PDS cutsets. In this independent case, the probability of each branch is independent from all other probabilities, and *conditional* probability of a branch equals the probability of the fault tree cut sets. It is necessary to model any dependencies between questions that are important for the results.

Problems arise if a fault tree of a question contains basic events that are used in other questions or in cutsets of PDS. In this case complex calculations may be required to show that the sum of conditional branch point probabilities equals one.

Consider the following APET:



Assume that branches 2 and 6 are modelled with fault trees. To preserve the sum of conditional probabilities as 1, we represent the probability of branch 5 with one basic event. The question is, what must the probability of branch 5 to be to fulfil requirement that sum of conditional probabilities equals 1?

The probability of branch 5 will be P(Not Y|2). This is A*B*NotA*NotC which ends up in zero. This will be correctly treated in a code like RiskSpectrum if logical success treatment is turned on.

The probability of branch 6 will in this case be P(Y|2. This is P((A+C)|AB) = 1.

Replacing “A+C” with one event will hide the dependency and is of course not a correct treatment.

Assume that probabilities for A,B and C are 0.01. Incorrect evaluation for P(5) treating question Y as independent gives P(5)=1-P(0.01+0.01) = 0.98. Correct evaluation gives P(5|2) = 0. Not a small difference.

If function Y is complex, sub-functions can be defined on the highest level, which can be negated with a NOR- or NAND-gate, thus keeping the model workable.

Introduction of NOT logic in a model usually increases computation tims. It is therefore adviceable to use not-logic only in those cases where success branch impact is expected to be significant.

In principle, it would be possible to solve the APET using full probabilistic success calculations, and the sum of conditional probabilities would automatically be 1 for each branch point. However, this type of calculation is so complex that it can be used only for very small models, since the calculation must include all PDS sequences to treat correctly basic events that are common for L1 and L2PSA.

Using L1PSA tool, it is possible to construct numerically correct L2PSA model for point values. For a set of fixed numbers, it can be demonstrated that the sum of conditional probabilities equals 1 for all branch points, but this is more difficult in the case of uncertainty analysis, since L1PSA tools usually treat probabilities of basic events as independent of each other (see information in volume 1 appendix about use of externally defined uncertainty distributions in RiskSpectrum that takes care of dependencies during uncertainty analysis).

Example: Consider APET branch point, which contains three branches A, B and C. In level 1 tool, probabilities of A, B and C are typically expressed by basic events or fault trees such that P(A)+P(B)+P(C) = 1. In Monte Carlo simulation using L1PSA code, basic events A, B and C usually are simulated independently, and P(A)+P(B)+P(C) = 1 does not hold. For each simulation run with level 1 analysis tool, sum of conditional probabilities is not 1. Instead of simulating conditional probabilities of entering branches, the whole model fluctuates, as shown in table below. In that case, the result will is not be meaningful.

Simulation run	P(A)	P(B)	P(C)	SUM
1	0.3	0.4	0.5	1.2
2	0.1	0.9	0.6	1.6
3	0.2	0.3	0.2	0.7

In principle, uncertainty analysis can be performed with a level 1 tool in the same way as with EVNTRE and LHS codes: by generating samples of entire data sets in such a way that the sum of conditional probabilities remains as 1. Depending on the amount of mutually exclusive events that need to be considered to get a reasonable accuracy in the uncertainty analysis, preparation of such sample sets may be tedious and impractical.

For physical phenomena, some PSA developers (e.g. AREVA) uses external Monte Carlo programs which generate samples of data which are then integrated in the PSA tool. For correlated phenomena like RPV rocket and DCH, a sample set as required here is readily available and can be directly used in the uncertainty analysis in RiskSpectrum PSA, automatically taking into account the correlation.

However, as long as each release category is dominated by the same type of event, then the quantitative impact on level 1 type uncertainty results is small. This can be checked by studying the importance of the events for each release category. The more specific L2PSA tools like EVNTREE and KANT can also handle uncertainties in the phenomena by the application of user defined functions in the branch points.

Problems related to minimisation

As the word states, minimal cut sets are based on minimisation. The minimisation is performed according to Boolean substitution laws.

Example: Consider Boolean simplification rule A+AX = A. This is based on truth table below:

Truth value of A	Truth value of X	Truth value of result
0	0	0
0	1	0
1	0	1
1	1	1

Truth value of X does not affect the truth value of the result (for example, “core damage occurs”), and the term containing X can be removed from equation A+AX.

Minimisation is safe as long as we deal with only two alternatives True and False, which is the case if the PSA model contains only one consequence. Then minimal cutsets are equivalent to each other in the sense that every minimal cutset leads to the same consequence. Here minimal cutsets are one-dimensional, since the only property that changes from one cut set to another is the frequency of the cut set.

The situation changes when multiple consequences are introduced, for example as Plant Damage States (PDS) or Release Categories (RC). In this case minimal cut sets do not hold just one True value, as Boolean logic requires. Instead, each minimal cut sets holds property Class, where Class is either PDS name or RC name. Inside each class, minimal cut sets hold equivalent definition of True, but between classes minimal cut sets are not comparable - which means they are not minimisable, since minimisation requires comparison. In practical PSA work, this may cause problems, which are illustrated with some examples below.

Generation of Plant Damage State Cut Sets

Figure 1 shows a simple level 1 event tree with 3 outcomes leading to CD. The initiating event is I, and the two questions contain systems A and B plus a common support system X. The end points of the event tree are classified to plant damage states PDS1 and PDS2.

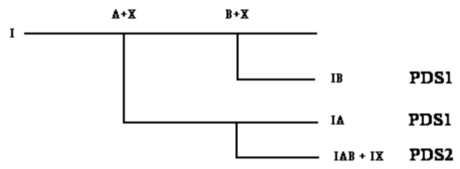


Figure 1. Simple event tree with removal of impossible cut sets.

The minimal cutsets leading to different sequences are listed at the end of each branch. The first PDS1 sequence contains only cutset IB. It does not contain cutset IX, since in the sequence the first question succeeds, implying that X does not fail. It would be logically impossible for X to succeed and fail in the same sequence. IX is deemed as illogical cutset and it is discarded from this sequence.

The second PDS1 sequence contains only IA. It does not contain cut set IX, since in the sequence the second question succeeds, implying that X does not fail. It would be logically impossible for X to fail and succeed in the same sequence. IX is deemed as illogical cutset and it is discarded from this sequence.

The PDS2 sequence contains cut set IX. In the event tree logic, if X is failed, PDS2 is the only possible end point, since X fails the first and the second questions.

Based on this discussion, we can draw the following conclusions:

1. It is incorrect to first generate minimal cutsets leading to CD and then to divide these cutsets to Plant Damage States, since this would cause minimisation of cutsets over different classes.

> **Example:** The (not minimal) cut sets leading to CD are IB + IA + IAB + IX. Minimal cut sets are IB + IA + IX. Thus, IAB would be lost from PDS2 due to minimising between PDS1 and PDS2.

2. To generate minimal cutsets for plant damage states, the cutsets must be generated separately for each PDS using at least logical success event removal for each sequence. It is incorrect to generate cutsets for different plant damage states by building simple fault tree logic of failure branch points of sequences leading to each PDS.

> **Example:** Simple fault tree logic for PDS1 is formed by forming an OR-gate that contains all accident sequences that belong to PDS1: TOP=I*(A+X) + I*(B+X). This leads to minimal cutsets IA + IB + IX. Thus, cutset IX incorrectly appears in PDS1, although it is logical impossible. The impossibility of IX can be detected only if the fault tree logic for PDS1 is expressed as per sequence: I*not(A+X)*(B+X) + I*(A+X)*not(B+X), i.e. taking success path information into account. As mentioned above, this is usually standard procedure in PSA quantification.

3. Since it is impossible to include success states in all minimal cut sets of practical PSA models, as this would lead to extremely long minimal cut sets, the sum of PDS frequencies will be larger than the total CD frequency. The model can be created in such a way, that this problem does not arise for normal component and system availabilities. In case of large probabilities (more common in L2PSA), there is one solution to create separate basic events for each success branch, thereby making sure that the total branch probability is 1.0.

Most L1PSA tools can generate correct minimal cut sets for plant damage states, given that the user defines correct calculation settings. The examples above are demonstrations of pitfalls that can produce adverse effects if calculation is not done in correct order or with correct settings.

Generation of Release Category Cut Sets

Similar problems as for PDS cutsets, also exist in using minimal cut sets for solution of the release categories in APETs.

Consider the following example:

Cut set ABC from level 2 sequence 1 leads to a minor release, and cut set ABCX from sequence 2 leads to large release (X could be hydrogen burn, for example). If sequences 1 and 2 are classified to the same release category, their cut sets are merged, the latter cut set is minimised, and it never finds its way to the output. In the result, *there remains nothing to reveal that hydrogen burn and large release were present*.

The only way to find missing hydrogen burn and large release cut set ABCX is to refine classification in such a way that ABC and ABCX enter different release categories, and ABC does not minimise ABCX. In principle, it may be possible to avoid this problem by structuring the APET in such way that questions dealing with large releases are asked before questions of small releases, but this may introduce other modelling problems.

It has to be noted that it is part of the analytical work to make sure that sequences are assigned to the correct release categories and to if needed split a release category into several or combine release catefories that are enough similar. A release category frequency dominated by sequences with smaller release compared to the sequences with large release may be conservative or non-conservative depending on the sequence chosen as representative for the release category.

2.6.3 Plant damage states quantification and results presentation

Quantification of the plant damage states is usually performed to provide intermediate results, in support of screening and for checking of the model. The quantification is made in the L1PSA model, thus there are no dependencies between the L1 and L2PSA that need consideration. However, it is important to consider success events in the paths to have results as accurate as possible for the sequence and PDS frequencies, e.g. sequence/PDS minimal cut sets need to be checked with regard to mutually exclusive events. The kinds of results that usually are calculated and presented include:

- The core damage frequency,
- The conditional core damage probability given the initiating events,
- The plant damage state frequencies,
- The conditional plant damage state probability given the initiating events.

Plant Damage State (PDS) matrices provide useful evaluation tools and sources of information. PDS matrices can present absolute or relative values, i.e. with frequency or percent values. Below are examples of relative PDS matrices for Olkiluoto 1 unit (2001) from different points of view. Each matrix provides different insights into L1/L2 interface. The matrices provide also a good reasonability/credibility check. Especially if the PDS classification and calculation is performed manually, it is recommended to prepare the matrices first using absolute values to check that all sums agree, and then to prepare the relative matrices.

The first matrix in Table 7 is a summation over initiating events and plant damage states. It represents the relative contribution of each initiating event in each PDS. The rightmost column displays the relative contributions of each initiating event, and the bottom row displays the contributions of each PDS. The sum of all cells is 100%. The matrix is shown in graphical form in Fig. 10 The second matrix in Table 8 displays the conditional probabilities of each initiating event to enter each PDS. The sum over each initiating event is 100%. The third matrix in Table 9 displays the relative contribution of each initiating event in each PDS. The sum over each PDS is 100%.

Table 7 PDS matrix of Olkiluoto 1 displaying relative contributions of each initiating event (left) and PDS (top) to total frequency. The sum of all cells is 100%.

IE \ PDS	CBP	COP	FCF	HPL	HPT	LPL	LPT	RCO	RHL	RHT	ROP	VEN	VLL	Total
ERT-T0					0.00		0.00			0.02	0.00		0.00	0.02
ERT-T1A					0.00		0.00			0.00	0.00			0.00
ERT-T1B	0.00													0.00
ERT-T2	0.01													0.01
ERT-T3	0.00													0.00
EXT-TE					0.06		**5.97**	0.30		0.06	0.00		0.00	6.39
EXT-TF					0.70		0.26	0.20		**2.63**	0.02		0.00	3.82
EXT-TP					0.01		0.01			0.01	0.00		0.00	0.03
INT-A0		0.06				0.00		0.00	0.01					0.07
INT-R0						0.35								0.35
INT-S1		0.00		0.03		0.11		0.06	0.10		0.00			0.31
INT-S2		0.00		0.01		0.29		0.17	0.15		0.00			0.63
INT-T3					0.17		0.00	0.04		0.00			0.00	0.21
INT-TE					**1.70**		**5.24**	0.01		0.45	0.02		0.00	7.42
INT-TF					**2.40**		0.04	0.79		0.09	0.08		0.00	3.40
INT-TP					**1.02**		**1.08**			0.45	0.03		0.00	2.58
INT-TT					0.10		0.03	0.22		0.10	0.15		0.00	0.60
INT-Y							0.35							0.35
IRT-A0	0.00		**14.17**											14.17
IRT-A1H	0.00													0.00
IRT-A1L	0.01													0.01
IRT-A2H	0.19													0.19
IRT-A2L	0.01													0.01
IRT-A3	0.00													0.00
IRT-A4	0.00													0.00
IRT-B0	**59.33**													59.33
IRT-B3	0.06													0.06
IRT-B4	0.04													0.04
IRT-T0					0.00		0.00			0.00	0.00		0.00	0.00
IRT-T1A					0.00		0.01			0.00	0.00			0.01
IRT-T1B	0.00													0.00
IRT-T2	0.00													0.00
IRT-T3	0.00													0.00
Total	59.65	0.07	14.17	0.04	6.15	0.75	12.99	1.79	0.26	3.82	0.30		0.00	100.00

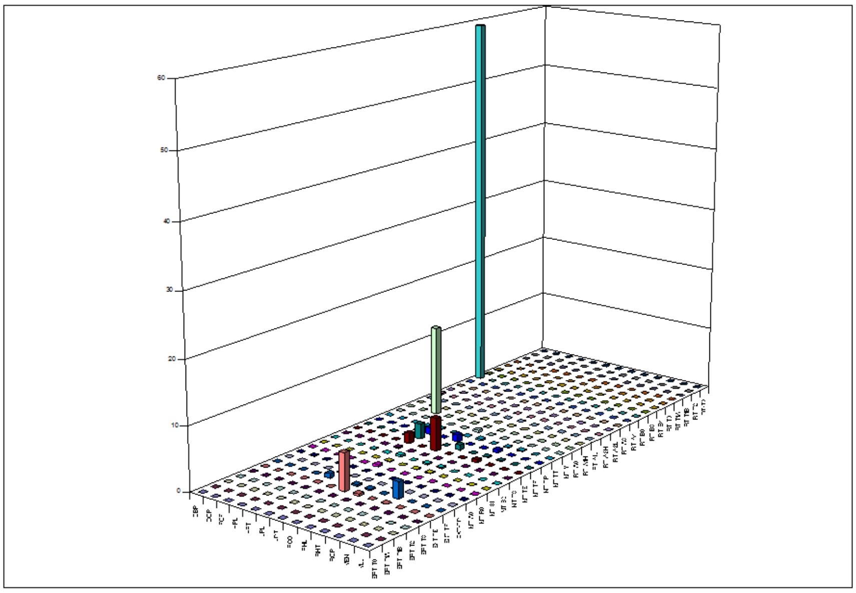


Fig. 10 PDS matrix of figure Y2 in graphical form.

Table 8 PDS matrix of Olkiluoto 1 displaying conditional probability of each initiating event (left) to enter each PDS (top). The sum of each row (each initiating event) is 100%.

	CBP	COP	FCF	HPL	HPT	LPL	LPT	RCO	RHL	RHT	ROP	VEN	VLL
ERT-T0					1.28		6.15			**92.21**	0.34		0.03
ERT-T1A					7.98		**49.00**			**42.66**	0.35		
ERT-T1B	**100.00**												
ERT-T2	**100.00**												
ERT-T3	**100.00**												
EXT-TE					0.93		**93.50**	4.66		0.91	0.00		0.00
EXT-TF					18.28		6.89	5.27		**69.01**	0.55		0.00
EXT-TP					18.75		**36.05**			**44.62**	0.57		0.00
INT-A0		**88.55**				1.32		0.77	9.36				
INT-R0						**100.00**							
INT-S1		1.05		8.45		**36.56**		19.92	**33.93**		0.09		
INT-S2		0.73		2.02		**46.01**		27.06	24.06		0.12		
INT-T3					**80.44**		0.83	17.62		1.11			0.00
INT-TE					22.89		**70.59**	0.18		6.09	0.24		0.00
INT-TF					**70.47**		1.12	23.33		2.62	2.45		0.00
INT-TP					**39.50**		**41.71**			17.53	1.26		0.00
INT-TT					16.96		5.41	**36.40**		16.81	24.41		0.00
INT-Y							**100.00**						
IRT-A0	0.00		**100.00**										
IRT-A1H	**100.00**												
IRT-A1L	**100.00**												
IRT-A2H	**100.00**												
IRT-A2L	**100.00**												
IRT-A3	**100.00**												
IRT-A4	**100.00**												
IRT-B0	**100.00**												
IRT-B3	**100.00**												
IRT-B4	**100.00**												
IRT-T0					20.73		**43.05**			24.48	11.74		0.00
IRT-T1A					9.25		**77.12**			12.69	0.94		
IRT-T1B	**100.00**												
IRT-T2	**100.00**												
IRT-T3	**100.00**												

Table 9 PDS matrix of Olkiluoto 1 displaying relative contribution of each initiating event (left) to each PDS (top). The sum of each column (each PDS) is 100%.

	CBP	COP	FCF	HPL	HPT	LPL	LPT	RCO	RHL	RHT	ROP	VEN	VLL
ERT-T0					0.00		0.01			0.42	0.02		7.40
ERT-T1A					0.00		0.01			0.03	0.00		
ERT-T1B	0.00												
ERT-T2	0.01												
ERT-T3	0.00												
EXT-TE					0.97		**45.99**	16.58		1.53	0.02		2.67
EXT-TF					11.35		2.02	11.22		**68.91**	6.88		**55.94**
EXT-TP					0.09		0.09			0.36	0.06		0.69
INT-A0		**88.92**				0.12		0.03	2.52				
INT-R0						**46.24**							
INT-S1		4.56		**66.89**		14.90		3.40	**39.55**		0.09		
INT-S2		6.52		33.11		**38.74**		9.55	**57.93**		0.25		
INT-T3					2.72		0.01	2.04		0.06			0.21
INT-TE					27.63		**40.31**	0.76		11.82	5.99		13.07
INT-TF					**38.98**		0.29	**44.22**		2.34	27.43		4.71
INT-TP					16.57		8.28			11.83	10.75		3.73
INT-TT					1.66		0.25	12.20		2.64	**48.41**		11.50
INT-Y							2.66						
IRT-A0	0.00		**100.00**										
IRT-A1H	0.00												
IRT-A1L	0.02												
IRT-A2H	0.32												
IRT-A2L	0.01												
IRT-A3	0.00												
IRT-A4	0.00												
IRT-B0	**99.46**												
IRT-B3	0.10												
IRT-B4	0.07												
IRT-T0					0.01		0.01			0.01	0.06		0.09
IRT-T1A					0.02		0.07			0.04	0.04		
IRT-T1B	0.00												
IRT-T2	0.00												
IRT-T3	0.00												

2.6.4 Release frequencies quantification and results presentation

This question is mainly detailed in Volume 1 chapter 5 and 6. Some additional comments are provided here.

The source terms and frequencies of the individual Release Categories should be used to determine the summated frequencies of different types of accidents for comparison with numerical safety criteria where they exist. These would typically be in the form of a frequency target for LERF / LRF; however, in some regulatory frameworks, true risk targets are also used. Whatever the risk metric, the magnitude and characteristics of the environmental releases provide an important input to the assessment of risk in their own right.

Another format for displaying source term results and comparing with safety criteria is a complementary cumulative frequency distribution (CCFD), based on the frequency of releases exceeding X, where X varies from the smallest to the largest postulated magnitude of offsite release, typically expressed as a group release fraction for radiologically significant isotopes. For this purpose, the frequency of exceeding a given fractional release should typically be provided, together with the statistical significance (e.g. mean, median, 95th percentile), if available. This format of presentation of results conforms to the presentation of results of PSAs for other types of systems (such as for the air transportation safety analyses).

Generally, the frequency and magnitude of offsite releases should be considered together for the interpretation of the PSA and its applications, because the product of these two quantities defines the overall risk from the plant. The insights gained from this quantitative evaluation of radionuclide releases should be discussed together with the results of the uncertainty analysis.

The scope of this guideline does not extend to Level 3 PSA; however, CCFDs for released activity may also provide insights about potential offsite consequences (this is a metric that only applies to extended PSAs, or Level 2+). To achieve this, the core inventories for each of the individual radionuclides represented in the fission product groups reported in the source term assessment should be known. Dedicated tools are able to calculate the core inventories (e.g. ORIGEN developed by Oak Ridge National Laboratory, DARWIN/CRISTAL developed by CEA supported by IRSN/EDF/AREVA). Some estimates are typically given in the Final Safety Analysis Reports (FSARs), and these should be used for a selected number of radionuclides.

In addition there may be some simplified deterministic calculations of offsite doses, calculated in a manner similar to the offsite doses as calculated for Design Basis Accidents. If this approach is used, it is recommended that the IAEA guidelines for estimates of doses in safety demonstrations (see for instance among other IAEA safety series documents, IAEA-TECDOC 953/1997 [21] and IAEA-TECDOC-955/1997 [22]) should be followed. This approach is a surrogate for Level 3 PSAs, without the need to assess in detail the site specific issues. Such results can be used to assess:

- Whether a Release Category falls above or below a pre-defined dose threshold for members of the public (compliance with safety limits),
- Ranking of Release Categories,
- Give importance measures for systems, events and phenomena,
- Show importance of SAM measures.

However, care should be taken in quoting individual doses resulting from severe accidents as the concept of the dose unit Sievert breaks down at levels in the region of 1 Sv and the dose unit Gray applies at higher exposures. Also, individual doses calculated for members of the public close to the site far above the threshold for early health effects, or even early fatalities, are not very meaningful. At these extreme levels the concept of societal harm (for example in terms of the expected number of fatalities) is more meaningful.

2.6.4.1 Example - L2PSA for French 900 MWe NPP (IRSN modelling)

In the IRSN L2PSA for 900 MWe PWR, and as mentioned in sub-section 7.5.5.1, Release Categories are defined by 37 specific parameters related to initial reactor state, Containment Failure Modes (CFMs), accident kinetics.

Due to the large number of modalities for each parameter, several thousands of Release Categories are generated by the Accident Progression Event Tree (APET) quantification. To make the final presentation easier, the thousands of Release Categories generated by the quantification of the L2PSA model are gathered into larger categories, called “Regrouped Release Categories” (R-RCs).

These larger categories are defined as a function of the considered CFMs, phase of accident, delay between the initiating event and the beginning of atmospheric releases.

For each sequence in the IRSN APET, the quantification of all top events (severe accident phenomena, human actions, systems behaviour in severe accident conditions) along the complete time evolution is performed even if a first CFM is obtained early in a sequence. Therefore, many L2PSA sequences include several CFMs.

To present the R-RCs frequencies, two methods are used at IRSN [23].

- Method 1: R-RCs are defined by the first CFM in the accident progression (classic method). In this case, the sum of all R-RC frequencies is equal to the total core damage frequency from L1-L2PSA,
- Method 2: R-RCs include all L2PSA sequences having one particular CFM, even if this CFM is not the first one to occur. In this case, the sum of all R-RCs is higher than the core damage frequency from L1-L2PSA.

With the classic method, later CFMs, even although leading to important radioactive releases, may be hidden by earlier ones. But the IRSN second method allows for calculation of the real frequency of late CFMs.

To present the R-RCs releases and doses, representative RCs are defined for R-RCs, and calculations, taking into account uncertainties, are performed for these RCs (short and long term consequences in the vicinity of the damaged plant).

The dose considered for analysis is the total effective dose equivalent, integrated over 15 days by a one year old child at 2 km from the damaged plant.

Frequency of exceeding a given dose is provided (CCFD), and to rank the R-RCs, a risk measure has been defined, consisting of the product of the frequency of R-RC with the dose defined above.

Moreover, the R-RCs are gathered into "classes" (9 classes for the 900 MWe L2PSA) according to the dose magnitude, and are represented on a frequency-consequence diagram (dose vs. frequency).

Regarding the emergency measures to be taken in case of radioactive release, the following calculations have been performed for each R-RC:

- For a given distance to the damaged plant, delay before which these measures have to be taken,
- For a given delay after the beginning of accident, distance below which emergency measures have to be taken.

Regarding the long term consequences, the extent of contaminated lands has been assessed (based on Cs 137 isotope and contamination thresholds defined following the Chernobyl accident).

2.6.4.2 <u>Example - Generic Iso-risk Contours for major hazard industries</u>

The term '*individual risk*' is sometimes used in relation to risk criteria. The usual definition of individual risk is: The likelihood that an individual will experience an adverse effect (the type of adverse effects needs to be defined additionally, e.g. fatality or the chance to receive a dangerous dose following a given exposure).

Application of this person-linked definition is problematic, as it involves assumptions about the movements and presence of individuals, which are not always fundamental to understanding the risk situation around the source, for example as would be needed as an input to emergency planning.

Some countries, e.g. the Netherlands, use the concept of location-based risk. Location-based risk is calculated as the risk that a person who is continually present and unprotected at a given location will die as a result of an accident within the establishment. Hence, location-based risk describes the geographic distribution of risk for the establishment in question. When more quantitative frequency classification is introduced, as would be produces from a L2PSA, it is possible to produce iso-risk curves. Again it is not dependent on whether people or residences are present (see Figure 11) [28].

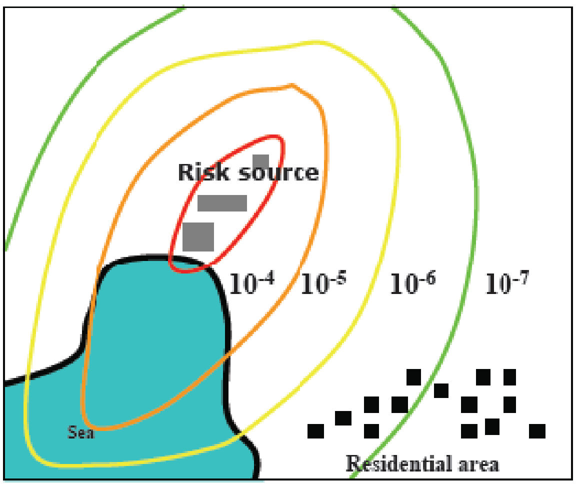


Fig. 11 Example of individual iso-risk contours reproduced from [28]

2.7 CONSIDERATION OF UNCERTAINTIES

2.7.1 Overview

There are numerous sources of uncertainty in L2PSA, and consequently the results of L2PSA are also subject to uncertainty. Uncertainty is not limited to L2PSA : L1PSA typically indicates a large range of uncertainty about the frequencies of plant damage states.

There is a widespread notion that L2PSA issues are particularly uncertain. But the relative degree of uncertainty in L1PSA respective in L2PSA is very much plant specific. There are, for example, results of PSA which indicate that the uncertainties from L1PSA are by far more significant than from L2PSA.

This tendency can, for example of some particular plant design, be due to the fact that the failure of the containment under core melt conditions in L2PSA is almost certain, while there is uncertainty for the failure probability of safety systems in L1PSA.

The degree of uncertainty may be very different, and uncertainties of the accident sequence progression do not necessarily lead to uncertainties of the L2PSA results: imagine formation of a leak in a pressurised containment, and discharge into the surrounding reactor building. It may be very uncertain which of the building doors or ventilation ducts leading to the environment would fail first, but on the other hand it is certain that at least one of them fails. In this case, the radioactive source term into the environment will probably not be very much affected by details of the flow path through the buildings. Therefore, in this example the source term is affected by little uncertainty, while details of the previous events inside the plant could be very uncertain.

L2PSA projects occasionally refrain from explicitly performing uncertainty analyses because the required resources are considered substantial, and / or because the uncertainties deem to be not quantifiable. The second argument is not very convincing, if at the same time the analysis is presented with “best estimate” or “point values”: how could one defend, for example, a point value for a containment failure probability of 10%, if no idea exists about the uncertainty range of that value? The issue of limited resources may be more serious. Uncertainty analysis means identifying the uncertain input elements, quantifying their range of uncertainty including the distribution, and application of rather conventional Monte Carlo tools. There is no principal methodological problem involved in explicitly considering uncertainties. The difficulty is mainly in the formalisation of “what is known” in the distribution function.

Given these conditions, performing an uncertainty analysis becomes a cost-benefit issue. The cost depends on the precision which is required, but according to experience, it can be up to additional 30% compared to a point value analysis. The benefit depends on the objectives of the PSA, and on some (plant) specific PSA results, see below.

If the objectives involve comparisons of L2PSA results with any quantitative criterion (e.g. to exceed a certain release quantity, or to limit the containment failure probability), taking uncertainties into account seems to be indispensable. The mean values of L2PSA results may be well below given limits, but the distribution tails may be above the limits. Such cases need to be justified .

If the objective is towards identifying most important contributors to consequences of interest, or the identification of critical equipment / procedures / phenomena, an analysis without explicitly considering uncertainties may be misleading.

A third argument for considering uncertainty will become clear with the following example: imagine a PSA where point value analysis indicates a best-estimate (beyond design) containment failure pressure of 1.0 MPa, and a containment pressure peak below 0.8 MPa after hydrogen combustion. This PSA will indicate zero containment failure probability due to hydrogen combustion, and no related release categorie at all. However, taking into account uncertainties, containment failure could no longer be excluded, and the related consequences would be orders of magnitude higher. This leads to this simple rule: to avoid that point value analysis hide important insights, low frequency of accident with high amplitude consequences should be systematically confirmed by uncertainty analysis.

From the many sources of uncertainty, the dominant ones have to be identified and their impact on the PSA results have to be quantified.

Two important tasks have to be distinguished within this context:

- <u>Sensitivity analysis</u> is the task of identifying those sources of uncertainty which have the most significant impact on the results of interest,

- Uncertainty analysis is the task of identifying the uncertainty of the result of interest due to the main sources of uncertainty.

2.7.2 Sources of uncertainty

There are (in principle) two types of analyses which require uncertainties to be taken into account:

a) Deterministic analyses of the accident progression and source term using mostly thermal hydraulic and source term integral codes for simulation of accidents (e.g. with MELCOR, MAAP, ASTEC), but also more specific tools (FE codes for beyond design behaviour of the containment structures, codes for energetic phenomena assessment (steam explosion, DCH, …), etc …). Uncertainties within such deterministic analyses should be assessed by appropriate variation of uncertain input parameters, or by selecting different submodels (if provided by the code).

b) Determination of the probability of occurrence of specific events (human actions, failure of a component, hydrogen ignition, …). Uncertainties of this type will be addressed within the probabilistic event tree analysis. In addition, uncertainties inherent in the deterministic analyses and their results finally have to be adequately transferred to the event tree analysis.

The different sources of uncertainties can also be classified as epistemic uncertainties or aleatory (stochastic) uncertainties. Uncertainties due to lack of knowledge are called epistemic uncertainties and uncertainties due to the stochastic nature of some phenomena or events are called aleatory uncertainties. Nevertheless, classification between epistemic and aleatory turns out to be delicate to perform in practice.
Uncertainties to be considered in L2PSA are of different origins. Examples are provided hereafter.

Completeness

The main trust of the PSA model is to assess the possible scenarios (sequences of events) that can lead to releases of radionuclides. However, there is no guarantee that this process can ever be complete and that all possible scenarios have been identified and properly assessed. This potential lack of completeness introduces an uncertainty in the results and conclusions of the analysis that is difficult to assess or quantify. However, extensive peer review can reduce this type of uncertainty.

Model uncertainties coming from L1PSA results

If quantified, the frequency distribution of each PDS can be considered as a consequence of uncertainties inherent in the L1PSA and which are transferred to L2PSA via the L1–L2PSA interface.

Model uncertainties coming from practical limitations of the L2PSA event tree methodology

One example of model uncertainty is loss of detail due to aggregation. Grouping L1PSA sequences or cut sets into PDSs as the input to the L2PSA for practical reasons also introduces uncertainties due to the resulting loss of some modelling detail.
Further, the APET binning process in RCs introduces uncertainty through the possibility that the attributes used by the analyst to group 'similar' accident progressions are incomplete. These elements of uncertainty are also difficult or not possible to quantify.

The only solution to reduce this source of uncertainty is to increase the complexity/completeness of the global L2PSA: a higher number of PDS, associated to more thermalhydraulics/source term deterministic calculations or a higher number of RC and source term calculations can reduce this type of uncertainty by taking into account the specificities of the numerous accidental sequences. The difficulty is here to find a compromise with the resources available to develop the study. In addition, very detailed APET results may be difficult to understand and they may be difficult to evaluate.

Model uncertainties coming from the precision of the deterministic codes or real knowledge

Another model uncertainty example concerns the uncertainties in computational models that arise from either incomplete knowledge of the phenomena or to inadequacies or simplifications introduced into the mathematical models of the phenomena.

In some cases, scale effects from application of available experimental data to full-scale reactor conditions can introduce uncertainties.

Moreover, experiments simulating severe accidents are difficult to carry out under reactor specific conditions.

Parameter uncertainty

Parameter uncertainty refers to the uncertainty if an influential parameter value can only be represented by an uncertainty distribution.

2.7.3 Identification of key uncertainties by means of sensitivity analyses

The first step is obviously the identification of the key issues where uncertainties should be taken into consideration because they can modify the final conclusion of the study. Detailed information is provided in chapter 3 to 7 on uncertainties to be taken into account in the different chapters.

Sensitivity analysis in deterministic calculation is a useful tool to guide the selection of sources of major uncertainties. There are two approaches: In the first approach, a single uncertain parameter is modified, and the analysis is re-run with this single modification. This method will provide the influence of the single uncertain parameter, given that all other uncertain parameters remain unchanged. In the second approach, numerous uncertain parameters are changed at the same time, and the analysis is re-run many times with different sets of uncertain parameters. Regression analysis will then identify the degree of influence of the varied parameters on the result of interest, taking into account the variability of all other parameters. Obviously both approaches require a considerable number of deterministic calculations. Depending on the complexity of the model (e.g. an integral accident analysis code) a formally satisfying identification of key uncertainties may become impractical. In this case, experience of the PSA staff will be needed for identifying the most relevant uncertainties.

Sensitivity analysis in the probabilistic calculation is useful to identify which factors have the largest influence on the frequencies calculated. In principle, the same approaches can be applied as for the deterministic analyses, as described in the previous section: In the first approach, a single uncertain branching probability of the APET would be varied, and the influence of this variation on the result (e.g. on the frequency of a release category) would be identified. In the second approach, many uncertain branching probabilities would be varied at the same time, and numerous such sets of branching probabilities would be used to run the APET quantification numerous times. By appropriate regression analysis, the most significant uncertain branching probabilities could be identified. In contrast to the deterministic analyses mentioned above, the effort for performing APET calculations is generally much smaller. Therefore, a formal sensitivity analysis of the uncertainties in the APET is recommended.

If the uncertainty of some phenomena is shown to have low influence on the uncertainty of results of interest, the further treatment of these issues can be based on best estimate point values.

2.7.4 Uncertainty Analysis by means of a Monte-Carlo simulation with the probabilistic event tree tools

2.7.4.1 <u>Introduction of uncertain parameters with uncertainties distribution in the APET</u>

The probabilistic tools (Riskspectrum, EVNTRE, KANT, SPSA) considered here all include a Monte-Carlo algorithm that should be simple to use to quantify uncertainty distribution of the RC frequencies and other results of the event tree analysis. The main difficulty for the L2PSA developer is to introduce relevant distributions of probability for each uncertain key parameters of the APET. This distribution of probability may come from expert judgement or be justified by dedicated studies in support of the L2PSA APET. It is highly recommended to develop a specific methodology to make the link between sensitivity / deterministic analysis and final event tree modelling with distribution function on uncertain parameters.
Monte Carlo simulations should be applied to determine the uncertainty of the results of interest (uncertainty analysis).

Example
An example of a method based on users function was described in [6] for the L2PSA developed by IRSN for the French PWR L2PSA.

Models are constructed with respect of the simple rules explained below:

- Two separate physical models are linked by a limited number of state variables transferred by the APET,
- These state variables provide relevant information on the plant state for the evaluated physical phenomena: physical conditions (primary pressure for e.g.) or systems information (pressurizer valves aperture for e.g.),
- Other variables are used to take into account uncertainties on the result and are defined by a probabilities distribution; these uncertain variables can have different natures and have to be discussed by experts; in the quantification of the APET, the values taken by these variables are obtained by a Monte-Carlo method.

Uncertain variables can cover all type of uncertainties, for example:

- Parameters of a sophisticated code correlation or model which is obviously (from expert's judgment) not well-known but has a strong impact on the results (for example, heat exchange coefficient between corium particles and containment atmosphere in a direct containment heating model),
- Parameters related to a system function (for example, the RCS pressure when controlled by open pressuriser safety valves – the valves close themselves below a pressure threshold which uncertain),
- Expert's judgment on the accuracy of a code result (e.g. mass of corium relocated in the vessel bottom at vessel rupture or hydrogen mass in containment at vessel rupture depending on partial combustion),
- Statistical uncertainties due to the construction of the model.

Fig. 12 gives a schematic view of a APET physical model :

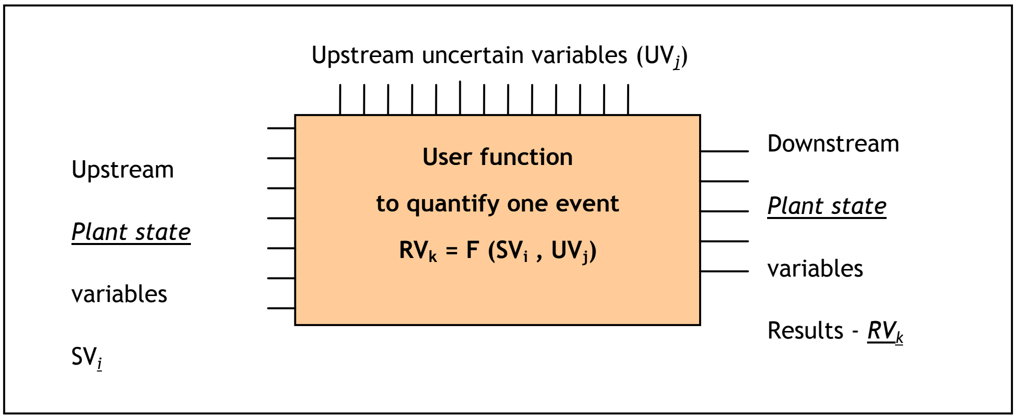


Fig. 12 Example of User-Function modelling

This scheme was found to be simple and robust enough to model all the different situations encountered in the L2PSA APET modelling.

2.7.4.2 Specific recommendations to calculate uncertainties distributions

- When carrying out analyses considering uncertainties, parameter dependencies have to be maintained. Otherwise it is possible to combine cases that are not realistic, or they may even be physically impossible. Another important issue is maintaining individual probabilities and overall frequencies. The sum of the probabilities for a single branching point in an event tree has to be 1.0, and the sum of frequencies of the end branches in the APET has to be the same as the sum of PDSs leading to the branches.
- As far as possible, the parameters to be varied should be primary ones, and not of the integral type. For example, a containment leak rate depends on containment pressure, gas composition, temperature and leak size, and thus the leak rate should not be the parameter to be varied, but leak size should be used instead.
- Integral parameters can be applied if adequately supported by specific separate uncertainty analyses. Use of specific studies may be advisable in some cases to simplify the overall uncertainty study and to reduce the calculation effort. Nevertheless, the dependencies must be maintained; for the case described above,

the leak rate can be an uncertainty parameter but its dependency on other parameters has to be considered.

- It is advisable to make the uncertainty analyses integral ones to take into account the dependencies and feedback of the system. The uncertainty of source term analysis for example could be dealt with separately for the parameters affecting the source term model itself, but then the feedback of the fission product release on the thermal-hydraulics of the RCS and containment is lost, unless the model is coupled with the thermal-hydraulics modelling.
- Integral uncertainty analyses benefit from linking the source term analysis directly with the APET calculation,
- There are several approaches for evaluation of uncertainty. The Monte Carlo method requires numerous repetitions of individual calculations with different sets of parameter combinations. Of course, there are methods to enhance the calculation speed and the reliability of the Monte Carlo calculations, such as biasing, Russian roulette, and multiplication that are used in particle transport programs. However their application in PSA calculations is rarely considered,
- Reliability and confidence levels of a study may be achieved by refining pure Monte Carlo calculations, as well. Latin hypercube sampling (LHS) introduces parameter selection in a proper way that the probability distribution range of each individual parameter will be covered. This method can be considered as an enhancement of Monte Carlo technique, as it will significantly increase the reliability of the results with less calculation effort than pure random sampling,
- Input distributions must not be truncated to make the extreme values of the parameters possible in the analyses. If this is not taken care of, possible overlapping of parameters leading to e.g. containment failure might be ignored. These kinds of parameters are for example containment pressure distribution and containment fragility curve.

2.7.4.3 Specific recommendations to quantify an APET with Monte Carlo algorithm

The number of Monte-Carlo runs should be sufficient to capture all information, especially for the rare events. For example, in case of containment failure due to high pressure, the event may occur with maximum pressure (in the sense of density of probability) and minimum containment strength. In that case, the use of containment fragility curve instead of distribution of probability allows decreasing the number of Monte Carlo runs. Mathematical methods exist to estimate the required number of Monte-Carlo simulations, but one can also simply check the stability of the L2PSA results in function of the number of simulations.

2.7.4.4 APET results presentation with uncertainties

Uncertainties regarding APET results (L2PSA sequences or release category frequencies) are generally obtained by the use of Monte-Carlo methods, either by simple random sampling or stratified sampling (Latin hypercube). This method allows an easy propagation of uncertainties through APET quantification.

For any result, a probabilistic distribution is generated and cumulative probability levels for the results can be calculated (e.g. the 5th, 50th and 95th percentiles representing 5%, 50% and 95% probability that the ‘true’ result is below the respective level at which each of these probabilities is stated).

For a L2PSA with quantitative assessment of uncertainties, the results are displayed using histograms, probability density functions, cumulative distribution functions and tabular formats showing the various quantiles of the calculated uncertainties, together with the distribution mean and median estimates.

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3 HUMAN RELIABILITY ANALYSIS

3.1 INTRODUCTION

This chapter gives an overview on performing human reliability analysis (HRA) in L2PSA. The objective of human reliability analysis in the context of the PSA is to identify, represent (in the logic structure of the PSA) and analyse (quantify) all human errors, before and during the accident, which contribute to plant risk as defined in the PSA. Section 3.2 gives an overview of the HRA process and methodology structure. PSA practitioners can also choose not to give credit for certain operator actions for various reasons, for example conservative assumptions, low importance, etc. In those cases HRA will be limited.

In the context of L2PSA, HRA mainly focuses on the analysis of the severe accident management (SAM) actions taken into account in the accident scenarios. The analysis is similar to the analysis of post-initiator actions carried out in L1PSA, even though SAM actions have several features different from actions prescribed in Emergency Operating Procedures (EOPs). These issues are discussed in Section 3.3. Section 3.4 provides HRA examples from L2PSA, and section 3.5 presents a summary of selected methodologies.

3.2 OVERVIEW OF HRA PROCESS AND METHODS

Fig. 13 illustrates the human-machine interaction, where a human being is seen as an actor that supervises the process and who acts based on observation and a cognitive process leading to a decision of appropriate actions. The possibilities to succeed or fail in this interaction are seen to be dependent on factors, called performance shaping factors (PSFs), such as stress, time window and quality of human-machine interface. In HRA, both the human-machine interaction and the PSFs are addressed.

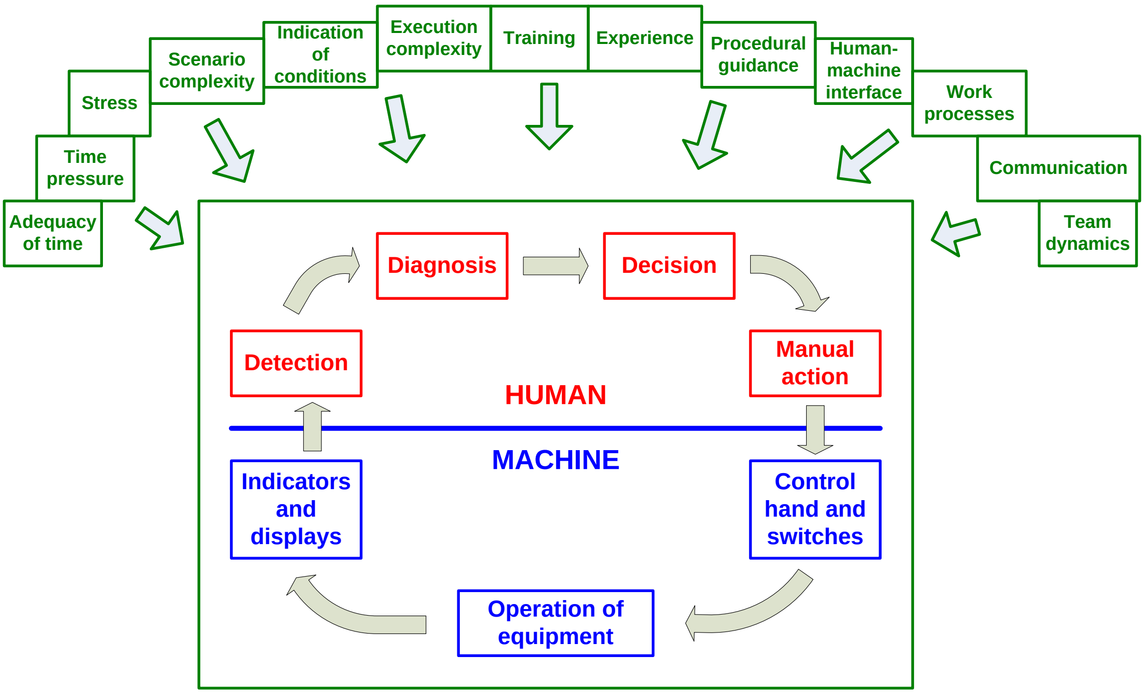

Fig. 13 Human-machine interaction cycle and performance shaping factors accounted in human reliability analysis.

The main steps of HRA of PSA include:

- Definition of system failures of interest. This refers to system functions considered in PSA,
- Identification of related human interactions. This task is a joint effort with systems analyses and sequence analyses and should lead to a number of relevant human interactions to be modelled in PSA,
- Qualitative analysis of human interactions. This includes identification and characterisation of the human-process interaction and PSFs so that the quantification can be performed. This step is very important when identifying recommendations,
- Quantification of relevant human errors. This step is particularly dependent on the HRA methodology used,
- Integration of results from HRA in the PSA model, including for example evaluation of effects of human errors on the system failure events,
- Analysis of dependencies. The dependencies between human actions may have already been identified and handled in earlier steps, but it is recommended to check the effect of dependencies when the actions are integrated in the PSA model,
- Sensitivity analyses using the PSA model,
- Interpretation of results, identification of recommendations, possible recalculation and modelling of actions,
- Production of documentation,
- Internal review. If possible the operating and maintenance personnel of the plant should be involved in the review.

The steps may vary depending on the work flow of the PSA project and the initial status of the PSA model (e.g. start from scratch or update of a living PSA).

There are several HRA methodologies in industrial use and a number of others under development but not yet applied in a full-scale PSA. Those methods have some unique and some common features, and although a comparison can be made of these methodologies it is difficult to identify a methodology which is clearly better than others. [38] identified 17 HRA methodologies considered to be of potential use in major hazards directorates. The methodologies were classified into first generation, second generation and expert judgment based methodologies.

The first generation methodologies were the first to be developed to help risk assessors predict and quantify the likelihood of human error. The well known first generation HRA methodologies in the context of PSA include:

- THERP, Technique for Human Error Rate Prediction [50],
- SHARP, Systematic Human-Action-Reliability Procedure [42],
- ASEP, Accident Sequence Evaluation Program [49],
- SPAR-H, Standardised Plant Analysis Risk HRA [41].

In the first generation approaches, the human interactions are analysed using a task analysis technique in which a task is broken down into subtasks and then the PSFs such as time pressure, equipment design and stress are considered. By combining these elements, the assessor can determine a nominal Human Error Potential. First generation methodologies focus on the skill and rule base level of human action and are often criticised for failing to consider such things as the impact of context, organisational factors and Errors Of Commission (EOC). Despite these criticisms they are useful and many are in regular use for quantitative risk assessments.

The second generation HRA methodologies are an attempt to consider context and EOC in human error prediction. The well known second generation HRA methodologies in PSA context include:

- ATHEANA, A Technique for Human Event ANAlysis [52],
- CREAM, Cognitive Reliability and Error Analysis Method [44],
- MERMOS [45], [46], [46],
- CAHR, Connectionism Assessment of Human Reliability [48].

The second generation methodologies are still generally considered to be under development but in their current form they can provide useful insight to human reliability issues. MERMOS is one in regular use, by EDF (Electricité de France) where the method was developed. The second generation methodologies pay more attention to the cognitive portion of Human Failure Events (HFEs). An important feature in more recent methodologies is the emphasis of dependency.

Expert judgment based methodologies refer to a structured approach for experts to consider how likely an error is in a particular scenario. Success Likelihood Index Method using the Multi-Attribute Utility Decomposition (SLIM-MAUD) is perhaps the best known methodology of this type in the context of PSA [40].

3.3 HUMAN ACTIONS IN L2PSA

3.3.1 Human action categories

Humans play a significant role both in the cause of accidents and in emergency response. The human actions can be classified in the same way for both L1PSA and L2PSA:

- Pre-initiators consist of those actions associated with maintenance and testing that degrade a system's availability. They may cause failure of a component or component group or may leave components in an inoperable condition,
- Initiators are actions contributing directly to initiating events,
- Post-initiators are the actions involved in operator response to an accident once it has been initiated.

Another common division in human errors is the division based on the types of errors [53]:

- Error of Commission (EOC) — performing the wrong action. A human failure event resulting from an overt, unsafe action that, when taken, leads to a change in plant configuration with the consequence of a degraded plant state. Examples include stopping safety-injection pumps which are running, closing valves and blocking automatic initiation signals,
- Error of Omission (EOO) — not performing the correct action. A human failure event resulting from a failure to take a required action that leads to an unchanged or inappropriately changed plant configuration with the consequence of a degraded plant state. Examples include failures to initiate standby liquid control system, start auxiliary feedwater equipment and failure to isolate a faulted steam generator.

The main focus in PSA is on EOO's, while EOC's are generally considered out of the scope due to the difficulties to systematically and comprehensively identify EOC's. EOO's are typically modelled in PSAs because they are easily defined and limited by the requirements of the emergency operating procedures. The U.S.NRC HRA Good Practice document [24] recommends that EOCs should be addressed in future PSAs and as a minimum a search should be performed for conditions that make EOCs more likely.

Table 10 introduces the human action categories considered in PSA. As a minimum, types A, B, C1 and C3 should be analysed from EOO point of view. Type C2 specifically addresses the EOC aspect, but generally EOC may be relevant even for other types.

Table 10 Types of human actions considered in PSA.

TYPE	DESCRIPTION	IMPACT ON PSA	L2PSA ASPECTS
A	Human actions before the initiating event during normal operation that degrade system availability	Miscalibrations, misalignments explicitly modelled in the PSA (system fault trees)	L2PSA may include some systems not considered in L1PSA
B	Human actions that contribute to initiating events	Not explicitly modelled in the PSA for full power mode (except when using fault trees to model initiating events). Treated at IE data level. Explicitly considered for low power and shutdown PSA	Not relevant in L2PSA
C1	Human actions during the accident following the correct procedures	Human failure event (HFE) explicitly modelled in the PSA (event trees and fault trees)	Main task in HRA for L2PSA. Includes analysis of actions made by operators and TSO using EOPs and SAMG.
C2	Human actions during the accident that, due to the inadequate recognition of the situation or the selection of the wrong strategy, make it worse	Identified EOC explicitly modelled in the PSA (event trees and fault trees)	Critical to identify erroneous actions that may lead to the containment failure, e.g. due to wrong timing of the action
C3	Human actions during the accident, trying to recover the situation; for example repairs of equipment	Recovery actions explicitly modelled in the PSA (normally treated at sequence level)	As in L1PSA, important to be consistent to what extent and under which conditions recovery actions are accounted for.

3.3.2 Post-initiator actions to be considered

Identification of human actions is based on comprehensive co-operation between event sequence and systems analysts. The following list can be used as a starting point for potential operator actions to be included in the L2PSA [47]:

- Operator actions specified in the EOPs, but not credited in L1PSA as it is considered ineffective in preventing core damage,
- Operator actions specified in the EOPs that are assumed failed in the L1PSA, but are recoverable,
- Operator actions specified in the SAMG.

Actions can also be divided into those performed by:

- Operators:
  - Diagnosis by control room staff,
  - Action outside the control room (repair, operating dedicated severe accident equipment...).
- Technical Support Organisation (TSO) actions (using EOP and/or SAMG information, short term or long term actions).

Of these operator actions only those actions which can be effective in preventing containment failure need to be modelled in L2PSA. Table 11 lists operator actions generally considered in L2PSAs for different reactor types.

Table 11 Operator actions considered in L2PSAs for different reactor types.

Reactor type	Severe accident management actions
Gen II PWR	• Depressurisation of the primary circuit • Core reflood (may include recovery of AC power) • Containment spraying • Containment venting • Containment water filling for reactor cavity flooding • Hydrogen risk management • Manual closure of containment isolation valves • Isolation or feeding of an affected SG
Gen II BWR	• Depressurisation of the primary circuit • Core reflood (may include recovery of AC power) • Boration after to prevent recriticality • Containment spraying • Containment venting • Lower drywell flooding • Containment water filling – cooling of RPV from outside • Manual closure of containment isolation valves
Gen III PWR	Similar to Gen II PWR

3.3.3 Issues that must be carefully treated in HRA for L2PSA

There are a number of issues that must be addressed in HRA for L2PSA:

- The dependency between L1PSA and L2PSA. Two dependency cases may be distinguished:
  - Operator actions common to L1PSA and L2PSA, e.g. recovery of core cooling. The difference is that in L2PSA the time window can be longer, there may be more symptoms to help diagnosis, and SAMG as well as crisis organisation may support the crew to perform the action. HRA in L1PSA and L2PSA should be performed in an integrated manner to consistently assess the conditional HEP for the Level 2 action,
  - Operator actions relevant only in Level 2 but that are dependent on the plant damage state (PDS). The key issue is to check whether the PDS provides enough information for accurate assessment of the context for the task or PSFs. From HRA point of view, PDS definition may need to be refined to account for issues such as associated event sequences (failed safety functions), initiating events, equipment failures and previous operator errors. Dominant PDS-specific MCSs need to be analysed to assess the preconditions for Level 2 operator actions.
- The interaction between the operators' team and the crisis organisation, including the availability of communication,
- Procedures used in Level 1 actions vs. Level 2 actions. The operator actions modelled in L1PSA (before core damage), tend to follow EOPs, which are quite specific in terms of procedure steps and if-then logic statements. The operator actions modelled in L2PSA (after core damage) use SAMGs as a basis. SAMGs are usually more general in nature than EOPs. During the severe accident the decisions are based on the SAMGs and the training provided to all relevant staff for use of these guidelines and procedures. It should be considered that these procedures are more complex than during the normal operation or design basis accident and in general the symptoms are not exact. A gap may exist between EOP and SAMG and there may be potential operator reluctance to perform certain actions,

- The link between human action and modification of the accident progression. The analysis of human actions that can modify the progression of an accident and L2PSA needs to include some kind of dynamic iteration to recalculate the available delay for next actions,
- The assessment of the impact of the action for the case where actions have both negative and positive influence (see SARNET benchmark on hydrogen, for example),
- Treatment of available time vs. failure probability. Many HRA methods use a time correlation curve to assign a base failure probability for a human action. This is better suited for operator actions with relatively short time windows for success, i.e. Level 1 human errors. The use of time correlation curves at Level 2 should be examined and considered carefully as the accident progression can have much longer time scales. One solution is to define a minimum human error probability for long time windows of several hours or even days,
- Treatment of uncertainties. All the above issues include uncertainties that need to be addressed, preferably in same manner as other uncertainties in L2PSA. Reliability parameter uncertainties are handled in the parameter uncertainty analysis. Some model uncertainties may be explicitly modelled, some may need to be analysed in sensitivity studies. The remaining uncertainties should be documented as part of the qualitative uncertainty analysis.

Compared to HRA in L1PSA, the time windows for different steps are typically long, the means to detect and diagnose are dependent on accident management instrumentation and associated information displays, the procedures are given in SAMGs, and more personnel can be involved in the decision making process, e.g., the technical support centre (TSC) or crisis management organisation. These aspects also mean that possibilities to validate data using full-scope simulator are limited and the human error probability assessment has to rely on expert judgments.

In the L2PSA, the human actions are defined by the SAMGs, while in the L1PSA the operator actions are defined by EOPs. The main difference between EOP and SAMG is that the SAMG consists of guidelines, not procedures. The principle distinction is that EOP should be applied step by step while SAMG can offer several options that should be choosen in relation with the crisis organisation. Each step in the guidelines is part of the overall process that should be used to reach a decision regarding the appropriate actions to take during a severe accident.

Example: WOG SAMG

In this section the WOG SAMG (Westinghouse Owners Group) guide is described in more detail. The generic WOG SAMG contains two main sets of guidelines: one for the control room and one for the TSC. For the control room, there are two guidelines (SACRG-1 and SACRG-2). The first one is designed to handle a fast-evolving severe accident: it is applicable as long as the TSC is not operational. The second one is applied as soon as TSC is fully operational. It facilitates communication with TSC and transfer of information.

Most of the guidance is addressed to the TSC: it contains diagnostic tools, guidelines and computational aids.

The TSC diagnoses the plant status and challenges to fission product barriers as indicated by the diagnostic flow chart (DFC) and the severe challenge status tree (SCST) key parameters and their setpoints.

If a SCST setpoint is exceeded, it corresponds to serious conditions related with a severe challenge guideline (SCG) and the severe accident management strategy must be implemented.

If a DFC setpoint is exceeded, the TSC goes to the appropriate Severe Accident Guideline (SAG) and evaluate the need to implement the related SAM strategy.

A limited number of key parameters are used for diagnosis of the status of the plant and application of the four SCG and eight SAG. Computational aids are available to help the TSC with decision-making in the guidelines.

Two guidelines deal with long term monitoring of implemented strategies and with SAMG exit.

Four points have to be considered in every SAG:

- The possibility to implement the strategy based on the plant status,
- Potential positive and negative impacts of the strategy and decision for implementation,
- Success of the implementation,
- Long term concerns of the implemented strategy.

For SCG, the second point in relation with negative impacts is not applicable.

Human factors were considered during the WOG SAMG development process. They have influenced the following different aspects of the WOG SAMG:

- Decision-making process,
- Severe accident management diagnostics,
- Severe accident management guidance content,
- Severe accident management guideline steps,
- Development of computational aids,
- Control room / TSC communication,
- Identification of expectations.

As indicated in the “Severe Accident Issue Closure Guidelines” and recalled in WOG SAMG scenario templates, the industry initiative on SAM “calls for self-evaluation of utility severe accident management capabilities, not just upon completion of the initial implementation of the plant-specific severe accident management program, but also on an ongoing, periodic basis. Table-top drills should be utilised to ensure that personnel in the utility’s emergency response organisation (ERO) are familiar with the use of the Severe Accident Management Guidance (SAMG).” Based on this approach, after the implementation of a plant-specific WOG SAMG, it is recommended that the SAMG is validated.

3.4 EXAMPLES OF APPLICATION FOR L2PSA

3.4.1 Use of HORAAM methodology in French PWR L2PSA (IRSN)

In the late nineties, the IRSN developed the “Human and Organizational Reliability Analysis in Accident Management” (HORAAM) model to take into account human actions in L2PSAs [57]. The model is based on the observation of the French crisis exercises. From this study, the following seven influence factors were selected. Experts were asked to discuss the relevance of the 7 influence factors (Table 12) and to quantify their weight.

Table 12 Influence factors in the HRA model used by IRSN

Influence factors	Description
Time available for decision	Time necessary to obtain, check, process information and make a decision about the required action.
Information and measurement means	This influence factor refers to the quality, reliability and efficiency of all measurements and information available in the control room and means of transmitting them to crisis teams.
Decision difficulty	Difficulty in taking the right decision (in terms of potential consequences of the action).
Difficulty for the operator	Difficulty of the action independently of work conditions: quality of the procedures (description of the actions), experience, training and knowledge in the control room or in the plant.
Difficulty induced by environmental conditions	This influence factor describes on-site conditions in which the actions decided upon have to be performed (local action or not, radioactivity, temperature, smoke, gas, exiguity,…)
Scenario difficulty	Difficulty of the global context of the current accident scenario in which a decision must be made.
Degree of involvement of the crisis organisation	Availability of the crisis organisation support.

The HORAAM model predicts human error probabilities and is based on a decision tree structure, the top events of which are the 7 influence factors. Easy to estimate, the 7 influence factors can only be assigned to 2 or 3 modes, characterised as “favourable”, “unfavourable” and possibly “medium”. Combination of the possible modes of the 7 seven influence factors along the branches of the decision tree results in Human Error Probabilities (HEPs) that vary from 10^{-4} to 1. A combination of several “unfavourable” modes rapidly leads to a failure probability of 1. Conversely, if all the influence factors take the “favourable” mode, the failure probability is 10^{-4}.

In the Accident Progression Event Tree (APET), the HORAAM decision event tree is an external model. Input data for the HORAAM model are the values of the 7 influence factors for a given L2PSA sequence. Each value is determined in the APET through a small quantification model (internal model) from APET variables regarding the given L2PSA sequence.

Human actions modelled in L2PSA are defined in SAMG. A few other actions, relevant for L2PSA and not modelled in L1PSA, are defined in EOPs and containment crisis organisation guide.

The SAMG contains 2 types of human actions:

- “Immediate actions”: actions that can be performed immediately by operators alone because the crisis organisation expertise is not required,
- “Delayed actions” that need the crisis organisation expertise.

The first HORAAM model did not consider that SAMG actions can be undertaken by operators alone and consequently there was no dependency between failures of L1PSA actions and L2PSA actions, since it was considered that the actions were performed by different crews.

For the L2PSA now in development, it is considered that the “Immediate actions” of the SAMG can be undertaken by operators alone. Subsequently, dependencies have been introduced between a given action required by the EOPs and the same action required by SAMG when it can be undertaken by the operators alone.

No uncertainties have been taken into account on the human failure probabilities, but sensitivity studies on L2PSA results have been performed for specific cases.

3.4.2 Use of THERP in the assessment of containment filtered venting failure (GRS

For German PWR the accident management includes filtered venting of the containment in the ex-vessel phase of a severe accident. In the framework of a L2PSA for a German Konvoi plant by GRS (completed in 2000) the influence of human factors on the unavailability of this action was taken into account. The assessment was based on the THERP (Technique for Human Error Rate Prediction) method [50]. The unavailability of filtered venting was estimated to be approximately 0.04 per demand, with approximately half of this unavailability due to human errors.

3.4.3 Scope of actions analysed by THERP in Spanish BWR (IBERDROLA)

For Spanish BWR the Severe Accident Guideline (SAG) contains all the human actions demanded during a severe accident. No other human actions were considered in the L2PSA. The SAGs are based on a similar methodology to the Emegency Operator Procedures (EOPs). The same human reliability methodology was used for both the L1 and L2PSA, based on the THERP method [50]. Typical human actions demanded in the severe accident are (HEPs mean values for Cofrentes NPP):

- Late RPV depressurization to avoid threats to the containment during vessel failure by DCH phenomenology (HEP=0,0014)
- Late RPV inyection to avoid RPV failure with core damage (HEP=0,2)
- Late boron inyection to avoid the re-criticality during the core degradation (HEP=0,0018)
- External power recovery for SBO before vessel failure (HEP=0,004) and before containment failure (0,002)
- Containment venting after hydrogen accumulation or containment pressurization (HEP=0,0013)
- Containment flooding before vessel failure to avoid containment erosion attack by MCCI phenomenology (No credit was done to the RPV external cooling) (HEP=0,2)
- Start of igniters (also demanded in EOPs) to limit the hydrogen accumulation in the containment (HEP=0,024)

3.4.4 Tractebel Engineering (Belgium)

In the framework of the update of L2PSA for Belgian NPP, a L2PSA HRA methodology has been developed. It is mainly based on the HRA methodology for the L1PSA of the Belgian units. The Technique for Human Error Rate Prediction (THERP) methodology and the Standardised Plant Analysis Risk – Human reliability analysis (SPAR-H) methodology complete the set of references used.

The L1PSA methodology for HRA is applied for the L2PSA as far as possible for consistency. The THERP methodology is used as a basis for the determination of the different factors of Human Error Probability (HEP) in L2PSA. SPAR-H methodology is used to complement the THERP methodology as it provides additional information. The THERP and SPAR-H methodologies are American methodologies (NUREG/CR) and their use is consistent with the American approach selected globally for the Belgian L2PSA.

The human errors that are considered in the L2PSA are related to the human errors that can occur during accident scenarios in which core damage has occurred.

The actions that are considered in the L2PSA can be found in the APET. The actions are selected according to their presence in the plant-specific WOG SAMG and their impact on the containment performance and on radionuclide releases; previous L2PSAs are also considered. The HRA methodology is applied to assess all of the selected actions.

The accident management strategies for prevention of core damage are dealt with in the EOPs. They are only applicable during the very early phase of the APET ("very shortly after core damage"). The Very Early Accident Management (VEAM) strategies found in the procedures FR-C.1 or equivalent and taken into account in the APET are the following:

a) RCS depressurisation,

b) RCS injection,

c) Injection into containment.

The transition from EOPs to SAMG is also considered in the HRA.

The accident management strategies to mitigate the consequences of a severe accident are dealt with in the WOG SAMG. They are applicable during the early and late phases of the APET if the transition from EOPs to SAMG is successfully performed. It is considered that no strategy will be applied at vessel failure as it is a very short timeframe. The Early and Late Accident Management (EAM and LAM) strategies found in the SAMG and taken into account in the APET are the following:

a) Reduction / Mitigation of radionuclide release,

b) Injection into SG,

c) RCS depressurisation,

d) RCS injection (and reactor cavity flooding after vessel failure),

e) Injection into containment (and reactor cavity flooding if there is a path between containment sumps and reactor cavity),

f) Containment depressurisation.

Each human failure event is a basic event that has to be quantified. The HRA methodology aims to establish rules for the quantification of those human failure events. In the methodology, a human failure event is defined as an event for which failure results from the failure to perform a task (in which diagnosis and actions are included).

To aid the quantification and to take into account the specificities of SAMG, each task is decomposed in successive subtasks. Seven subtasks were selected.

A human error probability is assigned to each subtask and the human error probability for the task is the sum of the human error probabilities of all the subtasks.

Each subtask can be linked to one of the following categories: action or diagnosis. The subtasks related to actions consist of operating equipment, performing line-ups, starting pumps or other activities performed whilst following plant procedures or work orders. The subtasks related to diagnosis consist of reliance on knowledge and experience to understand existing conditions, planning and prioritising activities and determining an appropriate course of actions.

Based on this explanation, five subtasks are related with diagnosis and two with actions.

For each subtask, the human error probability has to be assigned. The human error probability formula that has to be applied for each subtask is an adaptation of the formula used for the L1PSA that takes into account the possibility to recover the human error.

The formula consists of a base probability, non recovery probabilities by a member of staff responsible for the subtask and by another member of staff and the application of Performance Shaping Factors (PSFs).

PSFs are for the assessment of the human error probability of a subtask. The THERP methodology is the first methodology that has taken into account PSFs. The SPAR-H methodology for PSFs is taken as a basis for the implementation of the PSFs in this HRA methodology, as more detailed and complete information is provided. The eight PSFs of SPAR-H methodology are considered: available time, stress, experience and training, procedures' ergonomics, human-machine interface, complexity, fitness for duty and work processes. The PSF adjustment factor of SPAR-H has to be used when three or more PSFs are assigned negative ratings.

Dependency is also applied between the different tasks according to SPAR-H methodology.

Due to the fact that the methodologies used as references (THERP and SPAR-H) are not developed specifically for L2PSA HRA and its related specific SAM guidance and decision-making process, it is considered that there are uncertainties in the assignment of HEP and particularly the assignment of the different levels for PSFs. Consequently, expert judgment should be applied to quantify each HFE to enhance the confidence for HEP.

Sensitivity analyses are performed under the same conditions as for the expert judgment methodology (see Tome 1 Appendix on expert judgment for the use of expert judgment in Belgium).

There is no publication concerning the Belgian L2PSA methodology to date.

3.5 SUMMARY OF SELECTED METHODS

3.5.1 THERP/ASEP

THERP is the best known, most frequently applied technique for human performance reliability prediction [50]. It is a methodology for predicting human error rates and for evaluating the degradation of a human-machine system likely to be caused by human errors in association with factors such as equipment reliability, procedures etc. THERP uses performance-shaping factors to make judgments about particular situations. The THERP methodology and data are also used by many other HRA methods as a benchmark.

THERP models human error probabilities (HEPs) using an event tree approach, in a similar way to an engineering risk assessment, but also accounts for PSFs that may influence these probabilities. The probabilities for the human reliability analysis event tree, which is the primary tool for assessment, are nominally calculated from the database, but local data e.g. from simulators or accident reports may be used instead. The resultant tree portrays a step-by-step account of the stages involved in a task, in a logical order.

The technique is known as a total methodology as it simultaneously manages a number of different activities including task analysis, error identification and representation in form of HEP quantification.

ASEP (Accident Sequence Evaluation Program) is an abbreviated and slightly modified version of THERP [49]. ASEP comprises pre-accident screening with nominal human reliability analysis, and post-accident screening and nominal human reliability analysis facilities. ASEP provides a shorter route to human reliability analysis than THERP by requiring less training to use the tool, less expertise for screening estimates and less time to complete the analysis. ASEP tends to be used in the screening phase to identify those tasks that require a more detailed analysis using THERP.

3.5.2 ATHEANA

A Technique of Human Error Analysis (ATHEANA) is a second generation technique developed by the US Nuclear Regulatory Commission (USNRC). ATHEANA is a methodology for identifying plausible situations with a high likelihood of error and potential error-forcing contexts that may result in human failure to correctly perform an action, and for estimating human error probabilities (HEPs) for the human events modelled in probabilistic risk assessments.

ATHEANA covers the following steps:

- Identify human actions to be assessed,
- Define Human Failure Events (HFEs) pertinent to performing these human actions incorrectly,
- Determine the HEPs for the defined HFEs, including consideration of likely recovery actions,
- In addition, ATHEANA includes the following formalised, structured and documented processes, which comprise key distinctive features of the methodology:
    - Identify operational vulnerabilities that could set up potential unsafe actions (UAs) (e.g., procedure weaknesses and operator knowledge limitations and biases),
    - Identify plausible deviations from nominal conditions or plant evolutions that might cause problems or misunderstandings,
    - Identify important PSFs relevant to both nominal and deviation conditions.
    - Identify other aleatory factors that could significantly affect the likelihood of the HFEs and their uncertainties (i.e. investigating a broad range of potential influences).

These features especially relate to searching for error forcing contexts (EFCs) for the related HFEs and determining the corresponding HEPs.

The main basic steps of the ATHEANA methodology can be seen in [51] . Many of the steps in ATHEANA are typical good practices and so are not unique and do not really represent additional steps in performing an human reliability analysis. However, these good practices are formalised as specific steps in the ATHEANA methodology.

Additionally, ATHEANA's steps (involving the definition of an HFE and the subsequent determination of the factors likely to have the greatest influence on the probability of operators making the human failure of interest, within the context associated with a particular accident sequence) mirror what is typically done in any HRA. However, in doing so, ATHEANA examines the following considerations:

- Should an HFE be represented by one or more particular Unsafe Actions (UAs)?
- Should certain errors of commission (EOCs) also be addressed?

- Should additional aleatory influences, including different plant conditions and other contextual deviations, be considered for the PSA sequence of interest?

Finally, derivation of the corresponding HEP for an analysed UA or HFE (i.e., quantification) is performed on the basis of identified important influences on human performance, just as in any HRA quantification technique. The ATHEANA methodology currently uses a formalised expert opinion elicitation process to estimate the HEP, rather than using specific rule sets or similar structures to convert the effects of these important influences into an HEP.

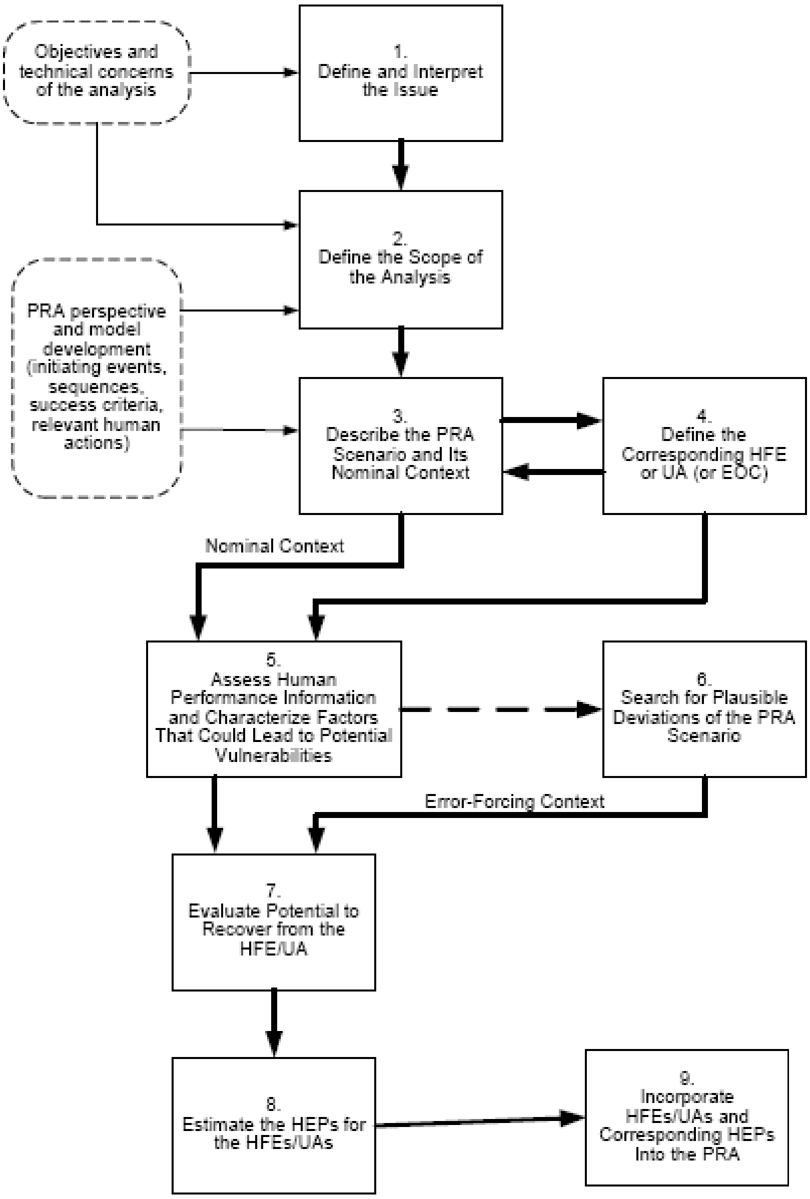

Fig. 14 Steps in the ATHEANA methodology

3.5.3 CREAM

CREAM (Cognitive Reliability Error Analysis Method) is a second generation methodology for human reliability analysis [44]. There are two versions of the technique, the basic and the extended version, which share two primary features; ability to identify the importance of human performance in a given context and a helpful cognitive model and associated framework, usable for both prospective and retrospective analysis. Prospective analysis allows likely human errors to be identified while retrospective analysis quantifies errors that have already occurred.

The concept of cognition is included in the model through use of four basic 'control modes' which identify differing levels of control that an operator has in a given context and the characteristics which highlight the occurrence of distinct conditions. The control modes which may occur are as follows:

- Scrambled control: the choice of the forthcoming action is unpredictable or haphazard. The situation in question may be displaying rapid alterations in unexpected ways thus eliminating the operator's ability or opportunity to make deductions about the next action required,
- Opportunistic control: the next action is determined by superficial characteristics of the situation, possibly through habit or similarity matching. The situation is characterised by lack of planning and this may be due to the lack of available time,
- Tactical control: performance typically follows planned procedures while some ad hoc deviations are still possible,
- Strategic control: plentiful time is available to consider actions to be taken in the light of wider objectives to be fulfilled and within the given context.

The particular control mode determines the level of reliability that can be expected in a particular setting and this is in turn determined by the collective characteristics of the relevant Common Performance Conditions (CPCs).

3.5.4 MERMOS

MERMOS [39] is the EDF reference method in use for HRA. This methodology was developed in the late nineties to take into account on the one hand the state-oriented procedures and on the other hand the computerised procedures of the N4 series. More recently, MERMOS was extended to other EDF series. From now on, this second generation HRA methodology has replaced the EDF FH6 first generation HRA methodology in the PSA models of all EDF units for assessing operator tasks in accidental situations.

One of MERMOS' main characteristics is that it doesn't only focus on operators but also takes into account the entire operating system including: the operators, the EOPs and the man-machine interface. The methodology consists of identifying the failure scenarios of the operating system and categorising as one of three types: strategy, action, diagnosis (SAD).

The scenarios of failure are characterised by two concepts:

- CICAs : important configurations of accident operation (describing a way of operating); a failure occurs when a CICA is not adapted to a specific situation and when the operating system does not reconfigure itself on time,
- Situation features: they explain the occurrence and the longevity of a CICA.

The analysis consists of identifying and then quantifying all the failure scenarios that can lead to the failure of a PSA human failure mission. The system is modelled in a functional way through the dysfunctioning of any of the SAD functions. A failure scenario is detailed enough to relate to real operation situations, but not too detailed so that its constituent elements are sufficiently likely to be quantifiable. A residual probability is used to cover the scenarios not identified by the analyst.

Quantifications are mainly made using expert judgment. They are determined according to discrete values [46] to increase the understanding of the analysis and to ensure a certain robustness and reproducibility of the results.

3.5.5 SLIM-MAUD

The Success Likelihood Index Methodology (SLIM) [58], [59] is based on expert judgment. It is a systematic HRA method for positioning the likelihood of success of a task on a scale as a function of differing conditions influencing the successful completion of tasks. This method can be implemented manually and through the use of an interactive computer program called Multi-Attribute Utility Decomposition (MAUD).

The main SLIM HRA methodology considers a relatively small set of PSFs. They include human traits such as experience and training and conditions of work. The quantification approach is not time-based per-se, but rather assesses the impact of time available on completion of a task through consideration of a time-based PSF.

Firstly, the analysts weight PSFs considering the task to be analysed and then they rate them considering the actual performance of the plant. Once done, weights and ratings are multiplied together for each PSF and summed across PSFs to arrive at the Success Likelihood Index (SLI). The SLI represents the consensus belief regarding the negative and the positive effects of the PSFs on the likelihood of success for the task considered. Then the SLI is converted to a probability considering the formula:

- log (probability of success) = a (SLI) + b

The coefficients “a” (the slope of the line) and “b” (the intercept of the line with the vertical axis) are determined considering SLI for two tasks in which the probability of success is known.

A variant of SLIM/MAUD is the Failure Likelihood Index Methodology (FLIM) methodology. In the FLIM approach, “Success” Likelihood Indexes (SLIs) are replaced by “Failure” Likelihood Indexes (FLIs). The main steps are identical in both methodologies: select experts that are familiar with the tasks to be quantified; ask them to calculate the SLIs or the FLIs, to calibrate the relative likelihood scale and to convert the index values to human error probabilities.

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4 QUANTIFICATION OF PHYSICAL PHENOMENA AND CONTAINMENT LOADING

This chapter presents most of the physical phenomena that influence the severe accident progression and containment loading. The reader should find some phenomenological description of each related phenomena and some recommendations on how it can be modelled in a L2PSA event tree.

Since many variations are possible in the precise way of developing and using L2PSA, the reader should be aware that the split fraction (or uncertainty distribution) values resulting from the quantification of the different severe accident phenomena presented in this chapter are not provided. As a result, the guidelines user should be motivated to calculate his split fraction following the practical recommendations and examples provided in what follows.

4.1 IMPORTANT STRATEGIES FOR DIFFERENT PURPOSES / END-USERS NEEDS

The ASAMPSA2 End-User survey [60] identified 6 areas of L2PSA applications to be prioritised in the development of this guidance document:

1. To gain insights into the progression of severe accidents and containment performance,
2. To identify plant specific challenges and vulnerabilities of the containment to severe accidents,
3. To provide an input to determining whether quantitative safety criteria which typically relate to large release frequencies (LRF) and large early release frequencies (LERF) are met,
4. To identify major containment failure modes and their frequencies, including bypass sequences; and to estimate the corresponding frequency and magnitude of radionuclide releases,
5. To provide an input to the development of plant specific accident management guidance and strategies,
6. To provide an input to plant specific risk reduction options, especially in view of issues such as ageing, plant upgrades, lifetime extension, decision making in improvements, maintenance, and cost benefit analyses.

Depending on the final L2PSA application, some differences may be justified in the method used to perform severe accident analysis. The following paragraphs try to provide some explanations on the 6 areas.

To gain insights into the progression of severe accidents and containment performance

Gaining insights into the progression of severe accidents and containment performance requires that the level of detail of accident phenomena modelling is sufficient in the APET and includes some uncertainties analysis. Important contributions of severe accident phenomena should be traced back from the release categories through the model, including information related to amplitude of release or kinetics.

It is important that the team performing the analysis is aware of conservatisms (non-conservatisms) and uncertainties in the deterministic codes to be able to define and understand the results of sensitivity and uncertainty cases.

It is also important for the team to be aware of the codes' limitations with respect to calculation of certain issues (e.g. time of combustion ignition or steam explosion, mass of corium relocated in-vessel) and to take this into consideration during the probabilistic assessment.

For such an application, the phenomena analysis should be as realistic as possible and include some sensitivity/uncertainties analysis. The dependencies between the events should also be modelled appropriately.

The L2PSA should provide some global views on the different possible paths of severe accident development for each PDS.

<u>To identify plant specific challenges and vulnerabilities of the containment to severe accidents</u>

If the objective of the L2PSA is only to identify plant specific challenges and vulnerabilities of the containment to severe accidents, the severe accident phenomena analysis can be restricted to the specific events that may threaten the containment and less attention can be paid to the dependencies between the events. Conservatisms may be introduced when looking individually at the specific plant vulnerabilities.

In that case, the L2PSA final results may not be realistic and the relative importance of identified risks can be biased. When presenting the final results, the L2PSA analyst should check that no risk is masked by any conservatism, especially if the release categories are based on a chronological definition (see Volume 1 chapters 5 and 6). In that case, the frequency of each containment failure mode should be calculated individually.

Such a L2PSA can provide useful preliminary views and help to identify the issues where further efforts are needed.

<u>To provide an input to determining whether quantitative safety criteria which typically relate to large release frequencies (LRF) and large early release frequencies (LERF) are met</u>

To demonstrate that some quantitative safety criteria (LERF, LER) are met, the required level of detail in the accident phenomena analysis depends on:

- The plant design,
- The definition of "large release".

If the plant design is extremely robust regarding both core damage prevention and severe accident consequences, the demonstration that some global LRF or LERF safety criteria is met can be done with some simplified (conservative) study of the accident phenomena. Nevertheless, even when the plant design is robust, if the regulator imposes a low level for the « large » release definition then some more detailed analysis of phenomena may be needed. This is true also for the source term assessment. An assessment of all coupling between the severe accident phenomena and the radioactive release should be performed (resuspension of fission products for example).

The source term assessment may not be needed for this purpose if the large release definition is only based on the containment failure modes (assuming that the failure of any component that would increase the leak rate of the reactor containment building would lead to "large" release).

<u>To provide an input to the development of plant specific accident management guidance and strategies</u>

For a plant without any specific severe accident management guidance or dedicated system, a L2PSA can be developed to obtain a first ranking of the risk of release after core damage. The results can then be used to support the definition of a first severe accident management strategy, and to demonstrate that the dominant risks of large release are effectively reduced by application of the strategy.

A preliminary severe accident phenomena analysis can be restricted to the specific events that may threaten the containment and less attention can be paid to the dependencies between the events. Some conservatism may be introduced when looking individually into some specific plant vulnerabilities.

When some specific severe accident guidance and measures have already been developed for a plant, the L2PSA model should take into account all relevant systems and human actions, including the possibility of failures. In that case, the L2PSA should correctly model the advantages and drawbacks (positive and negative impacts) of all actions performed during the severe accident progression and its conclusion should contribute to the optimisation of the severe accident management strategy (minimisation of the risks whatever the accident). The Human Reliability Analysis has to be precise enough to capture the situations with an unfavourable context for the accident management.

With that perspective of SAMG optimisation, it is recommended to progressively improve the degree of realism of the severe accident phenomena analysis for the L2PSA: a natural progression should be to develop firstly some simplified/conservative severe accident analysis allowing the implementation of a first SAMG strategy and then to complement the severe accident analysis for the optimisation of the SAMG. The uncertainty analysis should be performed during that second step.

<u>To provide an input to plant specific risk reduction options, especially in view of issues such as ageing, plant upgrades, lifetime extension, decision making in improvements, maintenance, and cost benefit analyses</u>

Depending on the specific issue (or “risk reduction option”), L1PSA or L2PSA can be more or less crucial for the application.

It is important that the L1 and L2PSA scope and level of detail cover the risk reduction options being addressed. It should be verified that some specific L1PSA or L2PSA assumptions do not mask, decrease or increase the benefit of a plant modification. In the case where the PSA is limited, the benefit of modification should also be estimated for the events outside the scope of the PSA. This is especially true for the modifications that concern the containment if they are beneficial for the events outside the scope of the PSA.

Regarding severe accident phenomena analysis, it may be needed to precisely identify the level of conservatism of some important contributor to the risk option reduction. Three cases can appear when discussing one specific case:

- The L2PSA accident phenomena analysis is obviously too conservative and the study has to be revised before any decision,
- The L2PSA accident phenomena analysis is realistic and uncertainties are low : it is typically a case where L2PSA should be useful,
- The accident phenomena analysis identifies huge uncertainties: The decision should be to continue some research activities (to decrease uncertainties) or to modify the plant design or operating procedures to avoid the considered situation at all.

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4.2 DEFINITION AND CALCULATION OF REPRESENTATIVE SEQUENCES FOR EACH PDS

Most L2PSAs are built using a set of “representative sequences” that are calculated by integral codes like MELCOR, ASTEC or MAAP. These “representative sequences” may be calculated from the initiator of the accident to different accident progression steps (vessel rupture, basemat penetration, reactor stable state, end of release …) depending on the L2PSA methodology used for the CET/APET.

The definition and the calculation of these “representative sequences” is clearly a key issue for L2PSA. If the “representative sequences” are not correctly defined, the PSA may have a weak physical sense.

This issue is highly connected with the L1-L2 interface (see: chapter 2).

4.2.1 The role of the “representative sequences”

In the L1PSA a huge number of sequences are found to lead to “core damage” end states. It would be impossible to make deterministic severe accident analysis for all these sequences and accordingly core damage sequences identified by L1PSA are grouped in Plant Damage States (PDS).

PDS are used for collecting those L1PSA event sequences leading to “core damage” end state which cause similar progression of events like extension of core damage, activity released from fuel, functioning of containment systems and containment behaviour. The PDS concept is based on the observation that many accident sequences are very similar from the point of view of accident progression in the containment, allowing the minimisation of the number of accident scenarios that must be effectively quantified by the L2PSA.

A L2PSA needs specific information about the physical progression of the accident. For example, the estimation of the delay between accident initiation and the beginning of the core degradation is important to assess the human reliability for performing SAMG or simply to assess the delay before containment failure and radioactive release. Moreover, most of the physical phenomena are linked together and should be quantified taking into account some realistic conditions. For example, the ejection of corium in the containment (case of DCH) depends on the vessel pressure which is obtained from the calculation of a “representative sequence”.

That is why a set of “representative sequences”, covering all the situations quantified in the L2PSA, has to be defined and calculated.

4.2.2 The selection of representative sequences

A PDS defines characteristics of the physical status of the reactor at a defined time, generally at the beginning of the core melt. Each PDS is an entry point to a Containment Event Tree (CET) or Accident Progression Event Tree (APET). Then, the probabilities of the questions in APET/CET are calculated with the help of each PDS attribute. In all cases, a method of selection is necessary to define a reasonable number of deterministic calculations. One way is to define one (or few) "representative sequence(s)" for each PDS.

The definition of the "representative sequences" must take into account:

- The sequence definition derived from the L1PSA,
- The specific events from the L2PSA which can change the physical evolution of the plant (mostly physical/chemical, but also human or automatic actions if any).

Some L2PSAs do not include the modelling of SAMG and human reliability. That allows a high simplification in the definition of the "representative sequences" because success and failure of human actions may increase the number of situations effectively treated in the L2PSA. However, it may be considered that such L2PSA studies are no longer state of the art.

4.2.2.1 Selection of representative sequences derived from L1PSA

This chapter discusses the definition of the "representative sequences" from the accident initiation to the beginning of core degradation.

The methodology used for the definition of the "representative sequences" derived from the level 1 study is highly linked with the methodology used for the definition of the Plant Damage States.

If the PDS are built with the objective to gather sequences with equivalent physical evolution of the plant up to the beginning of the core degradation, then the definition of representative sequences may be easy. In this case, one "representative sequence" can be defined and calculated for each PDS. In some cases, the same "representative sequence" may be valid for several PDS (if the PDS differ only on points impacting the severe accident phase, e.g. auxiliary building ventilation), allowing the reduction of the number of calculated sequences.

If the physical evolution of the plant (from initiating event to core degradation) differs for each L1 sequence within a PDS, the "representative sequences" selection should start with the PDS minimal cut sets with the highest frequency. It means that detailed L1PSA results are necessary for the selection of the "representative sequences" of a PDS.

It is assumed here that the L1 and L2PSA are consistent in the sense that all L1PSA sequences lead to core damage.

4.2.2.2 Selection of representative sequences taking into account the severe accident management

The sequences defined from the L1PSA analysis are defined up to the beginning of the core degradation. The severe accident progression must then be defined taking into account the human and system actions that influence the physical evolution of the accident, especially if Severe Accident Management Guidelines (SAMG) are to be credited in the PSA.

For example, three situations have to be covered by the selection of "representative sequences":

- Situations with availability of some safety systems, but without operator actions (e.g. the emergency organisation is not yet activated but required for the application of the SAMG application),
- Situations with availability of some safety systems with operators actions following the SAMG,

- Situations with availability of the safety systems with operators actions following the SAMG but with some human errors.

If the "representative sequences" are only derived from the Level 1 study, they should correspond to the first situation. The L2PSA APET/CET generates the other situations, which are not foreseen by the L1PSA definition of the sequences. These situations have to be covered by the "representative sequences" selection.

Of course, an independent treatment of each SAMG action (success/failure) may increase drastically the number of situations to be calculated for the L2PSA, and some simplifications have to be proposed (for example, application of all or none SAMG action). These simplifications can be justified by the situations' frequencies.

4.2.3 Calculation of the representative sequences

Generally, an integral severe accident code like ASTEC, MAAP or MELCOR is used for the sequences calculations. The code starts the calculation from the initiating event and takes into account the different assumptions leading to the core melt and containment loading. These codes can calculate the sequence until the reactor stable state, but the calculation can be stopped before depending on the methodology used for the APET.

It may be useful to first calculate the sequences without SAMG allowing identification of possible SAMG actions.

4.2.4 Use of the calculated representative sequences in the L2PSA – Situations which are not calculated

As explained above, all situations generated by the APET/CET cannot be perfectly covered by the set of calculated representative sequences. For the non-calculated situations, the L2PSA developer has to select the "more representative" calculated sequence.

It should be recommended to adopt some traceable approach for this. It can be done by using a selection tree, the development of which follows precise rules:

- Each situation may be characterised by a set of important variables,
- For each set of these important variables, the more representative calculation is selected.

Some particular sequences may be more difficult to calculate with integral severe accident codes:

- The ATWS (anticipated transient without scram) sequences, or, generally, situations with a critical core at the end of L1PSA,
- The homogeneous or heterogeneous dilution sequences,
- Thermal shock (reactor vessel breach as initial event),
- Some of the shutdown situations (e.g. open RCS reactor state and drop of a heavy load).

If assimilations to existing calculated sequences are proposed, they should appear as a limitation of the L2PSA.

4.2.5 Validation of the selection of representative sequences

The validation of the selection is necessary, and should lead to:

- The control of the quality of PDS grouping,

- The control of the deviation between the representative sequence and the other sequences in a PDS group.

The validation of the “representative sequences” selection can be performed by a few sequences calculations. The selection of the sequences can be done according to:

- The frequency of the sequences,
- Some engineering judgments.

According to the frequency, the most probable sequences (2 or 3) should be examined, calculated if possible and compared.

4.2.6 VEIKI example of selection

Different methods can be used for the “representative sequences” selection. The selection can be done according to probabilistic or deterministic analysis of the sequences.

First the representative sequence until the core damage is selected using only the L1PSA results:

- Selection from the sequences according to the frequencies of the minimal cut sets; the representative sequence belongs to the minimal cut sets of the largest frequency (in the case of a large number of PDS),
- Selection of the sequences according to their frequencies. This method is the grouping the minimal cut sets into sequences according to their behaviour. First the group of the highest frequency of sequence is selected then the dominant (representative) sequence is chosen from highest frequency group (in case of only a few PDSs). In this case more representative sequences can be selected for a PDS.

The deterministic methods need pre-calculations and according to the results of these calculations, the representative sequence can be chosen. This type of selection is not systematic and the basis of it is the engineering judgment on the assumption that all sequences in a PDS are similar. It can be used only for phenomena calculations:

- Sequence with the highest impact (e.g. produced hydrogen),
- Fastest load,
- Largest source term.

In Table 13, an example can be seen for the selection of a representative sequence according to the probabilistic approach. For the selection, the L1PSA results gives the basis. The L1PSA study has identified the accident sequences (minimal cut sets), and their frequency, leading to core damage. Together with the containment systems analysis, these sequences have been assigned into plant damage states (PDS). The PDS gathers accident sequences leading to core damage and having similar containment response characteristics.

The PDS is determined by the initial events, the availability of ECCS, the primary pressure and the containment events (bypass, spray, failure, etc). In Table 13, the minimal cut sets are given for a PDS and also the description of the “representative sequence” of the minimal cut set. First, the minimal cut sets were graded and the main minimal cut sets are selected from more than 300 minimal-cut sets. The selected minimal cut sets cover more than 98% of the sequences in the PDS. For each minimal cutset, the sequence description is also known. The sequence frequency is calculated taking into account the first 20 minimal cut sets. The “representative sequence” is the most frequent sequence; small LOCA with failure of ECCS operation (primary break size 10-20 mm, common cause failure of bus-bars: no ECCS and secondary feedwater and auxiliary feedwater pumps).

The selected representative sequences (set of minimal cut sets) are calculated. First a calculation is performed without any operator intervention and without any additional failure. Then the repair time and available time for effective recovery actions are determined. If the restoration of the ECCS and spray systems is possible then these representative sequences are also calculated.

Table 13 VEIKI example: minimal cut sets and sequences PDS

Risk Spectrum Analysis Tools - MCS, Version 2.00.00

Projekt: PAKS LEVEL 1-2 PSA FOR FULL POWER, UNIT 1

Frequency of core damage = 1.56E-06 [1/year]

						Sequences(group of minii		
No	**Frequency**	**%**	**sum %**	**minimal cutsets**	**Sequence description**	**Frequency**	**%**	**sum %**
1	4.00E-07	25.7	25.7	D2 10E$-ST-ALL	Small LOCA not Initiating ECCS Operation CCF of Bus-bars	1.20E-06	77.15	77.15
2	4.00E-07	25.7	51.41	D2 10B$-ST-ALL	Small LOCA not Initiating ECCS Operation CCF of Bus-bars			
3	4.00E-07	25.7	77.1	D2 10C$-ST-ALL	Small LOCA not Initiating ECCS Operation CCF of Bus-bars			
4	1.48E-07	9.51	86.6	D2 01V0D001-FRO--ALL	Small LOCA not Initiating ECCS Operation CCF of Service Water Pumps to Run	1.48E-07	9.49	86.64
5	6.81E-08	4.37	91.0	D2 01V0N001-PG--ALL	Small LOCA not Initiating ECCS Operation CCF of Filters Plugged	6.96E-08	4.46	91.11
6	6.14E-08	3.94	94.9	D2 10TL07D00$-FSS RLI_D2 -EO	Small LOCA not Initiating ECCS Operation CCF of FAN to Start No Break Isolation Before Boron Tanks Get Emptied	9.33E-08	5.98	97.09
7	2.08E-08	1.34	96.3	D2 10V$48S201-CCS--ALL RLI_D2 -EO	Small LOCA not Initiating ECCS Operation CCF of MOVs to Open No Break Isolation Before Boron Tanks Get Emptied	2.09E-08	1.34	98.43
8	2.00E-08	1.29	97.6	D2 10D$-ST-ALL RLI_D2 -EO	Small LOCA not Initiating ECCS Operation CCF of Bus-bars No Break Isolation Before Boron Tanks Get Emptied			
9	6.94E-09	0.45	98.0	D2 10TL07D00$-FRO--ALL RLI_D2 -EO	Small LOCA not Initiating ECCS Operation CCF of Fans to Run No Break Isolation Before Boron Tanks Get Emptied			
10	1.59E-09	0.1	98.1	D2 10D$-ST-ALL E-0:28_F_FN E-1:6_F_FN	Small LOCA not Initiating ECCS Operation CCF of Bus-bars A lépés előtti figyelmeztetést nem veszi figyelembe, A lépés előtti figyelmeztetést nem veszi figyelembe,			
11	1.18E-09	0.08	98.17	D2 10TL07D001-FSS 10TL07D002-FSS 10TL07D003-FSS RLI_D2 -EO	Small LOCA not Initiating ECCS Operation Fan Fails to Start Fan Fails to Start Fan Fails to Start No Break Isolation Before Boron Tanks Get Emptied			
12	5.56E-10	0.04	98.21	D2 1B-ST--ALL 10QD0$-FR-ALL	Small LOCA not Initiating ECCS Operation CCF of Bus-bars CCF of Diesel Generators to Run			
13	4.68E-10	0.03	98.24	D2 1B-ST--ALL 10QD0$-FS	Small LOCA not Initiating ECCS Operation CCF of Bus-bars CCF of Diesel Generators to Start			
14	4.30E-10	0.03	98.27	D2 10TL07D001-FSS 10TL07D003-FSS 10VX48S201-MOD-CC RLI_D2 -EO	Small LOCA not Initiating ECCS Operation Fan Fails to Start Fan Fails to Start Armature Module Fails to Carry out Opening No Break Isolation Before Boron Tanks Get Emptied			
15	4.30E-10	0.03	98.30	D2 10TL07D002-FSS 10TL07D003-FSS 10VY48S201-MOD-CC RLI_D2 -EO	Small LOCA not Initiating ECCS Operation Fan Fails to Start Fan Fails to Start Armature Module Fails to Carry out Opening No Break Isolation Before Boron Tanks Get Emptied			
16	4.30E-10	0.03	98.32	D2 10TL07D001-FSS 10TL07D002-FSS 10VW48S201-MOD-CC RLI_D2 -EO	Small LOCA not Initiating ECCS Operation Fan Fails to Start Fan Fails to Start Armature Module Fails to Carry out Opening No Break Isolation Before Boron Tanks Get Emptied			
17	4.09E-10	0.03	98.35	D2 10TL07D001-FSS 10TL07D002-MOD-FS 10TL07D003-FSS RLI_D2 -EO	Small LOCA not Initiating ECCS Operation Fan Fails to Start Motor Module Fails to Carry Out Start. Instruction Fan Fails to Start No Break Isolation Before Boron Tanks Get Emptied			
18	4.09E-10	0.03	98.38	D2 10TL07D001-FSS 10TL07D002-FSS 10TL07D003-MOD-FS RLI_D2 -EO	Small LOCA not Initiating ECCS Operation Fan Fails to Start Fan Fails to Start Motor Module Fails to Carry Out Start. Instruction No Break Isolation Before Boron Tanks Get Emptied			
19	4.09E-10	0.03	98.40	D2 10TL07D001-MOD-FS 10TL07D002-FSS 10TL07D003-FSS RLI_D2 -EO	Small LOCA not Initiating ECCS Operation Motor Module Fails to Carry Out Start. Instruction Fan Fails to Start Fan Fails to Start No Break Isolation Before Boron Tanks Get Emptied			
20	4.06E-10	0.03	98.43	D2 10TL07D00$-FSS	Small LOCA not Initiating ECCS Operation CCF of FAN to Start			

4.2.7 IRSN example of selection

In the PWR 900 MWe developed by IRSN, the L1-L2 interface provides transmission, for each L1PSA sequence, of all the information needed to characterise the accident progression (from core uncovery to atmospheric

releases). The PDS are based on a set of variables, called interface variables. Each accidental sequence of L1PSA is associated to a PDS through a set of values for these interface variables. The sequences characterised by the same set of interface values are grouped in a Plant Damage State (PDS).
22 interface variables concern initiator event, systems availability, containment state, residual power, activation of emergency plan (Table 14):

Table 14 IRSN example: L1-L2 interface variables (900 MWe PWR L2PSA)

PT – RCS break size	SF – Component cooling or essential service water systems
PL – RCS break localisation	AP – Water makeup to RCS availability
RT – SGTR number	BA – Safety injection water tank
VL – V-LOCA	SE – Secondary system break
AS – CHRS availability	SO – Pressurizer safety valve availability
BP – Low pressure safety injection availability	IE – Containment isolation
HP – High pressure safety injection availability	CR – Core criticality
GV – SG availability	PR – Residual power
LC – Electrical board availability (low voltage)	PU – Emergency plan
LH – Electrical board availability (high voltage)	RS – Electrical network availability
DL – Dilution	GR – Control rods blocking

Each of the 22 interface variables can take from 2 to 11 values. A set of values provides a precise description of systems states and L1PSA sequence. The IRSN L1-L2 interface leads to the definition of 224 PDS for power state, and 104 PDS for shutdown state. PDS are grouped in 13 accidental families.
The thermal-hydraulics “representative sequences” (up to beginning of core degradation) are defined from PDS following this method:

- Forfor each type of accident, only interface variables having an impact on the thermal-hydraulics progression of the severe accident are considered as grouping variables to define the representative TH sequences,;
- Aa thermal-hydraulics “representative sequence” is defined for each set of values of the grouping variables,;
- Thethe other variables are only used in the APET (for example, the status of containment isolation) or to define SAMG actions that can be undertaken.

For each representative sequence, two ASTEC calculations are performed:

- Oneone without any manual action after core uncovery;
- Oneone with all possible SAMG actions after core uncovery.

With this methodology, 66 Thermal-hydraulic transients and 108 core degradation calculations (78 without SAMG actions and 30 with SAMG actions) have been defined (Table 15).

Table 15 IRSN example: "representative sequences" definition

Accidental families	Number of PDS	TH transients	Core degradation calculations with ASTEC without SAMG actions	Core degradation calculations with ASTEC with SAMG actions
Large LOCA	15	7	7	2
Intermediate LOCA	31	12	12	2
Small LOCA	20	5	5	5
Very Small LOCA	14	6	6	2
SGTR	16	10	10	7
Secondary break	9	10	10	1
Loss of heat sink	15	9	9	6
Loss of SG feedwater	17	14	14	4
Station black out	16	5	5	1
V_LOCA	13	0	0	0
Partial black out	26	0	0	0
ATWS	22	0	0	0
Transient	10	0	0	0
Total	***224***	***78***	***78***	***30***

The 108 core degradation calculations do not cover all the situations met in the APET. For the non-calculated situations, a selection tree which is built manually selects the "more representative" calculated sequence.

4.3 IN-VESSEL CORE DEGRADATION PHASE

4.3.1 Core degradation

4.3.1.1 Description of accident phenomena

As per definition, L2PSA starts when core degradation and core melt is imminent. Meltdown ultimately occurs when the heat production rate exceeds the removal rate. This type of accident may be initiated by either overpower or undercooling conditions. Overpower most likely would follow a reactivity transient. Undercooling is related to reduce cooling through loss of coolant flow or loss of the coolant itself. The sequences herein described assume substantial or complete failure of engineered safety systems and should be recognised in the context of their very low frequency. [1, 2].

Blowdown and boiloff

If a break occurs in the primary system of a Light Water Reactor (LWR), the coolant leaves the reactor (blow-down) with a flow rate and phase (two-phase or single-phase) depending on the dimension and location of the break. As a consequence the core becomes uncovered, resulting in a rapid rise of the fuel rod temperature due to the decay heat generated in the fuel. For a large break, the flow rate leaving the core is very high and will be over within few minutes. In the case of small pipe break, the flow rate would be much lower. The blowdown would be followed by the more gradual boil-off phenomena. In PWRs, natural convection of steam to the steam generators and return of the condensate into the RPV may greatly influence this phase.

Heat-up phase

After the core is uncovered, heat transfer from the fuel rods to the steam is relatively low and the fuel temperature increases (heat-up phase). Natural circulation in the uncovered part of the core may transfer a significant fraction of the heat from the core to the upper plenum and to the primary system boundary. During this stage, radiation becomes the dominant heat transfer mechanism in the core. The fuel heating can induce the clad embrittlement and clad ballooning, which results in localised rupture and local blockage of the fuel assembly.

Zircaloy oxidation and hydrogen generation

As the temperature in the fuel increases above 1500 K, oxidation of the Zircaloy cladding by steam becomes an important heat source. This exothermic process involves the diffusion of oxygen in the growing oxide layer and release of hydrogen to the reactor vessel. The rate of oxidation is generally limited by the diffusion of oxygen through the oxide layer or by the quantity of steam available at the clad surface. Above ~ 1500 - 1700 K, the heat generation from the oxidation becomes so high that heat produced by the exothermic reaction exceeds the decay heat. In those regions where the steam flow is replaced by hydrogen, the Zircaloy oxidation phenomenon is obviously limited (steam starvation).

Stainless steel oxidation

The oxidation of stainless steel used in the core of LWRs is another potential source of hydrogen generation. Stainless steel - steam oxidation is a more complicated process than Zircaloy - steam oxidation because of the presence of several components in the steel (Ni, Cr and Fe). The rate of steel oxidation is small compared to the Zircaloy oxidation at the temperature below 1400 K But at higher temperature and close to the steel melting point, the rate of steel oxidation exceeds that of Zircaloy.

Quenching

If sufficient water is injected into the core, because of reflood from the bottom and/or top, the heat transfer from the core to the coolant greatly increases together with the steam generation. The high steam availability and shattering of oxide layers due to thermal shock can result in strong oxidation and subsequently in high hydrogen generation and increased core damage.

Chemical interaction among core materials

The composition of a LWR core is such that, during a severe accident, melting could occur in a variety of ways involving complex chemical reactions.

In a **PWR** core, the main reactions involve Ag-Cd-In alloy (control rods), Zircaloy (guide tubes), and Inconel (spacer grids). Ag-Cd-In alloy has a low melting temperature (1073 K) and it is likely to be the first material to melt in the core. The interaction between Ag-Cd-In alloy with the guide tubes material can enhance the core damage by the dissolution of the Zircaloy at a temperature well below its melting point (approximately 2030 K). In addition, the localised interaction between Zircaloy and stainless steel (or Inconel) can induce the formation of eutectics at relatively low temperature.

In a **BWR** core, eutectics may occur between control rod materials (B_4C) and its stainless steel cladding at temperature approximately of 1425 K, well below their melting points (2623 K and 1723 K respectively). The liquid B_4C - stainless steel products can undergo a subsequent relocation in the core with possible blockage formation; in addition, the slumping of the liquefied mixture against the adjacent box wall, can result in the formation of an iron - Zircaloy eutectic and consequent liquefaction of adjacent box wall itself.

<u>Cladding failure</u>

The Zircaloy clad that has not been previously oxidised can melt at about 2030 K and then relocate downward along the fuel rod. This relocation process can reduce the supply of Zircaloy clad that can undergo the oxidation process, limiting the rapid temperature rise and hydrogen generation from the exothermic phenomena involving intact fuel rods. The relocation of the molten un-oxidised Zircaloy is the *first main relocation process* occurring in the core during a severe accident involving in-vessel core melting phenomena. It should be pointed out that this Zircaloy relocation in the core could be prevented in those regions where Zircaloy has been oxidised, until the melting point of ZrO_2 is reached (2970 K).

<u>Material relocation and blockage formation</u>

Although UO_2 and Zircaloy are compatible at normal operating temperature, this is not the case at higher temperatures. When unoxidised Zircaloy clad melts, it can chemically dissolve part of the solid UO_2 pellet and ZrO_2 shell at about 1000 K below their melting points. The resulting molten mixture (Zr, U, O) flows down in the core (candling process) from the higher temperature zones to lower ones where it can solidify and generate a blockage in the flow channels able to inhibit flow and accelerate the core damage. Since the mixture generates decay heat, solidification and re-melting processes occur repetitively as water boils-off and core meltdown proceeds.

<u>Core collapse</u>

The collapse of free standing columns made up of de-cladded fuel pellets is *the second main material relocation process* occurring in the degraded core. The collapsed material creates a rubble bed just over the frozen relocated Zircaloy and liquefied fuel. In this condition, the coolant flow from the lower plenum to the core is virtually interrupted.

<u>Late phase of core melt</u>

The *third major relocation* happens with the failure of 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. This phenomenon is knows as MFCI (Molten Fuel Coolant Interaction): during this phase large quantities of steam can be generated causing a pressure spike (non-energetic interaction). In same circumstances, if the corium mass slumps into the lower plenum, water steam explosion or missiles (energetic interaction) are produced; these events can cause a significant damage to the reactor vessel integrity. Even in the absence of danger to the reactor vessel, a mild steam explosion will affect the debris size distribution and its coolability.

<u>Core debris interaction in the lower plenum</u>

Core debris relocation into the lower vessel may involve mixing of a large fraction of the core inventory with the residual water in the lower plenum. In the absence of steam energetic explosion, the core debris relocation and interaction with the lower vessel water can increase the extent of metal oxidation as well as steam generation. The result would be a transient pressurisation of the primary system. Oxidation of metals is expected to be higher if the water is saturated rather than sub-cooled due to the enhancing effect of a higher steam volume fraction in a saturated water pool. A schematic description of the phenomena involved at different temperature levels is reported in Table 16 [3].

Table 16 Fuel and cladding temperature levels of interest to LWR safety

T (K)	Phenomenon of interest
570 – 620	Normal operating temperature of cladding
970 – 1020	Borosilicate glass (burnable poison) starts to soften
1020 – 1070	Cladding starts swelling. Incipient melting point of alloy inside the control rods
1073	Melting of Ag-In-Cd
1220	Formation of Ni - Zr and Fe- Zr eutectics
1220	Perforation of fuel rod cladding start
1270 – 1370	Zr - H_20 reaction becomes significant
1425	B_4C - Fe eutectics
1500	Inconel – Zircaloy liquefaction. Appearance of visible liquid due to eutectic formation
1520	Zr – H_20 reaction heat generation comparable to decay heat generation rate
1570	Fe – Zr eutectic
1650	Melting of Inconel
1723	Melting of stainless steel
2030	Melting of Zircaloy-4
2170	Formation of Zr(O) – UO_2 eutectic
2623	Melting of B_4C
2970	Melting of ZrO_2
3120	Melting of UO_2

Concerning the dependences of the core degradation upon the initial core heat-up, Fig. 15 shows that for an initial core heat-up rate less than 0.1 K/s, the total core destruction occurs around 380 min after core uncovery; this value is reduced to less than 27 min if the core heat-up proceeds with a rate higher than 1 K/s. Of course these example data must not be directly applied in a PSA. Rather, plant specific analyses are needed to establish the proper time scales. Nevertheless, the example illustrates the high variability in the accident progression.

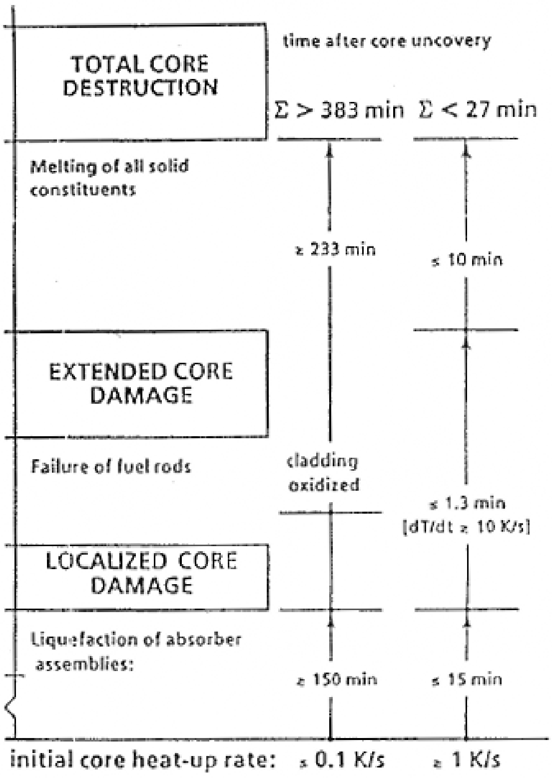

Fig. 15 Dependence of fuel damage process upon the initial core heat-up rate

4.3.1.2 Uncertainties in the phenomenology description

The evaluation of the degree of physical understanding is based on individual experience regarding the uncertainties of available experiments, as well as models verification and codes validation against experiments [4]. Three categories can be considered for the classification of the degree of physical understanding:

- **High**: the phenomenon is well understood, the processes are adequately modelled and well verified in general,
- **Medium**: the phenomenon is on the whole understood, uncertainties remain for unexplored parameter ranges or extrapolation to reactor scale. The main processes are described by adequate models but the verification is not complete by the limited data base,
- **Low**: the phenomenon is only partly understood, the models are rudimentary, the model verification is insufficient.

The level of understanding of different phenomena is reported below for different accident stages: initial core damage, oxidation and hydrogen generation processes, core degradation and melt progression, core debris interaction in lower plenum.

Initial core damage

The initial core damage covers the behaviour of fuel rods, absorber and structural components during the early phase of core degradation including heat transfer, mechanical behaviour, melting and relocation. The main uncertainties are related to the quenching phenomena (especially for high degraded core) and cladding embrittlement (see Table 17).

Table 17 Initial core damage: level of understanding of the main phenomena and consequences

Phenomenon	Understanding	Main consequences
Boiloff	*High* for intact or damaged core	Heat transfer regime locally switches from nucleate boiling to single convection with the vapour phase (dry out), causing significant increase of the wall temperature.
Quenching	*Medium* for intact or partly damaged core, ***Low*** for degraded core	Increase of steam generation, progress in oxidation process and hydrogen production
Fuel-cladding contact	*High*	UO2 - zirconium eutectic interactions
Ballooning	*High*	Failure of the ductile clad with possible oxidation of the inner surface and flow blockage.
Embrittlement	***Low***	Embrittled clad shatters can result in debris formation
Eutectics Zr - UO_2 formation	*High*	Fuel dissolution
Silver-indium-cadmium alloy melting	*Medium*	Cd evaporates and pressurises the absorber rod (clad deformation). Clad and guide tube fails due to the SS - Zr interaction with spreading of absorber melt into the assembly.
Boron carbide - SS cladding reaction	*Medium*	Clad failure and spreading of absorber melt. Attack of the canister walls (BWR) or rod assembly (PWR).

<u>Oxidation processes and hydrogen generation</u>

Oxidation of metallic components changes the material composition of the core and leads to additional heat generation and hydrogen production. These phenomena are generally well understood particularly in the case of oxidation of zirconium and stainless steel (see Table 18).

Table 18 Oxidation processes and hydrogen generation: level of understanding of main phenomena and consequences

Phenomenon	Understanding	Main consequences
Zirconium oxidation	*High*	Temperature escalation, enhancing of core heat-up and hydrogen generation
Stainless steel oxidation	*High*	Less exothermic and then less significant than Zr oxidation process
Fuel oxidation	*Medium*	Higher level of UO_2 oxidation can result in enhanced fission product release and degradation.
Boron carbide oxidation	*Medium*	Generation of compounds such as B_2O_3 (boric oxide), CO and CH_4. B_2O_3 affects fission product transport.
Oxidation by air	*Medium*	Oxidation by air is more exothermic than that by steam but without hydrogen generation. Fuel oxidation by air results in higher release rates for numerous fission products.

Core degradation and melt progression

In the late phase of core degradation, the heat-up and material interactions lead to relocation of core materials, formation of debris beds and of molten pools. Uncertainties are present for many of the phenomena occurring during this stage of the accident progression (see Table 19).

Table 19 Core degradation and melt progression: level of understanding of the main phenomena and consequences

Phenomenon	Understanding	Main consequences
Candling	*Medium*	Relocation of melt material (ceases by freezing or by reaching an obstacle).
Spreading	***Low***	Relocation of melt material
Blockage formation due to mechanical obstruction and/or metallic crust	*Medium* for core region, ***Low*** for lower region	Stop of material relocations. Material interaction. The crust formation can act as a crucible for melt arriving later.
Fuel melting	*Medium*	Relocation of melt material (process influenced by oxygen content and burn-up level).
Fuel rod collapse	***Low***	Fuel may break into pieces and form particulate debris.
Particulate debris production	***Low***	Contribution to the formation of a debris bed and blockage.
Core slumping	***Low***	Possibly oxidation, melt fragmentation and extensive vapour generation in contact with water (which might result in steam explosions). This process is a fundamental precursor for the debris -molten pool behaviour in the lower plenum.

Core debris in lower plenum

The phase of in-vessel core degradation before vessel failure is related to the behaviour of particulate debris and molten pool in the lower plenum. The level of understanding for these phenomena is generally medium / low as reported in Table 20.

Table 20 Core debris in lower plenum: level of understanding of the main phenomena and consequences

Phenomenon	Understanding	Main consequences
Debris bed formation and heat transfer	*Medium / Low*	Relocation of particulate debris from the core region, relocation of melt and its accumulation in the lower plenum. It can undergo heat-up and/or cooling by heat transfer with the surroundings (dependence on debris porosity and permeability).
Pool formation, stratification and solidification	*Medium / Low*	Formed by molten debris or by relocation of melt from the core region and its accumulation in the lower plenum. Molten structure material may contribute to the melt pool. Stratification is possible if immiscible liquid phase is segregated. If water overlies the molten pool, unstable film boiling takes place, with or without crust formation and possible re-melting.
Crust behaviour and in-vessel heat transfer	*Medium / Low*	Depending on the internal heat generation, heat conductivity, heat transfer conditions on both sides of the crust, and the surrounding temperature distribution, the heat flux from the melt to the crust will vary with location and with it the crust thickness. Cracks may be formed if the crust fails locally.

4.3.1.3 Examples of results from in-vessel core degradation to be used in APET

The in-vessel core degradation phase in L2PSA is mostly based on the integral codes like MAAP, MELCOR or ASTEC. These codes are supposed to provide some best-estimate description of the accident progression but detailed analyses of the results are needed before application in a L2PSA. In particular, for the issues classified “low” and “medium” presented before, some uncertainty/sensitivity analysis should be performed and considered in the APET if relevant, as well as consideration of information and assessment apart from integral accident simulations.

Many results can be extracted from in-vessel core degradation calculations and used in a L2PSA APET. The following tables provide some examples of results from ASTEC used in the L2PSAs APET developed by IRSN for the 900 MWe and 1300 MWe PWR [5]. The idea is here to mention some results from in-vessel core degradation which are of interest when building a L2PSA APET. This list may depend on the reactor design and also on the way to build the APET.

Information on the kinetics of the accident

- Time of signal emission for the water injection system,
- Time of signal emission for the spray containment system,
- Time of core uncover,
- Time when the core exit temperature reaches the threshold for SAMG application (TSAMG).
- Time of cladding failure (for fission product release kinetics),
- Time where the molten core reaches 5% before relocation (criteria used for the efficiency of in-vessel water injection),
- Time of formation of molten core,

- Time of the first corium relocation into the lower plenum,
- Time when the maximum mass of hydrogen in containment occurs,
- Likely time of hydrogen combustion during the core degradation (the most likely time is when a limit of ignition by the PARs is reached in one zone of the containment),
- Time of reflooding,
- Time of vessel failure.

Residual power

- At core uncover,
- At time when the core exit temperature reaches the threshold for SAMG application.

Pressure

- Primary pressure (at TSAMG, at first corium relocation, at reflooding, at vessel failure, average value during in-vessel accident progression).

Mass of corium

- Total mass of molten corium (at first corium relocation, at vessel rupture),
- Mass of the first corium slump,
- Mass of the relocated molten core (at reflooding time, at vessel failure time),
- Mass fraction of UO2, Zr, ZrO2, steel, ZrO, AIC, B4C (at first corium relocation, vessel failure),
- Mass fraction of molten core at vessel failure,
- Liquid mass fraction of relocated corium at vessel failure.

Mass of water

- Mass and temperature of water in the lower plenum (at first corium relocation – for in-vessel steam explosion),
- Mass and temperature of water in reactor pit at vessel failure,
- Mass of water in containment sumps at vessel failure,
- Temperature of water in containment sumps (at TSAMG, at the middle of the core degradation phase, at vessel failure).

Atmosphere containment composition, pressure, temperature

- Maximum mass of hydrogen in containment during the in-vessel core degradation phase,
- Mass of hydrogen in containment at vessel failure,
- Mass of the recombined hydrogen during the core degradation,
- Mass production of hydrogen due to Zr, B4C, steel oxidation at vessel failure,
- Mass of hydrogen released out of RPV at vessel failure,
- Mass of already burned hydrogen (at the most likely time) at vessel failure,
- Threshold of hydrogen inflammation by recombiners (the average hydrogen concentration in containment cannot exceed this value, even when including uncertainties on hydrogen kinetics production),
- Total number of moles of gas in containment at vessel failure,
- Mass of oxygen and steam in containment at vessel failure,
- Mass of hydrogen still remaining in the RCS at vessel failure,
- Molar concentration of oxygen, nitrogen, steam, hydrogen in containment at vessel failure,

- Containment pressure (at TSAMG, at first corium relocation, at vessel failure),
- Containment peak pressure due to hydrogen combustion (at the most likely time, at the worst time – maximal pressure peak),
- Temperature of containment average value at vessel failure),
- Temperature of the containment dome (at TSAMG, at the middle of the core degradation phase),
- For the double containment: temperature of annular space gas and wall (at TSAMG, at the middle of the core degradation phase).

Information for the vessel rupture analysis

- Temperature of the vessel head at first corium relocation (for the assessment of in-vessel steam explosion consequence),
- Temperature of metallic corium phase at vessel failure,
- Temperature of oxidic corium phase at vessel failure.

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[65] Method implemented by the IRSN for the evaluation of uncertainties in level 2 PSAPSA. Some examples. E. Raimond, N. Rahni, M. Villermain, Workshop on evaluation of uncertainties in relation to severe accidents and level 2 PSA Cadarache; France 07-11 novembre 2005

4.3.2 Induced RCS rupture including induced SGTR

This issue is important in sequences with elevated RCS pressure. It is important to assess the pressure evolution during the accident because the pressure dependent issues significantly influence the accident progression and consequences:

- Induced failure of steam generator tubes could lead to radioactive releases into the environment via the secondary side of the steam generator (containment bypass),
- Induced failure of a part of the RCS loop (e.g. hot leg or surge line) would decrease the RCS pressure, but could also generate very high thermal and mechanical loads to the containment, including hydrogen combustion,
- Continuous opening of a RCS valve (stuck open due to beyond-design load in the core melt scenarios, or by accident management action) would decrease RCS pressure and lead to release from the RCS into the containment.

If no such pressure reduction occurs in sequences with elevated pressure, challenges by RPV failure at elevated pressure have to be considered. This is addressed in subsequent sections of this document.

In the APET the following branching points should be provided for this issue.

1. When will induced failure of the steam generator tubes occur if the RCS pressure does not decrease?
2. When will induced failure of hot leg or surge line occur if the RCS pressure does not decrease?
3. When will a safety valve open (by accident management or malfunction) if the RCS pressure does not decrease?
4. Will there be an induced failure mode before RPV bottom meltthrough?
5. Which of the three induced failure modes is the first one to occur?
6. Is the first failure mode sufficient for pressure decrease (if there is a leak in a steam generator tube only, the pressure might remain elevated)?
7. Is there a second induced failure mode occurring, decreasing the pressure sufficiently before RPV bottom meltthrough?

4.3.2.1 Description of accident phenomena

Following core uncovery in a core melt accident the connecting pipes to the RPV, in particular close to the hot leg nozzles, and, in a PWR possibly also the steam generator tubes, are exposed to high temperatures. If these high temperatures occur in combination with a high RCS pressure (for scenarios with no or very late depressurisation of the RPV) significant strain is exerted on the components mentioned above. If this strain prevails for a sufficiently long time it might lead to a creep rupture of the respective components. As a consequence, the RCS will rapidly depressurise, while the containment will pressurise. In the case of a PWR, induced SGTR may also occur when the secondary side is dry and preferably depressurised (maximum pressure difference across the tubes). With the valves (in particular main steam safety valves) remaining open or leaking, a containment bypass exists which remains dominant at least until RPV depressurisation into the containment and/or until RPV failure.

The two phenomena are connected because the temperature history of the different reactor coolant system components (pipes, nozzles) and the SGTR tubes are subject to the same accident progression.

Once core damage begins, high temperature steam enters the hot pipe. Part of its energy is lost by natural circulation processes in the RPV upper plenum, heating up the upper plenum structures, particularly for a BWR. The structures that will potentially fail are the hot leg nozzles, in a BWR possibly also the main steam line until the isolation valve, in a PWR the hot leg, the surge line nozzle and part of the surge line (particular for a small leak in the pressurizer), and the SG tubes.

Regarding SGTR, the density difference between the upper plenum and the inlet plenum of the steam generator gives rise to a counter-current flow of vapour in the hot legs. Hot vapour will proceed onto the top of the hot leg and enter the steam generator inlet plenum. Having deposited energy in the steam generator, the steam will flow back into the RPV on the bottom of the hot leg.

At high pressure, the natural circulation is the dominant mechanism for the heat transfer to the structures in the steam generator if there is no small leak in the cold leg. The hot vapour cools as it flows through the tube bundle to the outlet plenum of the steam generator and re-enters via the inlet because of the water plug in the cold leg (counter current flow in the steam generator).

Such counter current flows in the hot legs and steam generators occur in case of liquid water in the cold leg (due to a hot leg break, pressurizer break or no break). In case of a break in the cold leg (including pump seal leakage) a counter current flow cannot be established. On the contrary a uni-directional flow prevails with improved heat transfer to the structures. As a consequence higher temperatures, in particular in the SG tubes, will occur especially if these tubes are not cooled by water from the secondary side. This may be partly compensated by the lower RCS pressure in case of cold leg leak.

The stress in the structures depends on the pressure difference across the structure.

For the pipes within the containment (hot leg for the PWR, main steam line for the BWR) the pressure difference is basically the RPV/RCS pressure.

For the steam generator tubes, on the other hand, the pressure difference can be much lower in case of a non-depressurised steam generator: The highest risk for induced steam generator tube rupture occurs when the primary side is at high pressure and the secondary side has been depressurised. Furthermore, pre-damage of the steam generator tubes (reduced pipe thickness) increases the risk for induced steam generator tube rupture and, in most cases, is a precondition for tube failure. The assumption which pre-damage is adopted in the PSA study is very much plant-dependent, due to enhanced safety rules (such as plugging of SG tubes when a certain damage fraction, e.g. 20% is exceeded). However, recent events occurring on some nuclear plants seem to show that periodic SG tubes inspection methods do not always have the capacity to record even significant tube weakness and this should also be kept in mind.

Generally speaking, passive failure is more important for PWR than for BWR because of the potentially higher pressure and higher temperatures. The consequences, in general, may be beneficial (avoid RPV melt-through under high pressure) or detrimental (loading the containment with a vigorous and extremely hot gas flow containing hydrogen).

4.3.2.2 Calculation of Induced Rupture with the Larson-Miller Criterion

The problem arising is how to predict a rupture time for components submitted to a given time dependent stress and temperature load. Considering the different consequences of different rupture locations, it is not only of interest whether a rupture occurs at all but it is also important to know which component will fail first.

Such predictions rely on tools classically used in mechanics. A damage function may be associated to each structure. Its value is in the range [0; 1] and rupture occurs when the damage value is one. For creep phenomena, the damage is commonly computed through the Larson-Miller correlation linking stress and strain.

In the Larson-Miller correlation, the Larson-Miller parameter LMP is correlated to the logarithm of the piping stress σ_r using fitting parameters. These parameters *A*, *B*, and *C* depend on the material properties of the hot components.

$$\log_{10}\sigma_{\mathrm{r}} = A - B \cdot LMP$$

This equation can also be written in the following way:

$$LMP = \frac{A}{B} - \frac{1}{B}\log_{10}\sigma_{\mathrm{r}}$$

The stress σ_r obtained for linear elastic behaviour is defined as dependent on the pressure difference ΔP (in MPa), the pipe radius *r* (in m) and the pipe wall thickness *x* (in m):

$$\sigma_{\mathrm{r}} = \Delta P \cdot \frac{r}{x}$$

Creep is an integral (time-dependent) response - the material response (creep strain) depends on the applied load (temperature and stress which is a function of pressure) and the duration of the load. If the load exerted on the material is constant over time there is the following relation between the time to rupture of the component t_r and the Larson-Miller parameter with *T* being the temperature in K.

$$LMP = \frac{T}{1000}(C + \log_{10} t_{\mathrm{r}})$$

This equation can also be written as:

$$t_{\mathrm{r}} = 10^{\left(\frac{1000}{T} \cdot LMP - C\right)}$$

A creep damage time-dependent function $CF(t_{end})$ can be computed from the rupture time as:

$$CF(t_{end}) = \int_0^{t_{end}} \frac{1}{t_r} dt$$

For reactor structures, pressure and temperature are not constant but are a function of time. In this case, an instantaneous rupture time $t_r(t)$ can be associated to each load couple (temperature - stress) and so to each instant as both temperature and stress are time-functions. The damage function is then expressed as a function of the instantaneous rupture time:

$$CF_{t\mathrm{end}} = \int_0^{t\mathrm{end}} \frac{1}{t_{\mathrm{r}}(t)} \mathrm{d}t$$

Creep rupture occurs when the damage fraction reaches 1.0. The corresponding time is adopted as the true time to rupture.

The three parameters A, B and C are material dependent. They are estimated through creep tests on specimens. Uncertainties about the parameter values may be important. On one hand, creep tests are harder to perform: they may be very slow at low temperature and are difficult to operate at high temperature. On the other hand, discrepancies in the results for a test under similar conditions are commonly very important.

4.3.2.3 Application to PSA

A method is described how to apply the creep rupture concept to a PSA, in particular how to define the corresponding branch probability (or probability distributions) for an APET.

The above mentioned method yields one value for the particular conditions of the plant and the material parameters that influences creep rupture. In PSA a distribution (percentiles) of branch probabilities is needed to quantify an APET and to provide the corresponding uncertainties to the APET which allows the quantification of the tree, e.g. with regard to the frequency of a Release Category, and their frequency distribution, combining all uncertainties from L1PSA and all of the L2PSA branches.

A practicable method to achieve this goal is the Monte Carlo method in combination with the Bayesian treatment of Variability and Ignorance.

"Variability", also called "aleatoric uncertainties", means uncertainties associated with the accident progression and the situation of the plant at the moment of interest. Those values are in the frame of a L2PSA normally derived from deterministic calculations with codes as MELCOR, MAAP or ASTEC and refer to:

- Temperature of the pipe,
- Pressure difference through the pipe,
- Pre-damage of the pipe, in particular steam generator tubes,
- Time when risk of partial failure ends, e.g. RPV failure.

On the other hand, "ignorance", also called "epistemic uncertainties", means uncertainties of physical parameters, e.g. the Larson-Miller Parameters A, B, C and the underlying material properties. In principle, those uncertainties can be reduced by better measurement.

In a practical sense two Monte-Carlo loops are used and the calculation is performed in several steps:

- Selection of distributions for the epistemic parameters,
- Draw a set of parameters according to the epistemic distributions.

At this point the aleatory loop begins:

- Selection of distributions for the aleatory parameters,
- Draw a set of parameters according to the aleatory distributions,
- Calculate a single value of the creep rupture time with the drawn epistemic and aleatory parameters,
- Draw a new set of aleatory parameters to finally end up with a distribution of the rupture time,
- By comparing this distribution with the time, when risk of passive rupture ends (e.g. RPV-failure) a single value for the branch probability is gained.

Now one returns to the epistemic loop:

- Select another set of epistemic parameters.

Within the (repeated) aleatory loop a new branch probability is calculated, as described in steps 3 to 7.

Repeating these two loops several times one finally ends-up with a distribution of the branch probability that can be used for error propagation in the APET.

Example:

Here an example is provided which shows how uncertainties of temperature and material properties are propagated with the Bayesian method to obtain an uncertainty distribution of the rupture probability.

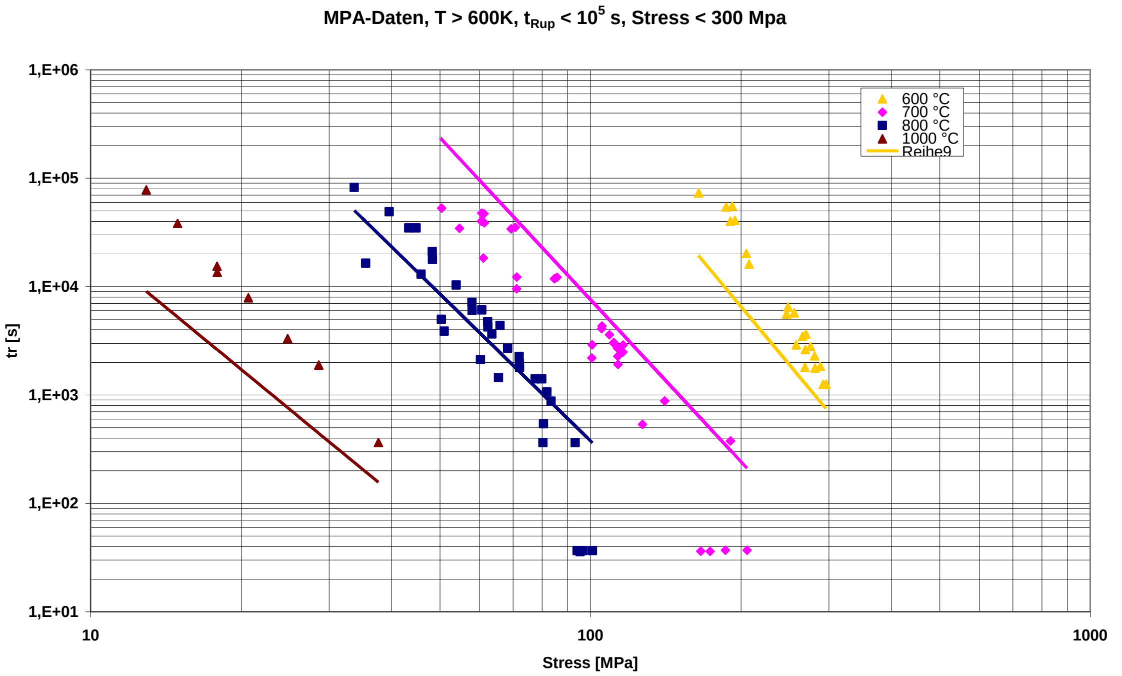

Fig. 16 Instantaneous time to rupture as a function of stress for different temperatures

Fig. 16 shows the instantaneous time to rupture for 20MnMoNi55 steel as a function of stress for different temperatures. Fitting this data, uncertainty distributions for the Larson-Miller fit parameters have been obtained.

To see the influence of temperature, Fig. 17 shows the result from a single representative MELCOR calculation run, performed for a high pressure transient related to a severe accident for a German 1300 MWe PWR nuclear power plant. As can be seen from this figure, the instantaneous time to rupture $t_{R,f}$ (t) drops by approximately 16 orders of magnitude as the temperature T(t) rises from approximately 600 K to approximately 1100 K, which reflects the exponential dependency between the time to rupture and the temperature. In a core melt accident the critical time window between onset of core melt and RPV bottom failure is in the order of several thousand seconds. As can be seen from the figure, this corresponds to temperatures above 900 K.

Because of this strong temperature dependence, the time until failure is nearly exclusively determined by the range of highest temperature values, which in this case corresponds to the time interval shortly before the time to RPV failure t_{VF}. In addition, relatively moderate uncertainty in the temperature can result in a very large uncertainty in the time to rupture. Uncertainties in the time dependent variation of the stress, on the other hand, do not have such a strong impact on the uncertainty in the time to rupture, due to the fact that there is only a linear dependence between stress and time to rupture. To take into account the stochastic nature of the time-dependent temperature T(t) and the time-dependent stress σ (t), these parameters are treated as random variables. However, in the context of the complete accident progression, it has to be remembered that the temperature depends on various parameters, one of them being the heat generation inside the RPV which in itself is strongly linked to the process of corium oxidation and hydrogen generation.

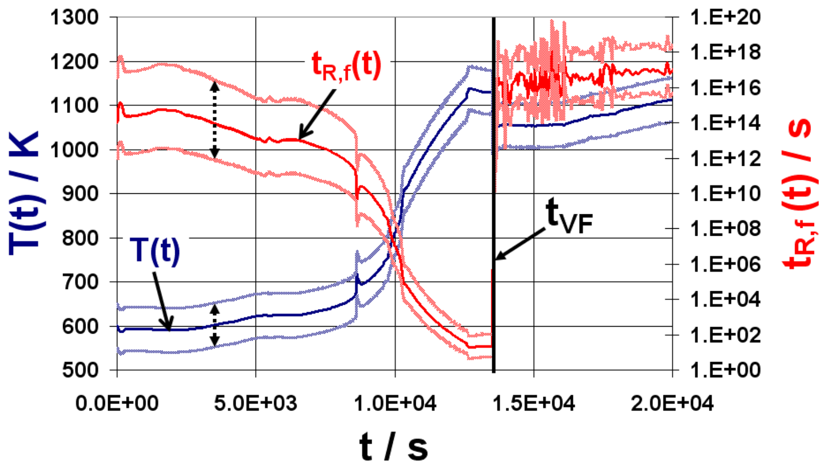

Fig. 17 Result of a deterministic calculation for high pressure scenario for a 1300 MWe PWR: Time-dependent temperature variation and corresponding instantaneous time-to-rupture.

It is reasonable to write the random variables T(t) and σ(t) as the sum of two terms, respectively:

$$T(t) = \overline{T}(t) + \Delta T \text{ and } \sigma(t) = \overline{\sigma}(t) + \Delta\sigma \qquad (8)$$

Here $\overline{T}(t)$ and $\overline{\sigma}(t)$ represents the non-random, time dependent contribution, and ΔT and Δσ the random, time independent contribution to T(t) and σ(t), respectively. The probability distributions of T (t) and σ(t) are thus determined by the probability distribution of ΔT and Δσ, respectively. The probability density functions (pdfs) of ΔT and Δσ can be parameterised in terms of distribution models, based on the deterministic calculations and the knowledge base.

As a simplification, it is assumed that the random variation in ΔT can be regarded as stochastically independent from the random variation in Δσ, in which case the joint pdf of ΔT and Δσ can be factorised into the individual pdf's of ΔT and Δσ.

Now sufficient information is available to calculate the branch probability and its distribution from the deterministic calculation. A MC-calculation with 1000 iterations in the outer loop and 1000 iterations in the inner loop is performed, for a total of 10^6 iteration steps, creating a sufficiently large sample size to approximate the pdf with acceptable precision. The pdf resulting from the 1000 point values of P($CD(t_{RPV})$>1) and its expected value are shown in Fig. 18. The broad distribution demonstrates the large influence of the uncertainty in T(t) on the final result. In fact, both P=0 and P=1 cannot be excluded as possible branch probabilities depending on the value of ΔT. It is of essential importance to incorporate this uncertainty into the event tree model to be able to quantify the confidence interval for the final L2PSA results.

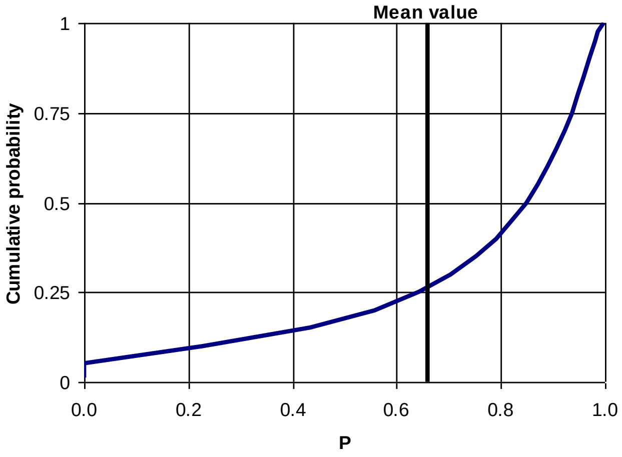

Fig. 18 Probability density function for the branch probability of induced RCS rupture based on the deterministic calculation in Fig. 16.

4.3.3 Hydrogen production

4.3.3.1 Description of accident phenomena

When metallic core components (in particular Zirconium of the fuel cladding) reach high temperatures in the presence of steam, these components will be oxidised and hydrogen gas will be released. The oxidation of one kg of Zirconium will produce approximately 0.044 kg of hydrogen. This reaction becomes significant above 1000°C (1273K). It is exothermic, so that its rate will increase until one of the reaction partners is no longer available. In a typical core melt accident, the heat generation rate by this chemical reaction exceeds the nuclear decay heat. The reaction rate will be limited by layers of Zirconium Oxide on the fuel pin surface, protecting the remaining metallic Zirconium, or molten Zirconium will drain downwards, leaving the zone of high temperatures and freezing temporarily in the lower cooler part of the core.

There are various types of zirconium-based cladding material. With respect to the hydrogen generation in typical severe accident conditions above 1000°C (1273K), there seems to be no significant difference among these materials. This is supported by recent experiments at FZK /Martin Steinbrück: *Oxidation of Zirconium Alloys in Oxygen at High Temperatures up to* 1600°C (1873K), Oxid Met (2008) 70:317–329/.

Safety significance

- The heat produced by the chemical reaction increases the core degradation rate,
- When mixed with oxygen, hydrogen may ignite and create combustion pressure. Depending on the location and degree of a combustion, the containment or vital other components (piping, doors, ventilation systems) may be damaged. Hydrogen combustion is one of the most significant threats to the containment and it could lead to large early releases,
- In small containments (e.g. in BWRs) the hydrogen could contribute significantly to the static containment pressure,
- The metallic oxides produced by the reaction have different properties than the metal components. This affects the following phases of the accident.

Various PSA have shown that hydrogen combustion can be one of the major contributors to early containment failure. Therefore this issue requires a rigorous analysis. In addition to the hydrogen production (which is addressed in this section) the hydrogen distribution inside the plant, potential ignition sources and the combustion process are of high significance. These issues are addressed in other sections of this guideline.

Depending on plant characteristics, the impact of the hydrogen issue may be different, for example:

- Inerting the containment may completely eliminate the issue inside the containment (but in case of leaks, combustion in the reactor secondary buildings could exist),
- Installation of hydrogen recombiners or igniters may reduce the hydrogen related risk. The performance of these accident mitigation measures has to be evaluated by the PSA. Particular attention is needed for situations with high hydrogen production kinetics and / or with dynamic releases into the containment and / or with sequences where steam in the containment atmosphere condenses, increasing the hydrogen volume fraction,
- If there is a high risk of large radiological releases to the environment due to other issues than hydrogen (e.g. due to containment bypass or insufficient containment function) the relative importance of the hydrogen issue may be lower,
- If the containment is robust with regard to high loads from hydrogen combustion, the relative importance of the hydrogen issue will be smaller. However, detonation and deflagration-to-detonation (DDT) phenomena may threaten even robust containments, e.g. by localised mechanical loads or missile generation and impact,
- Hydrogen issue must be considered for accident with containment venting (possibility of combustion in the venting lines).

It is one of the tasks within a PSA to determine the relative importance of the different issues to direct the efforts for the PSA accordingly.

4.3.3.2 Estimates for hydrogen generation based on NEA/CSNI-report

There is a NEA/CSNI state-of-the-art document which describes in-vessel hydrogen production from a general perspective [NEA/CSNI/R(2001)15, 01-Oct-2001, In-vessel and ex-vessel hydrogen sources]. Based on this document, and including a few amendments, the following recommendations can be made for application in PSA.

Zirconium oxidation in intact core geometry:

During core heat up, hydrogen is produced by oxidation of the Zirconium cladding with steam. The underlying phenomena are well understood although some uncertainties exist due to the autocatalytic nature of the process. Other sources of uncertainty are processes such as ballooning and deformation of the cladding during the early core degradation phase which can lead to reduction or blockage of cooling channels and to reduced steam flow. On the other hand, oxidation processes and hence hydrogen production could be slowed down since less steam is available. Nevertheless, it is commonly agreed that prediction of the hydrogen source rate with state of the art accident analysis codes is sufficiently accurate as long as the core geometry remains intact and no water injection occurs.

Steel oxidation in intact core geometry:

Steel oxidation may contribute between 10 to 15 % to the total hydrogen production. Similar to Zirconium, sufficiently accurate correlations are available.

B_4C oxidation:

B_4C is used as absorber material in BWR, VVER and some western type PWRs. B_4C can add significantly to the hydrogen source term. The oxidation process with steam is understood for the temperature range below 1400 K, and correlations are available. Above 1400 K, oxidation kinetics is widely unknown. Quantitative predictions for the high temperature range are not possible at the moment.

Hydrogen production during reflood and quenching:

Reflooding and quenching of the uncovered core is an important accident management measure to terminate a severe accident transient. If the core is overheated, this measure can lead to increased oxidation of the Zirconium cladding which in turn can trigger a temperature escalation. Relatively short flooding and quenching times can thereby lead to high hydrogen source rates. Strong hydrogen production can only be expected in the case of reflooding at the time where the core is strongly heated but still predominantly intact. In a PSA the magnitude of this time window has to be taken into account. An analysis of hydrogen production during reflood in a large PWR (EPR), using SCDAP/RELAP5 and MAAP4, has shown (as referenced in [NEA/CSNI/R(2001)15, 01-Oct-2001, In-vessel and ex-vessel hydrogen sources]):

- Both codes predicted similar hydrogen production rates during reflood and are in reasonable agreement with rates scaled from QUENCH experiments,
- The codes predicted very different total hydrogen masses, attributed mainly to the different treatment of relocated cladding and core material.

For French PWRs, the hydrogen production during reflood and quenching is considered as a possible threat to containment integrity due to the potential high kinetics of hydrogen release which may lead to temporarily high quantities of hydrogen in the containment before recombiners become effective. The main difficulty for the assessment is the high level of uncertainties related to the capabilities of predicting of hydrogen release kinetics. This is considered as an R&D issue ([66], [67], [68]).

References

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[67] Status of knowledge and modelling to simulate the reflooding of a severely damaged core, F. Fichot, N. Chikhi, O. Coindreau, J. Fleurot (IRSN)

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Hydrogen production during core melt-down:

The late-phase core degradation is characterised by high temperature regimes leading to various processes that influence hydrogen production. These include:

- Melt formation and relocation. The effective surface of reactive materials is changed, and melt is relocated to areas with lower temperatures and it freezes, with the potential of blockage or rubble bed formation and reduced steam flow in the regions above. Since unoxidised materials are relocated first, this relocation process tends to limit the hydrogen production,

- Physico-chemical interactions of reactor materials. These affect physical properties, for instance in reducing the melting temperature of material mixtures, which may accelerate the core degradation process. The oxidation behaviour of material mixtures is not well known and this contributes to the uncertainties in hydrogen production.

A major uncertainty in determining the hydrogen production during a severe accident is the timing of the cladding failure and loss of core geometry. Code comparisons have resulted in significant differences for the hydrogen mass produced, which can be mainly attributed to the different models used for cladding failure and related U-O-Zr melt relocation and oxidation.

<u>Hydrogen production during fuel-coolant interaction:</u>

In the late core degradation phase, hot melt from the in-core area is relocated to the lower plenum, which may be filled with water. According to [NEA/CSNI/R(2001)15, 01-Oct-2001, In-vessel and ex-vessel hydrogen sources] experiments with Zr/ZrO_2 and Zr/stainless steel, with oxidation degrees of up to 40% have indicated that typically 5 to 25 % of the metals are oxidised if no steam explosion occurs, and between 70 to 100 % in the case of a triggered steam explosion. Injection of pure oxidic melts into water, even without steam explosion, may produce hydrogen with an estimated 2 kg hydrogen per Mg UO_2. The associated phenomena are not well understood. To conclude, significant hydrogen masses could be produced during a short period. This topic is specifically addressed in the ongoing OECD SERENA2 programme.

<u>Influence of irradiated and MOX fuel:</u>

Burn-up and MOX use are assumed to accelerate the liquefaction of the fuel and therefore they have an influence on the timing of the accident sequence. As mentioned above, earlier loss of core geometry generally leads to less hydrogen generation, which means that no negative effects are to be expected for the hydrogen issue. But adequate models are generally not available in severe accident codes.

<u>The conclusions are as follows:</u>

During cladding oxidation of intact fuel rods by steam, the hydrogen source term can be predicted within the measurement errors of the experiments. Steel oxidation is modelled satisfactorily; whereas B_4C oxidation is widely unknown for temperatures above 1400 K. Major uncertainties still exist for oxidation during reflood processes. A very limited database is available for the late phase of severe accident scenarios, and where the rod-like geometry has degraded to a debris bed or a molten pool. In these configurations, main uncertainties in prediction of the hydrogen source term are related to oxidation behaviour of Zr-rich mixtures including both the kinetics of oxidation and the steam-debris interaction surface. For in-vessel molten core coolant interaction, only few data sets exist which provide limited information on the hydrogen source term. Irradiation and MOX is unlikely to lead to enhanced hydrogen production.

4.3.3.3 <u>Assessment of in-vessel hydrogen generation by means of integral codes</u>

Most PSAs are based on integral accident simulation codes (e.g. MELCOR, ASTEC, MAAP) for analysing the in-vessel phase of a core melt accident. These codes may provide “best estimate” assessment for hydrogen production and the associated release into the containment. Of course, the most recent code versions shall be used by experienced staff.

The hydrogen generation depends largely on the kind of accident sequence. Therefore for each of the different accident sequences which are relevant in a specific PSA, an integral analysis shall be performed identifying the hydrogen generation rate. In particular, the following types of sequences should be distinguished as a minimum from the point of view of hydrogen generation (other issues may call for analysing additional sequences):

- Sequences with low RCS pressure during core degradation (e.g. large LOCA with ECCS failure),
- Sequences with medium RCS pressure during core degradation (e.g. small LOCA or pressurizer valve leak with ECCS failure),
- Sequences with high RCS pressure during the core degradation (e.g. station blackout without depressurisation through pressurizer valves),
- Sequences with deliberate depressurisation during the core degradation (e.g. opening the pressurizer valves while applying SAMG).

Depending on the scope of the PSA, and on the probability of the related system and component failures, the following additional sequences may be important:

- sequences with reflooding (e.g. by a activation of pump or by accumulator discharge after RCS depressurisation); for these sequences, the injected flow rate may have a strong influence on the hydrogen production kinetics that should be taken into account in the PSA (high flow rate may stop the accident progression, lower rate may increase hydrogen production),
- Sequences with induced failure of primary side components (e.g. hot leg) due to high temperature and pressure loads.

For each sequence in the event tree analysis of the PSA, the appropriate hydrogen generation determined by the integral accident analyses should be applied as "best estimate" value. However, uncertainties of the code predictions must be taken into account in the PSA. Where high uncertainties exist (in particular in case of reflooding), code results shall be supplemented by additional analysis.

Appendix 1.5 in the SARNET-deliverable D71 ("Proposal of harmonised methods to assess hydrogen combustion and immediate consequences of vessel breach issues in a L2PSA") contains a table with the variation of hydrogen production for different accident scenarios which has been calculated with the ASTEC code. For a French 900 MWe PWR, the hydrogen released into the containment before RPV rupture varies from 110 kg to 779 kg (a factor of 7 between the lowest and the highest mass), and the maximum hydrogen flow rate into the containment varies between 0.05 kg/s and 0.15 kg/s. There is no correlation between the generated hydrogen mass and the flow rate.

Apart from different scenarios, the hydrogen generation which is determined by integral codes also depends on several uncertain parameters or models within the code. To obtain a more complete general view on the hydrogen issue, it is advisable to consider these uncertainties within a PSA. The following parameters or models should be considered in this context:

- Equations and parameters characterising the Zr- and steel oxidation,
- Models for the relocation of cladding material and fuel,
- Models for heat transfer between core debris and water in the lower core and RPV bottom,
- Nodalisation of the reactor.

It is recommended that uncertainty analysis should be performed with the focus on the hydrogen issue. Ideally, this uncertainty analysis would comprise a sufficiently large set of calculations, to ensure that the range of hydrogen production can be captured and supported by statistical data regarding its uncertainty.

Several solutions exist to perform this type of uncertainty analysis. In any case, the coupling between hydrogen production in-vessel and the temporal evolution of the containment atmosphere has to be considered. If manual action (water injection, spray system activation) may interfere, it has to be taken into account.

The Monte Carlo Dynamic Event Tree approach ["MCDET - A Probabilistic Dynamics Method Combining Monte Carlo Simulation with the Discrete Dynamic Event Tree Approach", Nuclear Science and Engineering 153, 137-156, (2006)] may be a promising advanced method to cover the aspects of uncertainty and the explicit consideration of the interaction of stochastic influences and the process dynamics. The accident sequences including characteristic results (such as hydrogen generation) and their respective probabilities are automatically generated through the dynamic-stochastic interactions over the time evolution.

Some other solutions were also investigated within SARNET activities [69]. An example of study with KANT is provided in chapter 2.4.4.2 on Dynamic Reliability Methods.

Reference

[69] E. Raimond, T. Durin - L2PSA - Comparison between classical and dynamic reliability methods. Specification and results of a benchmark exercise on consequences of hydrogen combustion during in-vessel core degradation- SARNET – Conference ERMSAR 2007 – Karlsruhe

4.3.3.4 Complementary considerations and quantitative recommendations for in-vessel hydrogen generation

In practice, there are limitations for the hydrogen generation analysis with integral codes:

- The potential number of different accident sequences may be unmanageably large. For example, if reflooding has to be considered there are many possible times for initiation of reflooding and many possible flooding rates,
- If the nodalisation scheme of the reactor is detailed, resources may limit the number of analyses,
- If the nodalisation scheme of the reactor is less detailed, the results may be less realistic,
- Even with numerous variations of uncertain input data, systematic properties of the codes remain unvaried, leaving an uncertainty which cannot be quantified by means of these codes.

As a complement to the hydrogen generation determined with integral code calculations, uncertainties according to the Table 21 below should be taken into account. This recommendation is mostly derived from the NEA state-of the art report mentioned above [NEA/CSNI/R(2001)15, 01-Oct-2001, In-vessel and ex-vessel hydrogen sources]. The concluding hydrogen generation assessment could then be a mean value based on best-estimate integral code results and an uncertainty distribution for the in-vessel hydrogen generation based on state-of-the-art reports for each of the relevant accident sequences. This uncertainty would comprise aspects from the plant specific analyses with integral codes, and from complementary considerations.

Table 21 Recommendations for in-vessel hydrogen generation

Issue	Recommendations for application in PSA
Zirconium oxidation in intact core geometry without reflooding	Calculate hydrogen source rate with state of the art accident analysis codes. Typical result should be approximately 0.2 kg/s for a 1000 MWe PWR as long as the core geometry remains intact.
Steel oxidation in intact core geometry	Calculate hydrogen source rate with state of the art accident analysis codes. Typical result should be about 10% to 15% of the total hydrogen production in this phase.
B_4C oxidation	Calculate hydrogen source rate with state of the art accident analysis codes. Above 1400 K, results are very uncertain, and complete oxidation of B_4C should be considered.
Hydrogen production during reflood and quenching	Reliable modelling of this issue is not state of the art, and in a PSA it is not easy to simulate the many possible reflood scenarios. Nevertheless, this point should be treated in L2PSA in relation with the modelling of SAMG. The hydrogen source due to reflooding might be defined as follows: Between 50% (for fast reflooding) and 100% (for slow reflooding) of that metallic zirconium will be oxidised which is located above the initial water level. Associated hydrogen generation rate shall be taken into account when assessing containment threats.
Hydrogen production during core melt-down	Uncertainties exist for U-O-Zr melt relocation and oxidation. Estimates performed for a typical PWR show that the degree of final cladding oxidation can be in the range of 20-90%. The typical values are between 30-50% depending on the sequence. The final cladding oxidation is lower for fast than for a slow sequence. The water flooding increases the final cladding oxidation. Associated hydrogen generation rate shall be taken into account when assessing containment threats.
Hydrogen production during fuel-coolant interaction	Zr/ZrO_2 and Zr/stainless steel, with oxidation degrees of up to 40% have indicated that typically 5 to 25 % of the metals are oxidised if no steam explosion occurs, and between 70 to 100 % in the case of a steam explosion. Injection of pure oxidic melts into water, even without steam explosion, may produce hydrogen with estimated 2 kg hydrogen per Mg UO_2. If SAM could interfere with this mechanism of hydrogen production it should be considered. Associated hydrogen generation rate shall be taken into account when assessing containment threats.
Influence of irradiated and MOX fuel	No increased hydrogen source is to be expected.

4.3.4 Vessel cooling from outside

4.3.4.1 Description of accident phenomena

If there was inadequate core cooling during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of the reactor vessel. However, it is possible that the vessel head could remain intact by vessel cooling from outside in certain favourable conditions. Consequently the relocated core materials will be retained within the vessel.

The aim of reactor pressure vessel cooling from the outside is retention of molten corium inside the reactor pressure vessel or at least delaying of reactor pressure vessel melt-through. If it can be demonstrated that the

RPV will maintain its integrity in severe accidents, then all ex-vessel phenomena threatening the containment integrity can be excluded. This makes in-vessel melt retention (IVMR or IVR) an attractive severe accident management concept.

The basic idea of IVMR is to prevent RPV melt-through by flooding the cavity and transferring the decay heat from the molten corium on the RPV lower head into the water surrounding the vessel. There must be adequate heat transfer efficiency to ensure the RPV maintains its structural properties and is able to support the mechanical load resulting from the weight of corium and possible internal pressure. Note that a low internal pressure combined with a high level of water outside the RPV may reverse the pressure gradient and eliminate the mechanical load to the RPV bottom by buoyancy. In this case, if a leak occurred in the RPV, water would then flow into the RPV through this opening.

IVMR of core melt is a key severe-accident-management strategy adopted by some operating nuclear power plants and proposed for some advanced light water reactors (LWRs).

Applicability of IVMR to a certain plant design depends on the features of the plant. Features favouring the applicability of external cooling include low power density and large volumes of water in the primary and secondary circuit allowing long time delays in core melt accidents, RPVs with no bottom penetrations, suitable layout of the reactor cavity and lower compartment to achieve a natural cooling loop and allow potential flooding of the cavity.

The RPV may fail despite the fact that the reactor cavity is filled with water providing RPV external cooling. The time needed for the vessel failure is also an issue of interest. External cooling is also studied to identify the timing of the RPV failure in case of reactor cavity filled with water. In this case external cooling might delay RPV failure and when the RPV fails the corium will be spread to the flooded cavity. This is the case for example for the Nordic BWRs.

This chapter of the guideline is concentrated on the case where external cooling is used as a severe accident management concept. Moreover, it includes the most important issues which are to be addressed when the IVMR concept is applied, and also provide some insights on the modelling of the issues in L2PSA.

In principle, the following branches related to the system functions should be defined and quantified in the APET.

1. Cavity is filled with water as designed, water level can be maintained, and pressure level in containment can be kept as low as necessary,
2. Cavity is filled with water as designed, water level can be maintained, but pressure level in containment cannot be kept as low as necessary (this could occur if heat removal fails)),)
3. Cavity is filled with water as designed, but water level cannot be maintained (this could occur if water sources are exhausted, or if necessary pumping functions are lost),
4. Cavity is filled with water, but too late or not enough compared to design specification (this could be due to various system or human failures),

5. Cavity is not filled with water at all (this could for example occur in a SBO, if cavity flooding needs electric power).

The branches which should be taken into account with phenomenological issues are as follows:

1. The maximum local heat load to the RPV is as anticipated,
2. The maximum local heat load to the RPV is significantly higher than anticipated (e.g. due to a strong heat focussing effect at the top steel layer),
3. The maximum heat load to the RPV is significantly lower than anticipated (e.g. because the core melt process has been decelerated by intermediate water injection into the RPV).

Finally, the heat removal from the outside of the RPV has to be addressed:

1. The ex-vessel heat removal is as anticipated,
2. The ex-vessel heat removal is worse then anticipated (e.g. due to local steam pockets [BWR], or due to clogging of flow paths [insulating material]).

4.3.4.2 Issues to be addressed in L2PSA

To apply IVMR as an accident management strategy, some issues need to be resolved. Reliable submergence of the RPV with water and the heat transfer efficiency from a fully molten corium pool to the cavity water can be noted as the first priority. These issues shall be addressed and investigated in L2PSA. To show the applicability of IVMR as a SAM strategy feature these issues must be resolved satisfactorily for all the accident scenarios with planned IVMR.

The reactor cavity must be filled with water up to a level higher than the maximum melt level. If a natural cooling loop (between cavity and lower compartment) for long term heat removal will be available, the water level depends on the reactor cavity and lower compartment geometry. Water injection to the cavity can be either totally passive (for example from ice condensers) or partly passive (from bubbler trays with valves opening as in VVER-440/213 reactors). However, active systems can also be used. In fact, water can be injected to the cavity from a storage tank located inside or outside of the containment. Besides flooding the cavity, it should also be investigated whether steam produced in the cavity can flow upwards and natural circulation cooling could be achieved. In addition, it should be noted here that to use IVMR as a severe accident management measure, the frequency of sequences in which the cavity cannot be flooded quickly and efficiently should be very low. These sequences include for example containment bypass sequences, in which the water normally used for cavity flooding would be lost from the containment via a bypass route.

Efficiency of the heat transfer shall be investigated. It has to be studied under which conditions the actual heat flux into the water at the RPV outer surface is below the critical heat flux (CHF). The main task is to show that heat from corium pool can be transferred through the vessel wall in such a way that wall temperatures would not increase sufficiently to threaten the structural capabilities of the wall. The most limiting condition regarding the thermal loads occurs when the corium in the lower head is fully molten (except for boundary crust), when a focusing layer exists at the top of the pool and no decay heat is consumed for melting or heating up the debris or structures. This steady state condition is explained in detail later in this chapter.

4.3.4.3 In vessel heat transfer

As already stated above, one of the main issues for IVMR is to investigate whether and under which conditions heat from the molten corium pool inside the vessel is efficiently transferred to the water outside the vessel. Heat flux distribution from molten corium pool has to be known and the incident heat flux from the melt pool performing natural circulation inside the vessel should be lower than the CHF for all polar angles on the vessel outside surface. Knowledge of the average heat fluxes is not enough; heat flux distributions are of crucial importance. Both the actual heat flux through the vessel wall and the CHF depend strongly on the location along the wall.

The in-vessel heat transfer phenomenon depends on the parameters (mass, composition heat source, temperature and position) of the debris bed in the vessel lower head. The debris bed parameters are the result of the core degradation process explained in detail in chapter 4.2.1.

After initial melting and relocation, the core melt might form a temporary debris bed or molten pool on the lower support plate of the core and later it may relocate into the lower plenum following core support plate failure.

When the debris falls on the lower head, most of it will be quenched and re-solidified due to the remaining water available at the bottom. Following this, the water boils off and the debris starts to reheat due to decay power thereby forming a molten pool with layered structure. The chemical and physical material properties of corium have a strong influence on the configuration of the pool.

The presence of a miscibility gap for the different components of corium creates a separation of the different components according to their chemical affinities. Chemical interactions are integral to the formation of different layers. The different layers are either mainly oxidic or metallic. The amount of oxidic and metallic layers depends mainly on the amounts of non-oxidised zirconium and iron. Configuration of the different layers depends on physical properties (density, presence of the crust, etc.).

Different layers can be formed (for example: heavy oxide, metal, light oxide or solid metal, oxide, liquid metal or only two layers: oxide and metal).

For the following assessment, the most challenging pool configuration will be addressed: A fully molten dense oxidic pool with a less dense metal layer on top. The oxidic layer contains most of the decay power. Since the melting point of oxidic material is high, a solid crust will be formed at all boundaries as a result of cooling. Therefore the heat transfer problem simplifies to that of a volumetrically heated pool with isothermal (i.e. equal to melting point of corium) boundaries. The heat flux distribution from the oxidic layer is determined by the natural convection established in the pool. The thickness of the solid crust and the thickness of the RPV wall adjust themselves according to the heat flux distribution.

The upper metal layer of the pool consists of molten metal. This layer contains the metallic components that have risen to the surface of the whole pool during the re-melt phase and partly from steel structures collapsed to the pool during the accident because of strong thermal radiation upwards from molten pool. The heat input from oxidic pool below the metallic layer is distributed between the upward and sideward directions. From the upper surface of the metal layer, the dominant heat transfer mode is thermal radiation. As for the side boundary of the metallic layer, the heat is transferred into the RPV wall by convection. However, in the case of metallic melt, the crust is missing and the driving temperature difference for convection to the RPV wall sideward is the difference between the molten metallic pool temperature and the melting point of steel, which is almost 1000°C or K lower than the melting point of the oxidic crust. Additionally, the driving force (temperature difference) of the sideward convection from molten metal layer is significantly higher than from the oxidic pool. The typical pool configuration described above is illustrated in Fig. 19.

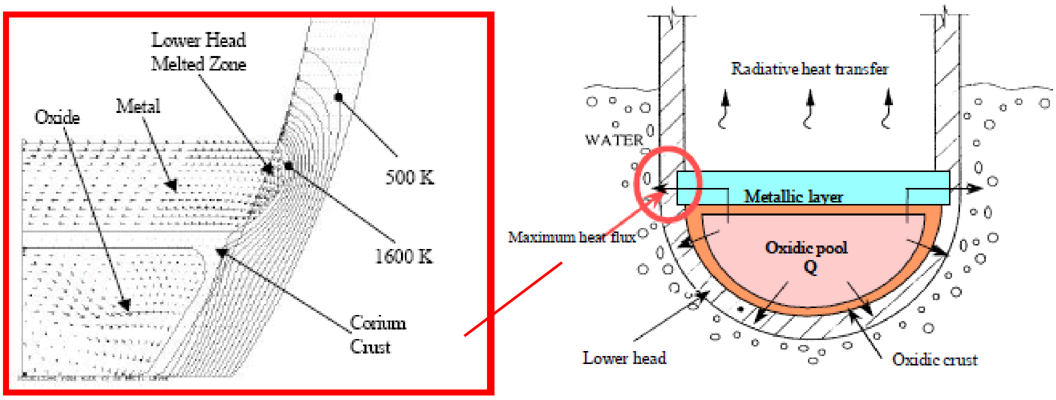

Fig. 19 Critical Pool Configuration /PAR 05/

The molten metallic layer in the stratified pool represents the heat focusing effect. A large fraction of the heat generated in the convecting oxidic pool below travels to the steel layer where some radiates to in-vessel surfaces above. However, a major fraction is transported to the vessel wall through Rayleigh-Benard convection which leads to a focused heat flux on the RPV wall. The focused heat flux is particularly high when the contact area of the steel layer with the RPV wall is small - it is a direct function of the steel layer thickness or of the mass of the steel melted and stratified from the pool.

Heat flux distribution from molten corium has been widely studied. The first studies were made in connection with core melt accidents for fast breeder reactors in the late 1970s and early 1980s. More recent work has concentrated on molten pool heat transfer in the geometry of a LWR lower head. The COPO /KYM 94/ experiments in Finland were carried out with a large-scale, two dimensional facility for the Loviisa plant geometry (elliptical lower head). Frantz and Dhir /FRA 92/ have carried out small-scale, three-dimensional experiments in a hemispherical geometry. ACOPO experiments /THE 95a/ were made with hemispherical geometry and BALI experiments in Grenoble /BON 94/ with cylindrical slab. In all of these experiments, the simulant fluid for corium is water (or Freon in case of Frantz and Dhir). Experiments on prototypic materials have been made in OECD RASPLAV Program /ASM 2000a/ [83] performed in the Kurchatov Institute in Russia, at KTH in Sweden with SIMECO facility /KOL 2000/ [89] and in the MASCA-program (MAterial SCAling), which was divided in two phases: MASCA-1 /ASM 04/ [84] and MASCA-2 /TSU 07/[99] . The LIVE experiments are under way at KIT (Germany) in simulant materials /BUC 2010/ [100].

The results of the RASPLAV and MASCA programs have shown that the typical configuration is a steady-state configuration and that a more complex configuration (three-layer configuration) can be observed during the transitory state. However, it is not clear whether the behaviour found in RASPLAV and MASCA programs is applicable to reactor scale.

An example for an assessment for the Loviisa plant including the focusing effect can be found in /KYM 97/. In the base case calculation for Loviisa, the heat generation in the corium is 9 MW, steel mass 50000 kg (corresponding to a layer thickness of 73 cm and the steel contact area to the RPV approximately 6.9 m²). The detailed evaluation in [91] for the base case reaches approximately 420 kW/m²heat flux from the steel melt into the RPV wall. A more recent publication /TAR 09/ confirms this assessment based on analyses with the ASTEC code. /TAR 09/ obtains slightly higher peak heat fluxes (550 kW/m²), but this is obviously due to neglecting thermal radiation from the top steel layer. Taking into account the completely different approaches in /KYM 97/ and /TAR 09/ there is good agreement between the results.

To define heat flux for the Loviisa case, a Monte Carlo study was made varying the most important parameters: emissivity of the metal pool surface, mass of steel layer, volume of oxide pool and power density in the oxide layer. Each parameter was defined with a probability density function (pdf) and the probability distribution for the sideward heat flux from metallic layer was calculated with Monte Carlo. This probability density function is shown in Fig. 20. It can be seen that with high confidence the heat flux in Loviisa case will be lower than 800 kW/m^2. Details of the study can be found in reference /KYM 97/.

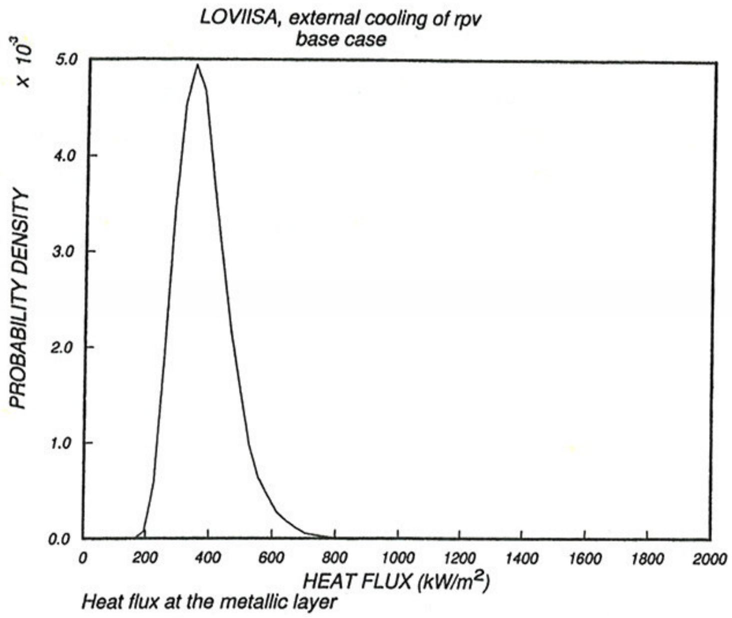

Fig. 20 Probability distribution of the sideward heat flux from the metallic layer in the Loviisa case /KYM 97/

Table 22 provides some examples of MAAP results regarding thermal boundary conditions on the RPV.

Table 22 Paks results: 40-60% of the heat goes up according to MAAP results

	MAAP uncertainty calculation				MAAP BE Results
	Average		25%	75%	
Oxid mass	55,1 t	33,9 t (0,61)	17,5 t	71,6 t	66,2 t
Metal mass	12,9 t	16,8 t (1,3)	7,3 t	34,0 t	33,1 t
Total debris mass	68 t	43 t (0,63)	18 t	111 t	99,3t
Metallic layer temperature	2270 K	526 K(0,23)	2012 K	2650 K	2027 K
Oxide temperature	2710 K	314 K(0,11)	2609 K	2721 K	2651 K
Decay heat at the bottom of the vessel	5,9 MW	3,4 MW(0,57)	2,8 MW	9,0 MW	8,7 MW
Heat flux to the bottom of the vessel	2,6 MW	1,8MW(0,69)	1,5 MW	2,7 MW	2,2 MW

Fig. 21 provides an example of distribution of thermal flux calculation for the Loviisa VVER-440.

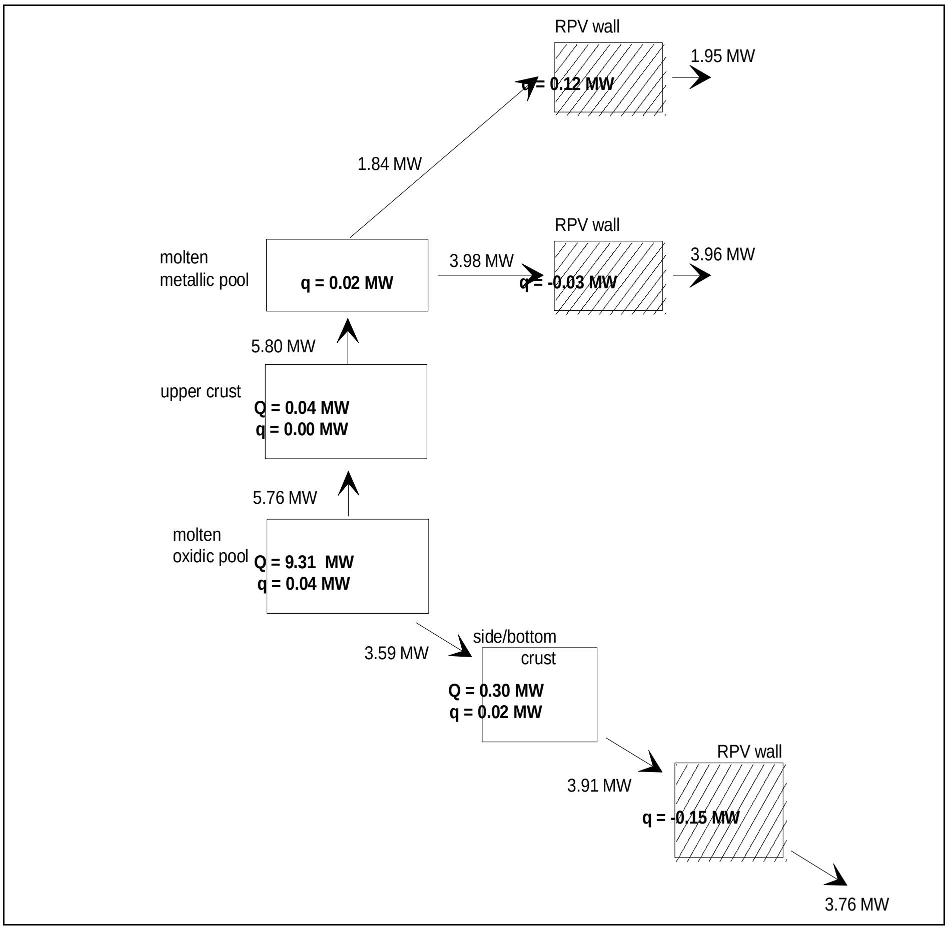

Fig. 21 Example of thermal flux repartition in the case of a VVER 440 severe accident (from LOVISAA - FORTUM - Residual power 9,31 MW)

4.3.4.4 Ex-vessel heat transfer and circulation of coolant

The first issue to be investigated in L2PSA for IVMR is to have a flooded cavity and a natural cooling loop, where the steam produced due to decay heat in the cavity can flow upwards and the water used as a coolant can flow downwards to the reactor cavity. Typically features facilitating the applicability of IVMR are; a reasonably small dead-ended cavity (which can be easily flooded), a suitable lower compartment layout to ensure flow routes to the cavity and availability of steam escape routes from the cavity. It should also be ensured that the cavity can be flooded fast enough in all accident sequences in which IVMR would be used as an accident management measure. Flooding of the cavity can be done either passively or with help of partially active systems (passive flooding with valve opening) or with fully active systems (pumping of water from external water source to the cavity). It should also be ensured that direct contact between the water and the RPV wall is available and not prevented by thermal insulation.

When the reactor cavity is flooded, it is also necessary to avoid any boiling crisis and ensure that the heat transfer mode on the RPV outer surface remains in nucleate boiling (or possibly single-phase liquid convection). If the heat transfer mode permanently changes to film boiling, then the surface temperatures of the wall increase dramatically, which will sooner or later lead to RPV failure.

The critical heat flux, CHF, that can be removed from the RPV outer surface was measured in the SULTAN /ROU 95/ and the ULPU /THE 95b/ facilities. CHF (and also actual heat flux from the in-vessel melt pool) increases with the polar angle on the vessel wall. The lowest CHF values were obtained at the bottom of the RPV. Values measured with ULPU facility indicate that the CHF at the bottom of the vessel, even at zero subcooling and without global circulation loop flow, is at least 300 kW/m^2. Subcooled water in the cavity and global flow loop will increase the value to approximately 400 kW/m^2. The value of the CHF is highest at the cylindrical part of the RPV. The highest values measured i.e. with ULPU facility (for Loviisa type of geometry and AP600 /THE 96/) for the cylindrical wall are ~1500 kW/m^2. For the AP1000, the CHF was enhanced by channelling the natural circulation flow of the outside fluid with baffles and by reducing the friction pressure drop in the flow circuit. Consequently, CHF values of ~1800 kW/m^2 were obtained /DIN 04/. There are nevertheless significant differences between CHF experimental values.

Possible restrictions in the flow areas along the flow routes from the cavity to the lower compartment might significantly reduce the value of CHF if flow oscillations exist. These possible two-phase flow instabilities should be carefully studied for the applicable geometry.

The remaining RPV wall thickness should be adequate to carry the mechanical external load (i.e. mass of corium and internals) and thermal stresses. This is further elaborated in the next section.

4.3.4.5 RPV structural analysis relevant for IVMR

Stresses and loads in the wall are multi-dimensional and relatively complex, but for the case of IVMR the actual interest is the ultimate failure criterion which can be estimated in a very simplified way. Assuming nucleate boiling on the RPV outer surface and assuming a thermal load caused by a molten corium pool, the failure pressure can be estimated from a basic equation of structural analysis for thin shells:

$$p_{fial} = \frac{2\sigma_{cr}\delta}{r}$$

In the equation above, σ_{cr} is the failure strength of steel, δ is the thickness of the wall and r is the average radius of the wall. Steel loses its structural capabilities at approximately 800°C (1073K) and the thickness of the intact wall has to be chosen to represent the portion of the wall at temperatures below 800°C (1073K). It should be mentioned here that the admissible stress above 400°C (673K) is not the nominal material strength. However, from this basic equation the minimum wall thickness can be calculated. If the primary system is at low pressure then the load will only be represented by the weight of the corium pool and the lower head itself. Thermal stresses acting on the wall can normally be neglected, when estimating the failure load of the RPV.

Material properties for the (German) RPV steel 20 MnMoNi 5 at elevated temperatures can be found in /MPA 89/. As a very rough first estimate it can be stated that for temperatures below 400°C (673K) a stress of 490 N/mm^2 can be tolerated for an unlimited time.

If the layer below 400°C (673K) cannot safely carry the loads, layers with higher temperature have to also be considered. For higher temperatures the load bearing capacity of the RPV material continuously decreases and becomes time dependent. Transient analyses may be necessary and state-of-the-art integral accident simulation codes such as ASTEC or MELCOR contain RPV failure models which can be applied for this purpose. However care must be taken to assure that the heat fluxes into the RPV wall are reasonably represented in such analyses. Specific uncertainties / sensitivity may be relevant for this issue.

When performing such analyses for low pressure core melt scenarios with practically zero pressure difference between containment and RPV, it may be predicted that very thin layers of RPV material are sufficient to carry the loads. If it is further considered that the hydrostatic pressure of the water outside of the RPV may partly balance or even exceed the hydrostatic pressure of the molten pool, unrealistic results may appear. Even for large reactors in-vessel retention may seem possible under such conditions. In these cases careful judgment is required in addition to the calculation.

The complex severe accident codes (for example: ASTEC, MAAP, MELCOR) are also used to calculate the RPV failure due to the stress and thermal load in a simplified way. For L2PSA these types of calculations can be accepted if carefully documented and reviewed.

4.3.4.6 Uncertainties important for in-vessel heat transfer

As it can be seen in Table 23 there are a huge number of phenomena that are not well known associated with the IVMR issue but most of them depend on the composition and configuration of the melt pool. It is the main source of the uncertainties. For the issues marked “good” or “reasonable” this guideline should give advice to the user on the pertinent data/codes/methods to be applied in a PSA.

The composition and the configuration of the melt pool as it is formed is a point of concern. To assess IVMR it is necessary to know the zirconium content in the melt, how much steel has been melted and whether the melt pool has been stratified. It is also necessary to understand the chemical reactions among the constituents with the availability of steam and the way they affect the physical configuration and stratification of the pool. The effect of possible stratified configurations on the thermal loading of the vessel wall also has to be assessed.

Corium pool stratification was studied in the RASPLAV and MASCA projects and effects seen in the MASCA program for the case of IVMR are evaluated in a separate paper /TUO 07/.

The RASPLAV experiments show that small changes in chemical composition of the melt might have profound changes in the physical configuration of the melt pool. In RASPLAV-1 experiment with sub-oxidised corium, melt stratified pool existed with the upper layer consisting of ZrC, ZrO_2 and some UO_2. The bottom layer contained the major part of the UO_2 loaded in the vessel and some ZrO_2. Density differences caused the established stratification. However, the second RASPLAV experiment with fully oxidised corium mixture did not show any stratification. Later it was found (this was proven by MASCA, phase one experiment) that an addition of only 0.3 wt% carbon was the cause of stratification found in RASPLAV-1 experiment. However, the RASPLAV experiments showed that the chemistry between the components of the corium mixture at elevated temperatures has to be investigated and this was the purpose of the MASCA program. The RASPLAV and MASCA-1 programs were performed under an inert atmosphere.

The MASCA-2 Project was initiated to perform chemical interaction experiments in small and medium scale facilities. The first part of MASCA-2 was performed in an inert atmosphere. The second part of the MASCA-2 experiments was performed with an addition of steam to the test section to provide more realistic severe accident simulation. The MASCA-2 results were remarkable. It was found in inert atmosphere that the melt pool may stratify into three layers: a light metal layer on the top, an oxidic melt layer in the middle and a heavy metallic alloy layer at the bottom. This more complex stratification may introduce further changes in the heat flux distribution on vessel wall as a function of the polar angle. A matter of great importance is also the portions of the total steel in the melt pool upper and lower steel layers. A layer thinner than a certain value will provide a heat flux, that can overwhelm the CHF at outside surface and could lead to vessel melt-through.

In the last experiment in MASCA-2, steam was introduced into the experimental apparatus after the three layer configuration was formed. In this experiment the two layer configuration was re-established. In an oxidising atmosphere, the Zr and U in bottom steel layer oxidise (before the steel does) and components of the bottom layer separate with steel rising and joining the top metal layer. It is not clear if the same results could have been obtained if the steam had been present all the time.

MASCA results showed that a melt pool may experience more complex stratified configurations than those assumed earlier (Figure 1 configuration). To show the applicability of IVMR the uncertainties considering the phenomena should be taken into account. It has to be shown that the heat flux from the lower head melt is less than the CHF of heat removal at all polar angle locations of the vessel, for all probable melt configurations (e.g. stratifications) with a sufficient margin to cover uncertainties.

MASCA results have demonstrated that the chemical interactions between the constituents of the corium pool play a very important role in the success of the IVMR as a SAM strategy for a LWR. This applies in particular to reactors of higher power than VVER-440s, where the margins are not as high. The melt pool stratification that develops due to the chemical interactions affects the magnitude and angular distribution of the heat flux imposed by the melt pool on the vessel wall. IVMR as a SAM strategy for high and very high power reactors can be considered only through further analytical and experimental efforts.

The issues mentioned above seem to lead to considerable uncertainties when estimating success probabilities for IVMR in a L2PSA. However, until now IVMR has been claimed only for reactors with favourable conditions and where IVMR is part of a formal SAM process. Large uncertainties will however be envisaged if IVMR is not part of a formal SAM process, but where the potential for IVMR has to be assessed nevertheless. In such cases the PSA will either have to spend considerable resources for analysing the issue or it will have to assign large uncertainties to the IVMR success probability.

For some L2PSA studies that aim to be realistic and include uncertainties, the uncertainties of both the conditional probability of vessel rupture and the delay before vessel rupture should be assessed. The variations on vessel rupture time can obviously have an impact on the atmosphere composition at vessel rupture, and then containment failure by DCH and combustion during the ex-vessel phase. Such dependencies are for example taken into account in the French PWRs L2PSA developed by IRSN.

Table 23 Summary of Phenomena Associated with the In-Vessel Retention Issue /ASM 08/

Phenomena	Experimental Programs	Knowledge-base
1. Decay heat and Fission Products		
Residual heat level		Reasonable
Partitioning of the decay heat between layers in case of stratified pool	Under discussion	Limited
FP and residual heat distribution between crust and pool	Under discussion	Limited
FP release from molten pool		Limited
2. Melt thermal hydraulics		
Single phase liquid pool	COPO, ACOPO, BALI, RASPLAV, SIMECO	Good
Complex mixtures	RASPLAV Salt, SIMECO	Limited
Stratified liquid pools	SIMECO	Limited
Oxidic and metallic pools (focusing effect)	Planned SIMECO, RASPLAV-Salt, COPO, BALI	Reasonable
Effect of crust formation on heat transfer	COPO, BALI, RASPLAV, SIMECO	Reasonable
3. Heat flux removal		
Gap formation and heat transfer	CTF[20], FOREVER[21], SONATA[22]	Limited
Boiling on downward curved surfaces	UCSB, Penn.St., SULTAN	Good
Debris bed dryout and coolability	POMECO[23]	Reasonable
Radiation from the upper surface		Reasonable
4. Melt relocation scenarios		
Formation of the initial molten pool in the core	CORA[24], PHEBUS-FP[25]	Reasonable
Melt pool growth and pathway of melt relocation to the lower head	PHEBUS-FP	Limited, depends on in-vessel design
Melt composition		Limited
Additives: FeO, B_4C, etc.	RASPLAV, PHEBUS-FP, CORA	Limited
Interaction with structures	MP tests	Limited
5. Melt composition and chemistry		
Mass of metallic and oxidic components		Limited
Chemistry in liquid phase (melt stratification)	RASPLAV	Limited
Hypostoichiometric oxides and metallic U behavior	RASPLAV, indirectly	Limited
Crust formation	RASPLAV	Limited
Intermetallic reactions	RASPLAV Planned	Limited
Corium properties (UO_2-Zr-ZrO_2)	RASPLAV	Reasonable
6. Vessel failure modes		
Vessel breach, high pressure	LHF (Sandia)[26]	Reasonable
Creep simulation and low pressure breach	OLHF[27], FOREVER	
Irradiated vessel		
Vessel impingement	MVI Project[28]	Reasonable
7. Transient processes		
Jet formation		
Steam explosion	FARO[29], KROTOS[30]	Limited
Fragmentation		
Dynamic loads		Reasonable
Vessel breach		Limited

4.3.4.7 Code calculations

Phenomena important for IVMR can be simulated with code calculations using integral codes such as MELCOR, ASTEC or MAAP. However, in many applications specific simulation tools are also used. With integral codes all the phenomena, i.e. in-vessel heat transfer, structural response of reactor pressure vessel, ex-vessel heat transfer and coolant flow behaviour outside the RPV, are calculated simultaneously. Specific simulation tools concentrate on specific issues in detail.

Very detailed simulations have been made with specific tools in the different areas. However, because of remaining uncertainties, reasonable results for PSA purposes can also be achieved with more simplified calculations. It is important that any tools used are reasonably validated against experimental results available and users of the tools understand the phenomena affecting IVMR.

4.3.4.8 Integral codes

The aim of this chapter is not to justify the use of one code at the expense of another code. Rather, this chapter aims at presenting the main characterisitics of the severe accident codes which are the most relevant to support L2PSA. Moreover, in addition to their use of validated codes, the L2PSA developers should be qualified for code utilisation. L2PSA developers should also be aware of the codes limitations, be able to interpret code results and underlying assumptions, and be able to modify the probabilistic assumptions (in comparison with deterministic calculations results). Finally, L2PSA developers should be aware that severe accident codes cannot always provide a very precise result.

4.3.4.8.1 MAAP

Overview

The MAAP code models the interactions between the in-vessel core debris, its surrounding crusts, any water present in the vessel and the RPV internal structures including the lower head. The energy losses from the lower head outer surface to its surroundings in the reactor cavity and heat transfer to the wider containment are modelled. Models are also included to represent potential actions that could stop the accident by in-vessel cooling, external cooling of the RPV, or cooling the debris in containment (ex-vessel cooling).

There are significant differences in the modelling of melt progression and RPV lower head heat transfer between the MAAP3 code and the MAAP4 code. In the absence of water cover, external heat transfer from the RPV to the containment atmosphere is calculated using natural convection heat transfer correlations. These correlations were also used in early versions of MAAP when water was in contact with the RPV external wall.

From version MAAP4.0.3 onwards, however, a specific model was included for the external cooling of the RPV. The aim of this new model was to support investigations of SAM actions to externally cool the RPV.

MAAP - external RPV cooling model

Within the model, heat transfer to an external water pool is calculated assuming natural convection if the surface temperature is less than the water saturation temperature. For higher temperatures a simplified treatment of boiling at higher temperatures is used. This is based on nucleate boiling calculated using the Rohsenow correlation /ROH 52/, which is modified at surface superheats greater than approximately 9K with an upper limit corresponding to a critical heat flux (CHF) of $4MWm^{-2}$. At greater surface superheats it is assumed that the heat flux remains at the CHF.

This is an approximation to the boiling curve assuming that the RPV surface is heating up and is initially below the temperature corresponding to CHF. This may not be the case in many situations given that this temperature is typically around 415K compared to the usual PWR operating temperature of around 570K. Thus for many severe accident sequences, the RPV temperature will exceed that corresponding to CHF and a full representation of the boiling curve would be preferable. Initially heat transfer will be in the film boiling regime which can correspond to heat transfer rates of an order of magnitude less than CHF.

From MAAP4.0.3 onwards there is more detailed geometric modelling of the RPV lower head region than in earlier versions to provide greater detail during the in-vessel phase. However, a single surface temperature is used for the heat transfer calculation for the entire immersed external RPV surface, although the immersed surface nodes would be expected to have widely different temperatures during periods when RPV integrity is threatened.

MAAP – Example of study to assess potential code modifications for ex-vessel cooling

As part of an external cooling SAM study in the UK /GRI 09/, the simplifications in the MAAP modelling were considered. A modified boiling curve was constructed based on the following correlations:

- Nucleate boiling,
    - The onset of nucleate boiling occurs when the surface temperature exceeds the local fluid saturation value and is characterised by a non-linear increase in heat flux with increasing wall superheat (T_{surf} - T_{sat}). The Rohsenow correlation /ROH 52/ was already used in MAAP4.0.3 and was retained in the modified model for the calculation of heat transfer up to the CHF.
- Critical heat flux,
    - The increasing steam generation rate with increasing nucleate boiling heat flux eventually limits the rate at which water can be replenished at the surface. This sets an upper limit to the obtainable heat flux - the CHF. Its value is generally a function of the thermophysical properties of the fluid and their dependence on the pressure and the sub-cooling of the bulk liquid,
    - Various correlations have been proposed /THE 97/, /ROU 97/ which generally agree on CHFs in the region of 1-1.5 MWm^{-2} when there is a small amount of bulk liquid subcooling due to the hydrostatic head in the facility,
    - The modified model used a correlation developed by Rohsenow and Griffith /ROH 56/ for saturated liquids which predicted similar,
    - Where the reactor cavity has been flooded with recovered water sources, for example townswater supplies, the water sub-cooling can be significant and in the range 30 - 60K at the critical time when the RPV integrity is first under threat by corium relocation into the lower plenum. An allowance was made for this in the modified model by using a correlation developed by Ivey and Morris /IVE 62/.
- Transition boiling regime (a logarithmic interpolation between CHF and the minimum film boiling temperature),
    - This region of the boiling curve is characterised by the heat flux decreasing as the wall temperature increases. The lower bound is the temperature corresponding to CHF and the upper bound is termed the minimum film boiling temperature at which point the surface is blanketed by a vapour film,

    - There was no well established correlation based on thermophysical properties, consequently the modified model calculated the heat flux in this region by using interpolation based on the logarithms of the bounding temperatures,
    - The minimum film boiling temperature was also difficult to establish experimentally with a wide range of published values and correlations. A survey of available information indicated that, for water at pressures of 1-5 bar, a reasonable value was 600K.

- Film boiling
    - In the film boiling regime, heat is transferred by conduction and radiation across the vapour film,
    - Much of the experimental work and the associated analysis is based on small scale facilities where the vapour film was laminar. /HSU 59/ showed that there was a critical Reynolds number associated with the transition to a turbulent film which lead to greater heat transfer rates than would be observed if the film remained laminar. It was shown that the transition corresponded to a critical film length of order 1-10 cm for water in the expected severe accident conditions. The RPV lower head dimensions are an order of magnitude greater so that a turbulent film boiling correlation was considered appropriate. A correlation developed by Bankoff /BAN 60/ which was consistent with the experimental data of /HSU 59/ was chosen to be appropriate.

Station blackout sequences were then analysed for a 3479 MWTh four-loop Westinghouse PWR with a large dry containment using MAAP4.0.3 with and without these amendments to the boiling curve model.

Comparison of the analysis performed indicates that these changes did not have a significant effect on accident progression in this case. There were changes to the detailed RPV surface temperatures in response to the introduction of water. However for this geometry and accident scenario the quasi-steady state heat fluxes did not approach the critical value (CHF).

Conclusions

The modelling of external cooling in the latest versions of MAAP (MAAP4.0.3 onwards) allows potential SAM actions to be analysed and is a significant improvement over earlier versions of the code. The code contains detailed modelling of the RPV lower head region and the many heat transfer mechanisms which are able to deal with the conditions expected in the severe accident analysis in most cases.

Nevertheless, the potential for overestimating the external coolability of the RPV in specific cases due to the simplifications in the external cooling models should be considered when performing analysis.

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4.3.4.8.2 ASTEC

Overview

Regarding the late-phase core degradation, the ICARE [2] module of the ASTEC V2 code is modelling the core degradation and 2-D relocation, interactions between the in-vessel core debris, its surrounding crusts, any water present in the vessel and the RPV internal structures including the lower head. The energy losses from the lower head outer surface to its surroundings in the reactor cavity and heat transfer to the wider containment can be modelled. Models can also be included to represent potential actions that could stop the accident progression by in-vessel cooling (coolant injection into RPV, but only for not too degraded cores in the current ASTEC V2 version, since new models for debris bed cooling are under development), by external cooling of the RPV (intentional flooding of reactor cavity) [1], or cooling the debris in containment after RPV failure (ex-vessel cooling)

Molten pool heat transfer and external RPV cooling model

The vessel lower head can be meshed axially and radially. The meshing is made of truncated cones that allow representing any shape of lower head. When the corium arrives in the lower plenum the following physical phenomena are taken into account:

- Quenching and fragmentation by residual water,
- Molten pool formation,
- Physicochemical phenomena: phase segregation, decantation,
- Crust formation at the upper free level and in contact with cold structures,
- Radiation heat transfer,
- 2-D heat conduction in RPV wall,
- Melting of structures (including RPV wall ablation).

As far as corium layers are concerned, several assumptions are done:

- Uniform horizontal spreading to form a perfectly horizontal layer (O-D approach),
- Homogeneous layer in terms of composition and temperature,
- Heat transfer by direct contact with the vessel walls, the internal structures, and with the other layers in contact,

- Crust formation between the pool and the walls because the wall temperature is lower than the corium phase liquidus temperature. The crust is not “physically” represented (no mass and energy balance for the crust) but acts as a thermal resistance for the thermal transfers between the layers. It is assumed that the crust thickness is very small compared to the layer dimensions.

Two models can be used for the corium configurations:

- Either the user can select and impose an arbitrary arrangement of corium layers,
- Or the code automatically manages the evolution of the corium layers, through models of most physicochemical phenomena like melting of debris and structures, phase separation and stratification, Rayleigh-Taylor instability and mixing.

The thermodynamic equilibrium between a metal layer and an oxide layer can be simulated. Regarding the thermal exchanges, the proposed laws for heat transfer between the layers and between the layers and the vessel take into account major findings from experimental programs such as BALI or COPO/ACOPO. Obviously, heat transfer laws are also configuration dependent.

The external RPV cooling can be managed by imposing boundary conditions on temperature between the external face of the RPV and the containment cavity (coupling between ICARE and CPA modules) or any dedicated thermal-hydraulics device (coupling between ICARE and CESAR). Thanks to control functions (using the module SYSINT for system interaction management), the user is able to set the heat transfer coefficient between the RPV and the fluid inside the cavity, made of gas and potentially liquid: any correlation from the literature in agreement with the RPV geometry and cooling system geometry can be coded. The heat flux is then computed by the core degradation module ICARE and imposed to the external environment of the RPV (either containment cavity managed by CPA module or specific thermal-hydraulics device managed by CESAR module).

An example of validation of molten pool model in lower head with external RPV cooling for LIVE L1 experiment in KIT can be found in [3].

Possible in-vessel retention (IVR) applications with ASTEC code

The ASTEC code enables two basic kinds of IVR application:

- Stand-alone use of the ICARE module,
- Integral application of ASTEC code, where all the modules responsible for in-vessel phenomena (including confinement models) are activated.

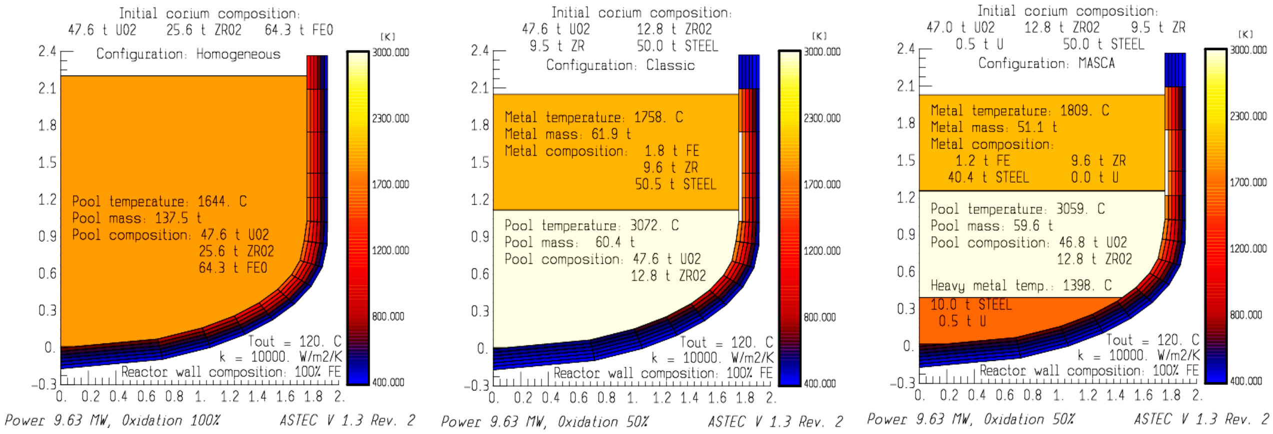
In the current V2.0 version, the 1st option above is strongly recommended (the 2nd option needs yet some numerical consolidation): only the lower reactor head with a corium molten pool is modelled considering the developed (quasi steady-state) molten pool configuration, when heat is generated in corium layer(s) and removed through RPV wall into coolant in flooded reactor cavity (which is represented by boundary conditions). The stand-alone ICARE application enables also modelling of radiative heat transfer from the surface of upper metallic layer into fictitious boundary condition with prescribed temperature, which represents non-relocated reactor structures above the molten pool. The molten pool configuration (arrangement of corium layers), corium mass, composition, and decay heat power generated in molten pool is defined by the user. The user can define arbitrary molten pool configuration, such as homogeneous configuration (considering that all corium components are homogeneously mixed), stratified configuration with light metallic layer above the heat generated oxidic pool (containing UO_2, ZrO_2 and majority of non-volatile FPs) or "MASCA" configuration with the presence of heavy metallic layer below the oxidic pool. Typical results provided by the code are heat-flux profile through RPV wall, wall ablation profile, mean temperature of corium layers, 2-D temperature field in RPV wall, etc. The stand-alone application of ICARE represents an efficient fast-running tool convenient for typical IVR applications, which allows performing of large number of sensitivity studies.

In the typical integral IVR application, the chosen accident sequence is modelled starting from initiating event through core heat-up, degradation and relocation, quenching of the molten corium by residual water in lower head, remelting of the debris in lower head and formation of developed molten pool. The ASTEC code allows calculating the decay heat from defined initial inventory of fission products in core fuel. Thus, together with molten fuel, also the non-volatile fission products (and corresponding decay heat) are relocated into lower plenum. Therefore in an integral IVR application of ASTEC code the time-dependent molten pool mass, composition and decay heat power is calculated by the code. The intentional reactor cavity flooding is modelled by replacing the (nearly) adiabatic boundary condition on outer reactor surface (representing the dry cavity) by user-defined heat transfer coefficient (representing the wet cavity) and actual coolant temperature in reactor cavity. After this time in the accident sequence, heat transfer from reactor outer surface into coolant in reactor cavity is modelled. As soon as the coolant in cavity (more precisely, in riser part of external reactor vessel cooling (ERVC) loop) becomes saturated, steam is produced here and released into confinement atmosphere. The most important IVR related outputs from such integral analysis are the time-dependent heat flux distribution through RPV wall and wall ablation profile. Another important information is e.g. coolant mass on confinement floor that is available for reactor cavity flooding or steam generation in flooded cavity and its impact on the confinement atmosphere. Thus, integral IVR applications of ASTEC code allow complex views on the analysed accident sequence and provide important links to accident management.

Application to reactor case

Besides new advanced PWR designs such as AP-1000, the in-vessel corium retention via external reactor vessel cooling has been recognized as a feasible and promising severe accident management strategy also for older VVER-440/V213 reactors. Analytical studies were performed [4] using ASTEC code in order to demonstrate that the safety margin of IVR concept, which was proposed for VVER-440/V213 units, is sufficient for preserving the RPV integrity during severe accidents. Both stand-alone DIVA (module predecessor of ICARE in the ASTEC V1 series of versions) and integral ASTEC applications were used in this work. Results of sensitivity study to molten pool configuration depending on the Zr oxidation index C_n are illustrated on Fig. 22.

An integral IVR ASTEC V1 application was performed for a MB LOCA sequence without availability of active emergency cooling systems. Reactor cavity flooding was applied when there was sufficient mass of coolant gathered on the confinement floor. The result of energy balance in reactor vessel is illustrated on Fig. 23. It can be seen that since about 8 hours the whole decay heat generated in developed molten pool is removed through RPV wall into coolant in flooded reactor cavity and hence (in the form of steam) into confinement.



homogeneous C_n ~ 100% | stratified 30% < C_n < 100% | "MASCA" 30% < C_n < 50%

Fig. 22 ASTEC IVR applications. Different molten pool configurations depending on Zr oxidation index

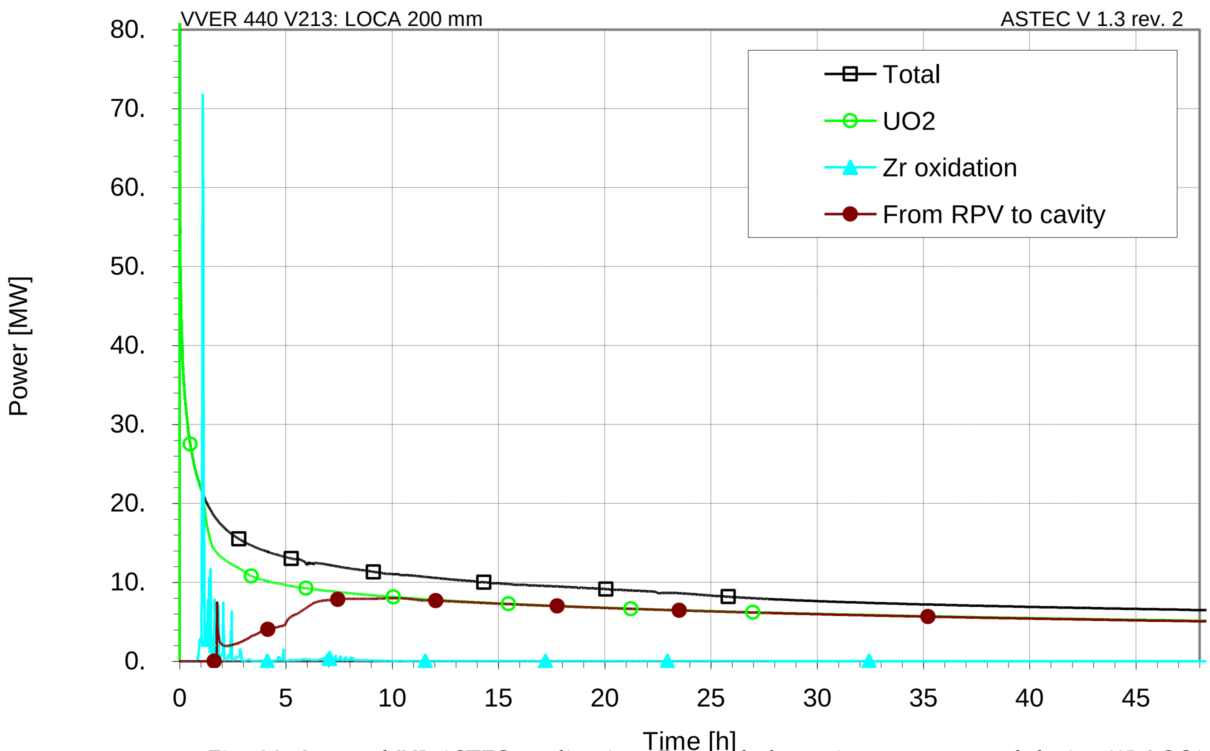


Fig. 23 Integral IVR ASTEC application. Energy balance in reactor vessel during MB LOCA.

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4.3.4.8.3 Specific simulation tools

Description of CORIUM-2D code

Main features

To analyse the dynamic thermal behaviour of corium-structure-coolant systems from an engineering point of view, an original simulation tool called CORIUM-2D was developed by a team of Italian researchers in the framework of international cooperation and is presently owned and managed by ERSE.

Considering the system as having a slow evolution with time, together with the large uncertainties on the distribution of materials and heat sources in the configurations to be considered, a fast-running approach based on mass and energy balances was chosen for this tool. Although this choice neglects the momentum transfer, it takes into account the liquid corium convection with some simplifications. The model calculates, in two-dimensions, the corium and structures temperature field assuming:

- Arbitrary number of cells, not equally-spaced nodalisation,
- Eulerian approach,
- Corium, structural material and coolant properties self-calculated as a function of temperature,
- Expansion or collapsing of the corium mass due to density variation and melting,
- Corium mass relocation due to the mixing among liquid cells,
- Levelling of corium pool (i.e. overlaying metallic layer).

Each cell of the nodalisation is assigned to a material type and it is assumed to have homogeneous composition; physical properties are evaluated at its centre-point (mean) temperature as a weighted average of the involved components, also distinguishing between liquid and solid phase.

A complete library for corium (also in the oxidic and metallic phase) and structural materials is available; it includes: UO_2, PuO_2, ZrO_2, Zr, carbon steel, SS, uranium, plutonium, concrete, H_2O. Sodium, lead and lead-bismuth eutectic have been recently added in the library for the severe accident analyses of fast reactors.

Numerical solution method and heat transfer models

For each cell of corium or structural materials, only the energy equation is solved:

$$m \cdot c_p \cdot \frac{\partial T}{\partial t} = Q_{dec} + Q_{in} - Q_{out}$$

being m = cell mass, cp = specific heat, T = temperature, Qdec = decay power in the cell, Qin = inlet power, Qout = outlet power. Qdec is computed accounting for the current transient time and the current configuration of material components while Qin and Qout are evaluated considering the current temperature and material distribution, and the following heat exchange mechanisms:

1. Heat transferred by convection within molten corium,

2. Conduction within solid phases (obeying to the Fourier's law),
3. Radiation within solid corium debris and among structures,
4. Solid to H2O heat transfer (natural convection and nucleate pool/film boiling),
5. Corium melting/solidification and crust formation.

To solve the master equation, an iterative modified Euler's predictor-corrector integration method is used.

Heat transferred from the pool to the surroundings

The model is based on Fieg's and Kulacki-Goldstein's experimental observations of internally heated liquids. The models provide some correlations which allow the determination of heat transfer from the pool to the surroundings. The pool is modelled as having three surfaces (top, side and bottom) each one characterised by a proper relation between the Nusselt Na and Rayleigh Ra numbers (Nu = Nu (Ra)). A characteristic conduction heat transfer correlation through liquid-solid interfaces is proposed for each surface.

Heat transfer within the pool bulk

Within 2 cells at distance Δx, two heat transfer regimes are taken into account: a) conductive (k ΔT/ Δx) 1 and b) convective (u' ρ cp ΔT) 2 with u' the velocity of the mass transferred between cells assumed to be the turbulent component of the flow regime. Three simplifying fundamental hypothesis are adopted to account for the convective heat transfer:

i) it is assumed that the characteristic fluid velocity distribution (and then the related Nusselt numbers) generated by corium natural convection is equivalent to the distribution which could be obtained under forced convection, so that:

$$Nu_{nat} = Nu_{forced}$$

This hypothesis allows estimation of a characteristic velocity u∞ of the molten corium having pool height H, at steady state, associated to each cell as:

$$u_{\infty} = 111.65 \cdot \frac{\mu}{\rho \cdot H} \cdot \mathrm{Pr}^{-0.375} \cdot Nu^{1.25}$$

with μ the dynamic viscosity and Pr the Prandtl number;

ii) within a generic cell, the fluid moves in all directions with its characteristic velocity (distinguishing only between horizontal and vertical motions, but not between up and down, and right and left);

iii) the velocity of mass transfer between u' cells may be thought as proportional to the actual velocity u of the fluid within each cell which may be computed from the characteristic velocity u∞ applying a correction factor f τ for mechanical inertia of the corium:

$$u' = \chi\, u = \chi\, f_{\tau}\, u_{\infty}$$

[1] *k is the conductivity of the pool and ΔT the temperature difference between the centre and the periphery of the pool.*

[2] *ρ is the density of the molten material and c_p the isobaric specific heat.*

where χ is the proportionality constant which may be set by the user.

Under this hypothesis the heat transferred within the pool bulk Q, through the area A, is evaluated as:

$$\frac{Q}{A} = \frac{k}{\Delta x}\,\Delta T + \chi\, u\, \rho\, c_p\, \Delta T$$

Debris bed model

In addition to a continuous solid medium, the code also allows the corium to be considered as a debris bed, i.e. as a pack of solid particles arranged in a volume filled with a motionless gas. In this case, the heat transfer mechanisms considered by the code are:

- Thermal conduction through gas,
- Radiant heat transfer between adjoining voids,
- Thermal conduction through solid,
- Thermal conduction through the gas film near the contact surface of two particles, radiant heat transfer between solid surfaces.

An "effective thermal conductivity" ke of the discontinuous material (i.e. UO2) can be defined as:

$$-k_e \frac{\Delta T}{l_p} = q_{gas} + q_s$$

where thermal flux qgas collects the heat transfer mechanism 1 and 2, while the thermal flux qs takes into account the remaining mechanisms 3, 4 and 5 referred to solid phase. The heat transfer model takes into account of the void fraction i.e. how the solid particles are packed. Two limiting cases are considered:

- Loose packing of equal sphere (corresponding to void fraction of 0.476),
- Close packing of equal sphere (corresponding to void fraction of 0.260).

Outside this range (void frac. < 0.260 means that smaller spheres are also present, while void frac. > 0.476 means that the particles are far from being spherical) a real void fraction should be calculated. The model takes into account the strong influence of the voids on the bulk corium conductivity.

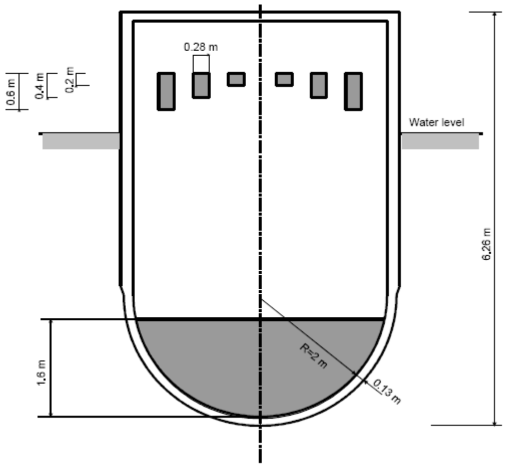
Recent improvements

Improvements have been recently implemented in the code to take into account:

- Radiation among oxidic solid debris (more efficient than conduction when the debris temperature approaches the melting point),
- More recent CHF (Critical heat Fluxes) values for inclined surfaces (Theofanous-Syri),
- Non-analytical solution for radiative heat transfer to reactor internals applicable to whatever plant geometry (using Monte Carlo method and accounting for relocation of molten materials),
- Implementation of heat convection exchange model for Na-cooled surfaces.

Application to reactor problems

The capability of CORIUM-2D code to analyse the thermal behaviour of structures potentially capable to confine the corium in the case of a LWR severe accident scenario is highlighted in the following sample problem. It is focused to the description of heat exchange mechanisms occurring in the case of in-vessel corium retention and coolability. The complete computational model is sketched in Fig. 24.



Corium height	1.6 m
Area of upper corium surface	12.06 m^2
Area of vessel-corium interface	20.11 m^2
Volume of corium in bottom head	11.80 m^3
Void fraction	0.20

Fig. 24 Layout and geometrical data for the in-vessel retention analysis in typical LWR

According to a small-break LOCA severe accident scenario in an advanced 1800 MWth PWR plant, a homogeneous mixture containing approximately 75% of core materials (UO_2, ZrO_2, Zircaloy and stainless steel) was assumed to be relocated into the RPV lower head, while the remaining uncollapsed corium mass is modelled as three separated blocks. To model a granular debris bed in the lower head, a void fraction of 20% has been assumed. The initial voiding of the corium volume will be followed by a progressive downward shrinkage as the corium is melting (the void fraction is assumed to become zero in a fully molten cell). The other essential parameter characterising a debris bed (or granular medium) is the mean diameter of the particle (assumed to be spherical) to which the value of 2.0 cm has been assigned.

The analysis was started at 22000 s (~ 6h) after shutdown, when the boil-off of water contained in the primary circuit and supplied by the emergency cooling systems was completed, and was carried out up 72000 s (20h).

Thus, at the starting time the corium-structure system was assumed to be dry with a mean temperature of only 400 K and the pressure vessel was assumed to be externally flooded by water at 320 K, as allowed by the plant design. The cylindrical computational domain is divided into annular rings with constant radial thickness and axial height of 4 cm. Connection cells with a fairly larger height (up to 60 cm) are also provided but they are either air (empty) cells not directly involved in the mass and energy balances, or structure cells where the axial conduction is almost negligible. In all, 55x87=4785 corium, structure, air and water cells are involved in the analysis.

Two radiation enclosures are implicitly defined: the first is bounded by the upper surface of the slumped corium, the external surfaces of the overhanging corium blocks and the inner surfaces of the pressure vessel, while the other comprises the RPV outer surfaces that are not flooded by water. In all, there are almost 330 separate radiating surfaces whose view factors are estimated by Monte Carlo method. Some results of the calculation are reported in Fig. 25 and Fig. 26.

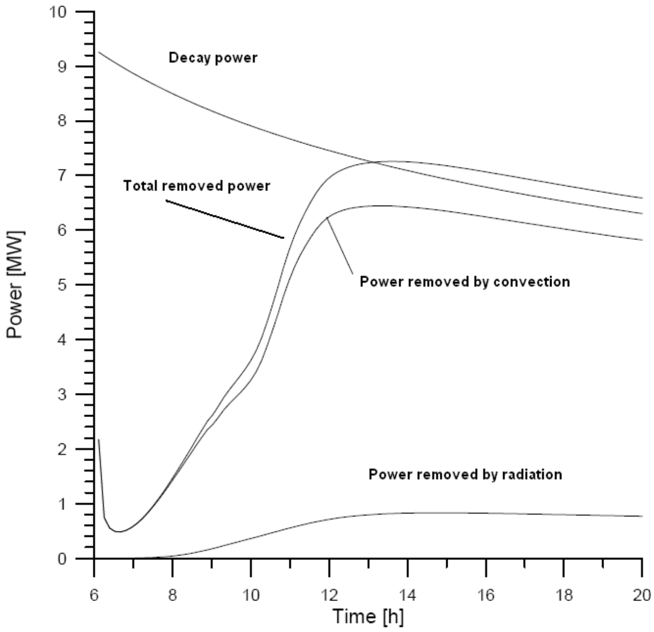


Fig. 25 Partition of removed thermal power between convection (external flooding) and radiation.

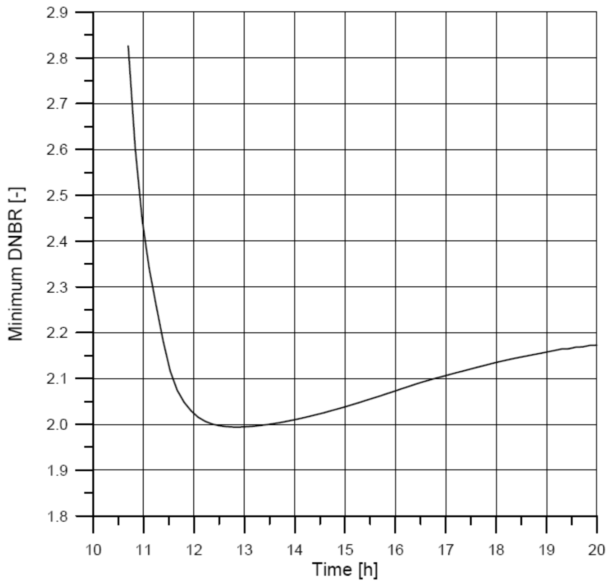


Fig. 26 Minimum DNBR at the structure-water interface.

Fig. 25 shows the different contribution of convection and radiation phenomena to the removal of the total decay power. The heat exchanged between the corium, vessel and water and the corresponding heat transfer regime, is computed by CORIUM-2D. As reported in Fig. 25, the power removed by radiation seems to play a minor role in comparison with the power removed by convection owing to external flooding. A minimum DNBR (Departure from Nucleate Boiling Ratio) above 2.0 (Fig. 26), assures safety conditions for the in-vessel retention case here presented.

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4.3.4.9 External cooling in L2PSA & APET/CET

In L2PSA the boundary conditions of IVMR have to be taken into account. If active systems are used to make external cooling possible, then the reliability of these systems should be assessed in L2PSA. Human reliability should also be taken into account if operator actions are needed. Actions needed are to be included in severe accident management guidelines (SAMGs). The strategy can be used for different reactor designs. The basis of this strategy is common (water cooling of the RPV) for all types of reactors but the activation means of RPV cooling are different. Therefore the L2PSA method to calculate the success of this strategy should depend on the design (see examples).

Unintended flooding of the cavity under normal operating conditions would produce significant risk in terms of RPV thermal shock and embrittlement. Therefore the systems and procedures for cavity flooding must reliably prevent spurious action. A complete PSA would have to consider these potential adverse effects of IVMR.

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4.3.4.11 Examples

4.3.4.11.1 Paks (VVER-440/213)

Main points of L2PSA model for external vessel cooling (IVMR) for VVER-440/213 (Paks).

EOP action:

Operator intervention for water flow from bubbler condenser trays: the operator shall open 12 valves (the water flows from the upper trays to the lower trays through these valves):

- Core exit temperature measurement T>550 °C (823K) (it should be in EOP).
- Operator uses either the normal energy supply or the dedicated severe accident diesel for the opening energy. (L2PSA includes: probability of the operator intervention between the signal and 5 or 15 or 30 minutes, probability of the successful opening of 11, 8, 6 valves with fault tree model of the energy supply and valves).

- Severe accident analysis of the representative sequence of the PDS to determine the timing of the accident without operator intervention (timing of support plate failure, start of vessel lower head heat up, lower head temperature exceeding 500°C (773K), lower head failure).

Water in the containment sump:

- Operator intervention after entering into SAMG
  a. Using SAG 2 : checking the water level in the containment (probability of the failure of the water level measurement).
  b. Decision to flood the cavity (probability of no flooding – primary pressure measurement problem, human error).
  c. Opening a valve (fault tree of the 2 water letdown valves).

Water in the cavity:

- Opening of a passive valve in the lower thermal shield (probability).

Coolability of the vessel based on severe accident analyses and operator intervention analyses:

- Short term failure criteria:
    - Time of the water to reach the vessel lower head (sequence specific), availability of other water sources (ECC tanks) and the timing of the operator intervention (number of open valves of the bubbler trays, delay of operator intervention, timing of the decision to let the water into the cavity).
    - Determination at that time, the:
        - load (melt quantity, composition, properties, primary pressure);
        - cooled surface and critical heat flux (CHF);
        - structural criteria (stresses and wall strength –Finite Element Method depends on vessel wall temperature).
- Medium term:
  The thermal failure question is whether the heat removal from the outer vessel wall is sufficient to maintain the steady state and the wall integrity (define the thermal load inside the vessel for the scenario and to determine whether it is below the heat removal limit, CHF at outer wall).
- Long term:
  It depends on the availability of the containment spray system. Probabilities for the restoration and/or long term working of the spray system are included in the L2PSA.

4.3.4.11.2 Loviisa (VVER-440 with ice condenser containment)

Loviisa NPP, a VVER-440 reactor with ice condenser containment, is owned and operated by Fortum Power and Heat Ltd., and located in south-eastern shore of Finland. In-vessel retention is an essential part and corner stone of SAM strategy designed and implemented for Loviisa NPP during late 1990's.

The case of in-vessel retention for the Loviisa NPP was resolved using ROAAM (Risk Oriented Accident Analysis Methodology). An extensive research program, which included both analytical and experimental studies on heat transfer in molten pool with volumetric heat generation and on heat transfer and flow behaviour at reactor pressure vessel outer surface, has been carried out by IVO (precursor of Fortum) for demonstrating the coolability of corium on the RPV lower head. This has been the analytical base for L2PSA.

Requirements for successful in-vessel retention (including long term stability) for Loviisa are primary system depressurisation, lowering of RPV lower head insulation and neutron blocks to ensure efficient flow circulation around the RPV and flooded cavity. Plant modifications were performed to fulfil these requirements and to fully show the applicability of IVMR for Loviisa. The Finnish Regulatory Authority STUK approved the in-vessel retention strategy for Loviisa in late 1995.

The use of ROAAM affects the L2PSA study of the Loviisa NPP. In the ROAAM study it was investigated whether IVMR will be successful (with high margins) if certain requirements were met. Analytical base and phenomenological uncertainties are included in the ROAAM study and L2PSA modelling concentrates on operator actions, system failures and water availability (the success/failure of basic requirements). To succeed in IVMR the requirements shall be fully met. If this is not the case, the sequence will result in containment failure.

In Level 2 the following issues are studied in the Containment event tree (CET) when success/failure of IVMR is evaluated:

- The necessary operator actions are included in the system failure models: diagnosing of the situation and performing the actions needed in primary circuit depressurisation, forcing open the ice-condenser doors and lowering of the thermal shield below the pressure vessel. All the operator actions will be performed according to EOPs when core exit temperature will rise higher than 450°C. Actions will be confirmed in SAM guidelines. Entrance criteria to the SAMGs are core exit temperature (higher than 700°C) or dose rate measurements (located in containment and outside the pant).
- System failures:
  - Fault tree model for failure of ice condenser door opening mechanisms (mechanisms implemented solely for SAM purposes) → ice condenser doors fail to open → no water for cavity flooding → IVMR will fail,
  - Fault tree model for failure of depressurisation valves (additional SAM lines installed) → if depressurisation fails IVMR will fail,
  - Fault tree model for failure of thermal shield lowering system (system implemented solely for SAM purposes) → thermal shield lowering fails → IVMR will fail,
  - Auxiliary systems are included in system failure models (power supply etc.),
  - In shutdown states the availability of systems have been separately evaluated (unavailability may be caused by scheduled maintenance etc.) and recovery actions needed have been included in system failure fault tree models (operator errors in diagnosing and performing the actions included in plant guidance).
- Water availability for cavity flooding (cavity will be passively flooded from lower compartment in all the cases where enough water is available from different sources: ice condensers, emergency core cooling system water tanks, water accumulators etc.):
  - Water availability from different sources for different sequences (sequence timing, availability of emergency water feed, initial plant state: power operation or shutdown) has been evaluated with containment TH (thermohydraulic) analyses, MELCOR together with COCOSYS and APROS codes have been used for containment calculations. These analyses have been the analytical base for water availability evaluations included in L2PSA models. In L2PSA model it has been separately studied for accident bins (PDS) if there is enough water for cavity flooding or not,

- Typically water availability is not a problem. During power operation, ice in ice condensers will be melted effectively and the narrow small sized cavity will be passively flooded. However for some sequences water availability might be problematic, i.e. in containment bypass sequences water will be lost outside of containment → IVMR will fail. Also in some of the shutdown sequences water availability is problematic if the decay power is low, initial water level in open circuit is low and emergency core cooling injection has failed.

4.3.4.11.3 References

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4.3.5 Consequences of in-vessel water injection (coolability, hydrogen production, RCS pressurisation…)

4.3.5.1 Description of accident phenomena

Without reflooding the core, damage would continue and lead to RPV lower head failure in most reactor designs. Reflooding may provide a chance to prevent this severe consequence. However, other factors need to be taken into account as follows. These issues should be modelled in the APET...

- The reflooding causes oxidation leading to additional and possibly rapid hydrogen production which may threaten the containment integrity,
- The reflooding could increase fission product release from the fuel caused by escalating fuel temperature and induced mechanical failure of the cladding. The enhanced release of fission products to the reactor coolant system increases the source term,
- Mechanical shattering of fuel assemblies or control rods caused by large temperature difference between coolant and the core components may result in formation of debris particles. This has an effect on further coolability of the damaged core,
- In a BWR core the first core structures to melt and relocate would probably be the control rods, leaving the upper part of the core partly void of neutron absorbers. Therefore the reflooding of a partially degraded core in a BWR may lead to recriticality event if the ECCS water is not borated. The recriticality may in turn lead to faster containment pressurisation and earlier release of fission products (see Chapter 4.2.10),
- The strong steam generation during reflooding and quenching may pressurise the reactor coolant system and induce leakage, such as hot leg failure or rupture of PWR steam generator tubes or induce the RPV rupture at high pressure with DCH risk,

- The reflooding with cold ECCS water with subsequent formation of sub-cooled water pool in the lower head in a situation with a molten corium pool in the core region may lead to energetic fuel-coolant-interactions that possibly also pressurise the RPV and threaten lower head penetrations. This is of special interest to plants that are able to credit in-vessel melt retention either by outside cooling of vessel head or by employing control rod guide tubes (CRGT) cooling,
- RPV reflooding may prevent vessel failure even at a later stage, as occurred in the TMI-2 accident. However, the coolability of corium relocated into the lower plenum incorporates high uncertainties.

Plant characteristics may affect the anticipated consequences of the degraded core reflooding:

- In most BWRs the ECCS coolant are designed to be sprayed to the core from above the rods. This is still valid for BWR reactors that during LOCA events lower the water level far below the top of the fuel and get temperature increases above 400°C (673K) on the fuel cladding. BWRs with internal pump have in some cases redesigned their ECC system to avoid spraying the core : instead the ECCS coolant is distributed into the downcomer (between the core shroud and the vessel) and the ECC-cooling only enters the core from beneath. In these cases the maximum cladding temperature is below 400 °C (673K),
- Research programs are assessing the possibility to cool molten core in the bottom vessel by water injection through the control rod debris cleaning flow path. Code development to cover this cooling mode is under development. This cooling mode could be used in assessing the time to vessel breach. If not included it will give conservative value for time of vessel breach,
- ECC water is always unborated in BWRs and when power comes back after melting control rods but not all of the fuel a recriticality event will occur. This will cause the reactor to produce steam which will be transferred to the condensation pool and increase the rate of heat-up. This will then generate high temperatures in the containment. In PWRs the ECCS is mostly borated with much lower probability for achieving recriticality conditions with cold coolant injection,
- Some PWRs and all BWRs have penetrations in the lower head. Fuel-coolant interaction may threaten the thermally weakened penetrations. These penetrations are possible means for the core melt to initiate vessel breach. A detailed assessment has to be done including last years code development to assess the effects from core melt on the penetrations and the time required for core melt to fail these penetrations. Knowledge about cooling effects from water injection through the control rod debris cleaning flow path is also essential in such an assessment.

<u>Coolability of fuel rods</u>

Separate-effect tests and integral experiments have been conducted or are in progress to investigate the quenching of a degraded core, where the fuel rods are in a mainly rod-like geometry at the time of core reflood. The INEL experiments, OECD LOFT LP-FP-2 [Coryell, 1994] and PBF SFD [Ostek, 1987], and CORA experiments 12, 13 and 17 [Hagen et al., 1996] contribute to the general database on the quenching of hot, damaged bundles. More recently, single rod and bundle tests performed at KIT in Karlsruhe within the ongoing QUENCH program have provided valuable new data, supplemented by the VVER test CODEX-3 at AEKI in Budapest. Bundle experiments have addressed integral effects, while single rod tests on hydrogen absorption and release by the cladding have sought to identify the mechanisms in detail.

Separate effects tests have clearly demonstrated the effects of oxide cracking and oxidation of newly exposed metallic surfaces, as well as the role of hydrogen absorption and release, under a wide range of cooling conditions (cold steam vs. reflood water to temperatures up to 1873 K). No separate-effects data are however available for temperatures above 1873 K. QUENCH bundle tests have shown temperature excursions and excess hydrogen production for quench from high temperature (2300 K) with a non-preoxidised bundle, while smooth cooling with no significant excess hydrogen production has been observed for quench from lower temperatures (1750-1870 K) and with pre-oxidation [Haste & Trambauer, 2000].

The present knowledge based on experiments is summarised in [W. Hering, Ch. Homann (Karlsruhe Institute of Technology), "Degraded core reflood: Present understanding and impact on LWRs", Nuclear Engineering and Design, Volume 237, Issue 24, December 2007, Pages 2315-2321].It can be concluded that successful cooling in the original core region is possible if the core temperature does not exceed 2200 K when reflooding begins and if the flow rate is at least 1g/s per fuel pin. If the flooding rate is below 2 g/s per fuel pin, the related hydrogen generation can be significant. However, there is substantial uncertainty in the hydrogen generation, in particular if there has been only a small hydrogen generation before reflooding started.

Coolability of control rods

The control rod degradation starts with chemical reactions between B4C and stainless steel cladding and liquefaction at elevated temperatures which causes clad failure and spreading of absorber melt. This may attack the canister walls (BWR) or rod assembly (PWR).

The interaction between stainless steel and B4C begins as soon as the temperature exceeds 1073 K. Liquid formation in the interaction zone starts at about 1500 K. Stainless steel melts completely in the range 1500 K … 1800 K. The molten stainless steel/B4C mixture interacts with control rod guide tube walls (Zircaloy) and commences the failure of the guide tube. Once a hole is formed on the guide tube wall, liquefied stainless steel-B4C mixture flows down, providing pathway for steam to oxidise the remaining B4C [Seiler et al., 2008].

The coolability of core is reasonable well understood and modelled before significant melt relocations for melt pools. The coolability of controls rods by reflooding can essentially be reached prior to melt formation in the control rod i.e. at temperatures below 1500 K. Since the control rods are not heat generating structures and stainless steel oxidation takes place only in minor scale, the coolability of stainless steel structures and even steel melts can be relatively easily achieved. Current state-of-the-art integral computer codes can model control rod melting/ coolability sufficiently accurately for PSA L2 purposes.

Cladding embrittlement and debris formation

The thermal stresses generated by rapid cooling by reflooding may cause mechanical damage to the zirconium cladding by cracking, chipping, fragmentation and shattering [Van Dorsselaere et al, 2006]. The fragmentation will lead to formation of an in-vessel particle bed. Furthermore, a bed of solid debris may also form in the RPV lower plenum as core melt migrates through the water pool in the lower plenum.

If the coolability of a debris bed in the core region or in the lower head is not possible, the re-melting of the debris occurs and a molten pool will be formed. The solid debris in the lower head may reduce the amount of liquid core melt that would be released from the RPV at vessel failure.

Oxidation and hydrogen generation

According to [Van Dorsselaere et al, 2006] the injection of water into a fuel bundle with minor degradation can lead to strong heat-up of the bundle but not excessively strong release of hydrogen. Additionally, water injection to a severely damaged core having molten pools or particulate debris beds is not expected to result in rapid and strong hydrogen production due to the difficulty of water to access the metallic components deep in the debris. The highest oxidation and hydrogen generation rates can be expected if water is injected into a moderately damaged fuel bundle characterised as housing molten mixtures that contain fuel (U-Zr-O).

In-vessel particle bed

The TMI-2 accident suggests that significant amounts of particulate debris may be formed during degraded core reflooding. The post-accident examinations revealed that the particle size in TMI-2 beds varied from 0.01 mm to 10 mm. The particulate debris obtained in the LOFT-FP-2 test ranged from 1 mm to 2 mm in diameter. The Phebus-CSD B9R test yielded particles of 0.02-0.15 mm in diameter. Besides the particle size, the porosity of the formed bed is another important parameter for coolability. The general understanding is that homogeneous beds with large particle diameters are favourable for coolability. Beds with smaller particles or stratified beds with fine particles sitting over coarse particles are much more difficult to cool. The access of water to the bed from bottom enhances the coolability, but it cannot be concluded that water always has access from below. In TMI-2 accident the particle bed in the core region was lying atop a solid non-porous crust. General correlations for coolability of debris beds can be found in [Alsmeyer, H. (Editor) Proceedings of the OECD Workshop on Ex-Vessel Debris Coolability Karlsruhe, 15-18 November 1999 FZKA 6475, Mai 2000].

Reactor coolant system pressurisation

The reactor coolant system may experience pressurisation caused by steam and gas (e.g., H_2) release during reflooding. The LOFT-FP-2 experiment and TMI-2 accident suggest that the pressure may increase by 20-40 bar. In a PWR the pressure increase may induce a break in the loops (e.g. in the hot leg or steam generator tube rupture, contributing to containment bypass scenarios). The pressurisation in the RPV lower plenum may contribute to melt ejection from high pressure and increase the risk of direct containment heating (DCH).

In-vessel steam explosion

The reflooding may influence potential in-vessel fuel-coolant interactions (FCI) if molten material drains from core region to the water pool in the lower plenum which is being replenished by the reflooding process. This may jeopardise the integrity of the lower head and thus render in-vessel melt retention obsolete. The current general understanding is that the possibility of RPV rupture by steam explosion is weak if the vessel is intact [D.Magallon et al., 2005] However in the case of reflooding the situation is specific and should explicitly be considered. The effects of rapid pressure increase and/or thermal shock induced by cold ECCS could seriously damage a RPV lower head that may have been exposed to elevated temperatures for a significant period of time.

Source Term

The experimental data from the LOFT-FP-2 test suggest that a significant amount of fission products may be released during the reflooding. In particular, barium, tellurium and strontium releases to the reactor coolant system may be enhanced. The main mechanisms for augmented releases are the formation of micro-cracks in the fuel as well as loss of cladding leaktightness by thermal effects. Increasing fuel temperature caused by exothermic reactions increases fission product migration in the fuel matrix.

In addition, temperature escalation and pressure spikes in the RPV and the loops may induce revaporation/resuspension of settled fission products.

4.3.5.2 Status of knowledge and main uncertainties

The presently available integral accident analysis codes allow the calculation of accident sequences including core reflooding. Although the quality of their results has to be considered as doubtful with regard to the reflooding issue – see the following sections – at present there is no alternative means of assessment available. Therefore, the results of such codes may be considered as best estimate, but a considerable uncertainty has to be attributed to these results according to the discussions in the following sections.

Hering & Hohmann [103 ; W. Hering, Ch. Homann (Karlsruhe Institute of Technology), „Degraded core reflood: Present understanding and impact on LWRs“, Nuclear Engineering and Design, Volume 237, Issue 24, December 2007, Pages 2315-2321] have summarised the present understanding of degraded core reflooding. Their conclusion is that the loss of rod integrity is currently determined by the temperature and cladding oxide thickness criteria. The applicability of these criteria for thermal shock induced failures has not been assessed. There is also lack of experimental data for production of reliable specific models. The models for fuel relocation and formation of particulate debris are currently very parametric, since the knowledge on the phenomena and related mechanisms is still limited. This necessitates the performance of sensitivity studies to obtain an envelope for possible overall effects of reflooding.

The hydrogen production kinetics during reflooding of moderately or severely damaged core still has significant uncertainties. The uncertainties in hydrogen prediction are also affected by core degradation, coolant transport and fuel coolant interactions. These uncertainties contribute to the risk via direct connection to hydrogen distribution and combustion phenomena in the containment.

Re-pressurisation of the reactor coolant system during reflooding is not precisely known and there is also a lack of experimental data. The occurrence and energetics of in-vessel fuel-coolant interactions in terms of RPV lower head failure or core melt discharge from an elevated pressure have significant modelling uncertainties but risk for RPV rupture should only be considered if the RPV is not intact.

The release of volatile or semi-volatile fission products from the fuel during reflooding is not well-known. The role of ruthenium release and chemical form during core-melt accidents with or without reflooding at shutdown states is still relatively poorly known. The applicability of models of revolatilisation of deposited fission products at reflooding conditions is still uncertain.

The coolability of in-vessel particulate debris beds is still difficult to estimate due to a lack of sufficient data on the particle sizes and the bed porosity for in-vessel conditions. Furthermore, the bed may include non-fragmented cakes and agglomerated particles. These uncertainties are further complicated by the 3D nature of a large debris bed. Thus, caution should be taken when crediting debris bed coolability in PSA L2.

4.3.5.3 Assessment of core reflooding with means of integral codes

A recent TMI-2 related benchmark calculation organised by OECD/NEA [TMI-2, 2009] provides a good reference for estimation of code capabilities for assessment of reflooding consequences. The integral codes MAAP4, MELCOR 1.8.6 and ASTEC V1 were exercised for the reflooding phase for an alternative TMI-2 accident scenario (not the real accident).

The increase of primary system pressure during reflooding was estimated and there was a 38% discrepancy between the 3 codes.

The MAAP4 and MELCOR codes, as well as one of the two ASTEC calculations, predicted a fast quenching at all elevations of the core while the second ASTEC calculation predicted non-coolable conditions in the middle of the core. This reveals that all the codes have significant deficiencies in their current coolability modelling.

All the codes significantly under-predicted hydrogen production during reflooding. The largest calculated H_2 generation was 29 kg (obtained with ASTEC), which seems much too low with respect to expectations based on the real TMI-2. The exercise reveals that all integral codes resulted in highly non-conservative estimates for hydrogen generation during reflooding for this sequence. Further code development is needed to give reliable estimates of H_2 production during reflooding. In the meantime, uncertainty studies with zirconium oxidation fraction ranging between 50 and 100 % (see ch. 4.3.1.2,Table 18) are recommended to be used for the estimation of H_2 generation during reflooding.

All three codes yielded a similar result that the end state of the core after reflooding was similar to the state before the reflooding. According to the calculations of the MAAP4, MELCOR and ASTEC codes, no significant additional core degradation took place during the reflooding phase of the alternative TMI-2 scenario. However, although this conclusion seems unrealistic, it cannot be verified since there are no reliable data of the core geometry and state during the TMI-2 accident. Further, the calculations for QUENCH tests suggest that the codes under-predict the additional core damage during reflooding, but the QUENCH tests are not fully representative of a real plant core.

4.3.5.4 Complementary considerations and quantitative recommendations for core reflooding issues

Hering et al. (2007) have presented a reflooding map illustrating the completeness of the current empirical data base to answer the question of coolability in terms of initial core damage state and reflooding mass flow rate (Fig. 27). A green colour in a square denotes that reflooding did not cause serious core damage. The yellow colour indicates that serious further damage took place following reflooding. A trend can be recognised that the necessary reflooding rate is increasing with increasing initial damage state of fuel.

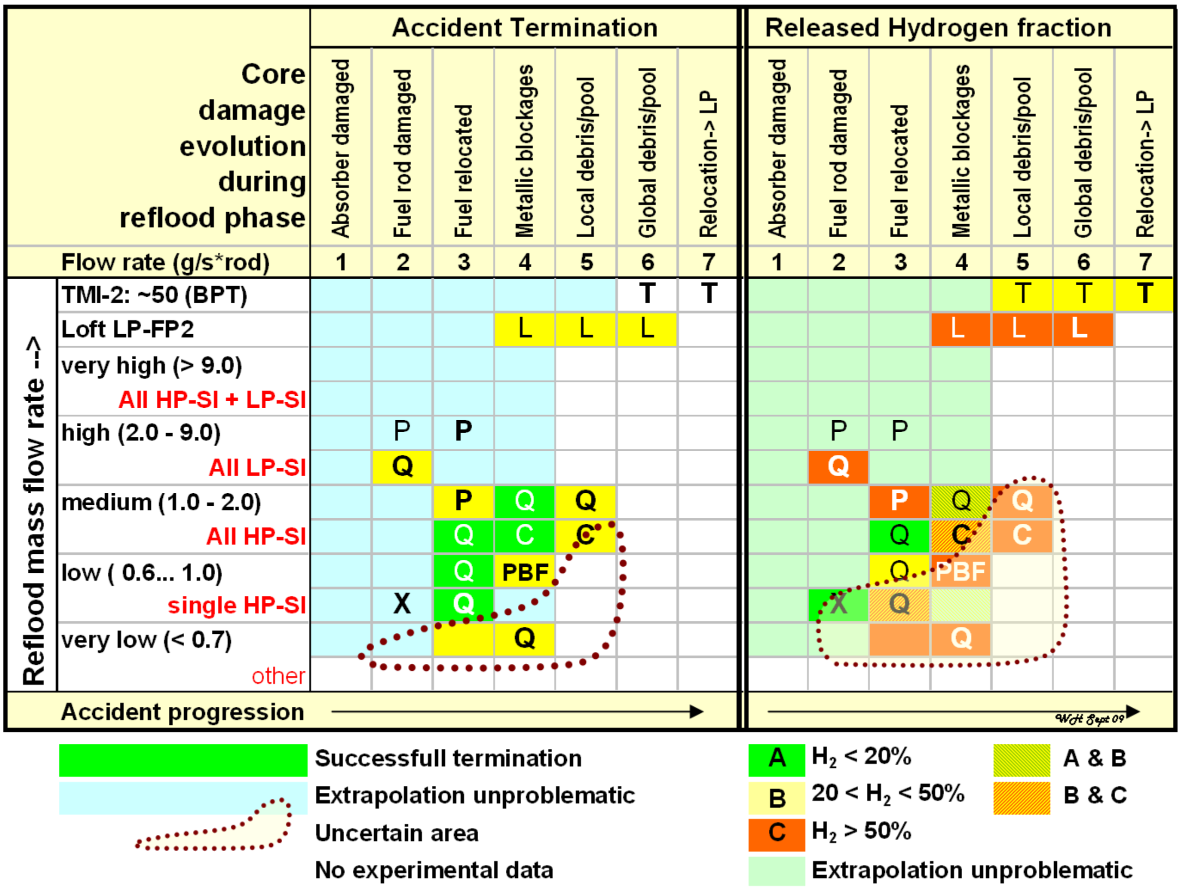

Fig. 27 Reflooding map by Hering et al. (2009). Letters denote the measured data source as follows: C=CORA, P=PARAMETER, Q=QUENCH, T=TMI-2, X=CODEX, L=LOFT-FP2, PBF=SFD-ST at Power Burst Facility.

The initial damage state of the core is the most important parameter. According to [Hering and Homann, 2007] the first criterion is the peak cladding temperature. The next criteria are the rate of core temperature escalation and steam availability. Steam starvation is the most dominant parameter for restricting hydrogen production.

The reflooding mass flow rate is another important parameter. It has been observed in the experiments that low reflooding mass flow rates lead to adverse effects if nearly all evaporated water is consumed by Zircaloy oxidation and nearly pure hydrogen is released into the containment. This situation can be expected in accidents where ECCS pumps cease operation or reflooding is performed (as severe accident management measure) with other smaller capacity systems originally not designed for emergency core cooling. The test QUENCH-11 suggests that reflood rate of water at 0.6 g/s per rod, corresponding to 30 kg/s of water for 860 MWe BWR or full capacity of a single High Pressure Injection Line, would not be sufficient to cool the core but rather lead to formation of a large molten corium pool in the core region. If the initial core damage state becomes higher, the required reflooding mass flow rate increases. An additional drawback of small capacity water injection is that the mass release to the containment would be pure hydrogen with little steam component. This in turn might exclude any benefit from steam inerted containment conditions relevant to hydrogen combustion.

According to Hering & Homann [2007] it can be deduced that when core temperature is below approximately 2200 K the risks of massive hydrogen release and accelerated core damage progression can be excluded if sufficient reflooding capacity is available. According to current understanding a reflooding rate above 1 g-water per second per fuel rod would be needed. From a practical point of view, the core temperature setpoint 2200 K corresponds to a core damage state where local molten pools have been formed in the original core region due to e.g. blockage formation, or significant amount of solid debris housed in the core region starts to melt. At core temperatures above 2200 K it is difficult to evaluate effects of reflooding. The outcome of initiation of reflooding above 2200 K depends on various parameters such as core configuration and size, water injection mass flow rate, water entrance location (bottom/top or both), RCS pressure, fuel type and possibly burn-up.

As for fission product release during reflooding, the integral codes do not have validated models to estimate the releases. More experimental investigations are needed to allow more accurate models to be developed. The release of ruthenium in oxygen-rich shutdown conditions needs further investigation.

Table 24 Recommendations for reflooding of degraded core.

Issue	Recommendations for application in PSA
Coolability of core by reflooding	Successful reflooding can be assumed if it occurs in the range of light blue area of the mapping 1 of Fig. 27. In practice, this is prior to formation of melt pool in the core region and with available reflooding capacity equalling at least 1 g/s/rod. It is conservative to assume that reflooding is <u>not</u> successful at a later stage if reflooding has not been successful earlier when core was in the original (most easily coolable) geometry.
Hydrogen production during reflood and quenching	Reliable modelling of this issue is not state of the art, and in a PSA it is not easy to simulate the many possible reflood scenarios. Nevertheless, this point should be treated in L2PSA in relation with the modelling of SAMG. The hydrogen source due to reflooding might be defined as follows: Between 50% (for fast reflooding) and 100% (for slow reflooding) of that metallic zirconium will be oxidised which is located above the initial water level. Associated hydrogen generation rate shall be taken into account when assessing containment threats. Hydrogen generation during reflooding should be assessed with state-of-the art integral accident analysis codes, but a considerable uncertainty range (a factor of five for short term peak flow rates and a factor of two for the total generation during the reflood should be assumed.)
Fission product release in reflooding situations	Fission product release from fuel during reflooding can generally be calculated with state-of-the-art integral codes. In case of highly oxidising conditions (i.e. air ingression to the RPV) the models for release of ruthenium need further research/code development. Furthermore, high vapour flow rates during reflood may cause resuspension of deposited volatile fission products from the surfaces of RPV upper internal structures. This should be considered particularly for containment by-pass scenarios with transportation of fission products with liquid (uncertainty analyses are recommended). Current integral codes do not have sufficiently validated resuspension models.
Long-term aspects of reflooding and sustained degraded core cooling	Maintaining of degraded core cooling depends on the successful operation of ECCS in recirculation mode. Success in long-term water recirculation is dependent on availability of heat removal systems to cool the recirculating sump water to the operating temperature range of ECCS pumps and on successful circulation of water from the containment sump to the RPV (avoiding the sump clogging also in the presence of corrosion products which may evolve in the long term.). In case long-term water injection fails, corium heat-up and melting may recommence.

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4.3.5.6 Modelling example - Loviisa NPP, L2PSA

Loviisa NPP, 2 units with VVER-440 reactor and ice condenser containment, is owned and operated by Fortum Power and Heat Ltd., and located in south-eastern coast of Finland. The power plant has gone through extensive modernisation and SAM strategy has been implemented to the plant as an application of integrated ROAAM (Risk Oriented Accident Analysis Methodology) [Siltanen et al., 2005]. SAM strategy is based on dedicated systems and in-vessel melt retention (IVMR) is cornerstone of the strategy. IVMR has been reliably resolved for Loviisa [Kymäläinen et al., 1997] and due to this the ex-vessel phenomena can be screened out. However, ice-condenser containment has rather low estimated failure pressure (1.7 bar^{abs}) and the threat of hydrogen has been one of the key issues carefully studied [Lundström, 1996].

Reflooding in Loviisa L2PSA

The Loviisa L2PSA addresses the issue of core reflooding. Core reflooding was studied in Loviisa L2PSA when the Loviisa operating license was renewed (L2PSA in 2006). The reflooding was included in containment event tree modelling and studied to find out if there is significant risk involved in hydrogen management or RPV external cooling (IVMR). In the Loviisa NPP SAMG the injection of water to the core is always suggested when there is water available, however some recommendations have been given. At the early stages of severe accident (when core degrades) it is suggested to inject as much water as possible to ensure core quenching. The possible peak in hydrogen release rate has been taken into account in hydrogen management by installing hydrogen igniters in the lower compartment (near the expected hydrogen source). In addition to igniters, the hydrogen management is based on efficient containment mixing and the use of passive autocatalytic recombiners. More information about Loviisa hydrogen management can be found below or from references.

If the severe accident has progressed further and the corium pool has formed on the RPV lower head, it is suggested to start water injection carefully (with a limited amount of water) to avoid possible stratified steam explosions and pressurisation of the primary circuit. Stratified steam explosion is assessed to be possible when water and the molten metal layer on top of a corium pool come into contact. In the situation where the RPV structure has been affected by molten corium for several hours, steam explosion might jeopardise RPV integrity. When water injection is started carefully, a solid crust will be formed on top of metal pool and steam explosion risk can be avoided. In L2PSA the operator error involved in reflooding during a later stage of the accident is modelled. If operator fails to start core cooling carefully, the risk of RPV failure will be higher.

For the hydrogen management, the core reflooding is separately studied for each accident class arising from L1PSA with different core degradation timing. Timing of core degradation is based on MELCOR analysis performed for Loviisa NPP. For each accident class the critical time window for hydrogen management will be defined. Basically this is the time period where reflooding of the core might cause very high hydrogen release rate. The recovery of emergency core cooling systems (ECCS) is modelled with recovery time distribution. Some of the ECCS faults can be recovered relatively quickly and easily and data is available for recovery time (i.e. loss of electricity). However, recovery of some of the faults might take considerable time and expert judgement may be needed for recovery time evaluations. Based on the available information a distribution for ECCS recovery has been formed. The recovery distribution and critical time windows are studied together to define the probability of reflooding during critical time window. This is done separately for each accident class. For the sequences where reflooding is successful the reliability of hydrogen igniting system is studied to define possible risk increase for hydrogen management.

It should be noted that reflooding is only studied in Loviisa L2PSA to define *additional* risk. Recovery is only taken into account in situations where it would cause additional risk for containment integrity. L2PSA does not take credit of ECCS recovery. When reflooding was studied with separate questions in containment event tree it was discovered that the additional risk is not very high for Loviisa. In more recent updates of the Loviisa L2PSA, the modelling is included in the fault tree level. In shutdown states the reflooding is not modelled. The sequences during shutdown states are slower and there is more time available for ECCS recovery actions prior core damage than during power operation states. It is assessed to be very unlikely that recovery, which has not succeed prior core damage (and prevent the core damage), would succeed in a very narrow time window which is critical for hydrogen management in shutdown.

Additional information of hydrogen management strategy of Loviisa NPP

In Finland the regulator, Radiation and Nuclear Safety Authority (STUK), requires that when hydrogen management is analysed, it has to be assumed that 100 % of easily oxidising core materials will react with water and hydrogen will be formed based on this assumption. In case of Loviisa NPP, this means that altogether the amount of hydrogen formed during severe accident is 800 kg. This has been the dimensioning criteria for Loviisa hydrogen management, which is based on efficient mixing of containment, passive autocatalytic recombiners and hydrogen igniters.

Different hydrogen release scenarios have been calculated as part of Loviisa hydrogen management strategy. In typical hydrogen release scenarios the hydrogen release rate is found to be around 0.1 kg/s. However, for Loviisa the possibility of significantly higher release rates (0.5-1 kg/s) of hydrogen has been taken into account by assuming that core cooling recovery might be done in a certain time period where hydrogen release rate will be increased. Time windows, in which this could be possible, were studied [Lundström 1996]. Recombiners can manage all the typical hydrogen release scenarios, igniters are required only in the cases with possibility of very high hydrogen release rate, namely in core reflooding situations.

The reliability of hydrogen management is studied in L2PSA. Due to deterministic criteria and requirements used in hydrogen management strategy development, hydrogen management through recombiners will succeed in all situations where basic requirements for successful hydrogen management are fulfilled. This means that in the cases where containment atmosphere will be efficiently mixed by forcing open ice-condenser doors (containment mixing was studied as part of hydrogen management strategy analytically and experimentally) and hydrogen release rate is typical (around 0.1 kg/s), the hydrogen management will be successful by passive autocatalytic recombiners. In these scenarios the only thing that will be studied through L2PSA is possible failure of ice condenser doors opening and possible unavailability of passive recombiners (only possible during outage). The reliability of igniters is additionally studied in cases where core reflooding is assumed to succeed in a certain time window.

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4.3.6 Containment atmosphere composition and containment pressurisation

4.3.6.1 Description of accident phenomena

Analysis of the composition of the containment atmosphere during the in-vessel phase of a SA is of importance to evaluate the potential for hydrogen combustion (see section 4.3.8) and the conditions for FP deposition (aerosols, dissolved species and vapour condensation) as a basis for the L2PSA source term calculations.

Information about the containment pressure is needed to assess the risk for containment failure due to quasi-static overpressure and to calculate the pressure peak/transient resulting from potential hydrogen combustion events.

In unfavourable circumstances (e.g. failure of condensation system in a BWR, or some particular sequences with RCS water injection and failure of CHRS), pressure may reach a value at which the containment venting system needs to be used but this is unlikely during the in-vessel phase. Production of steam and incondensable gases and evolution of containment atmosphere composition and long-term pressurisation in ex-vessel phase are described in section 4.5.5.

4.3.6.2 Factors influencing the containment atmosphere and pressurisation

During the in-vessel phase of a SA, mainly steam and hydrogen are released into the containment atmosphere (initially air or inert gas). The rate at which steam is released into the containment and its total amount depends on a variety of factors such as the leak size in the case of a LOCA or the availability of the ECCS. Condensation in wet wells or on structures in the containment as well as from the operation of spray systems can reduce the containment steam content. In BWRs the condensation pool is the key element for condensing steam and limiting the pressure.

For a LBLOCA as an initiating event the maximum containment pressure in the in-vessel phase is expected during the blow-down period when a large amount of mass and energy is released. The design pressure will not be reached as long as the break size is smaller than that of the design basis (for most reactor designs this is the very large double-ended guillotine break size of the main coolant line). This break size has a very low probability to occur and generally does not contribute significantly to PSA results.

For other initiating events, depending on the scenario the containment pressure develops very differently from a LBLOCA, but is almost certain to remain below the design pressure in the in-vessel phase (ignoring potential hydrogen combustion).

Hydrogen production is discussed in section 4.3.1. When hydrogen enters the containment via a leak/break or an open valve in the RCS its amount may be reduced by combustion or by mitigating measures like recombiners or igniters.

4.3.6.3 Distribution of hydrogen and other gases in the containment

The composition of the atmosphere in different parts of the containment can vary significantly. Sources (e.g. a leak in the RCS) and sinks (e.g. structures where steam condenses, or recombiners) of steam and hydrogen are localised. The initial containment temperature is also important with respect to the condensation potential.

Flow paths inside the containment will enable convection, providing connection from gas/steam sources to gas/steam sinks and enabling the equalisation of the atmospheric conditions. The formation of additional flow paths beyond those existing in the normal operational state between different compartments through doors, flaps or rupture discs depends on the respective pressure differences which develop during the accident evolution. Nominal leakages at closing elements like doors or removable concrete walls and ceiling elements can be important when determining the pressure differences attained in compartments and thus reaching the rupture/opening pressure difference of doors or flaps. This, together with the degree of compartmentalisation, influences whether natural convection can lead to a relatively fast homogenisation of the atmosphere. Otherwise inhomogeneous situations with stratification or large local variations of concentrations due to strong compartment separation can occur.

A further possibility for an inhomogeneous distribution develops from plume formation in the dome. A rising plume escaping from the hydrogen-enriched atmosphere in the equipment compartments (break location) to the dome region can lead to a hydrogen-enriched cloud in the dome for some time with higher potential for ignition. Depending on the hydrogen release rates and the efficiency of mitigation measures, the gas mixtures in the containment atmosphere eventually could exceed locally or even globally the flammability limit.

Without hydrogen mitigation measures, the gas mixtures might even exceed the detonation limit for some plant designs and accident sequences.
The distribution and concentration of hydrogen in the containment building can also be influenced by the containment spray systems. Spraying homogenises the distribution of hydrogen in the containment, but leads to “de-inertisation” of the mixture through the condensation of steam on water droplets and might change rapidly non-combustible mixtures to flammable ones.

BWRs

Particular issues arise for BWRs where the containment is divided in drywell and wetwell. Depending on the type of initiating event, steam and gas distribution may be very different in these two compartments.

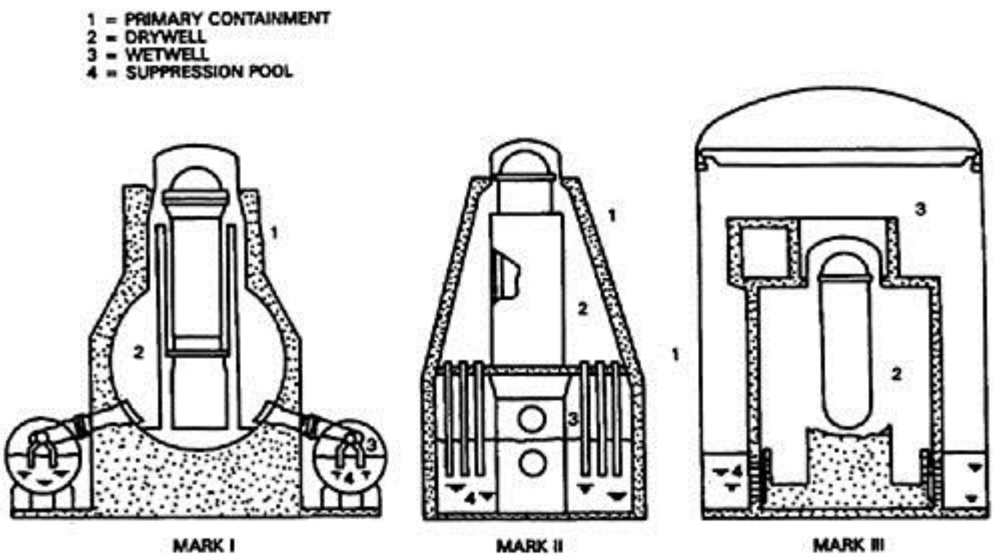

Fig. 28 BWR-GE Containments Types

Many BWRs, notably the General Electric Mark I and II designs (Fig. 28) and similar designs by other vendors, include protection of their drywell against hydrogen explosions by permanent inertization with nitrogen. It is similar for the wetwell of the Mark I BWR (torus containment). For the BWR Mark III (Fig. 28) , there is a rather continuous flow of hydrogen, air and steam from the drywell to the wetwell (suppression pool). The steam condenses in the suppression pool, whereas air and hydrogen escape from the surface of the pool. This is a burnable mixture which – if not ignited – may build up a dangerous concentration of H2 in dry air. Hence, the Mark III has igniters at the surface of the pool, which generates standing flames. Some risks exists here with the very high temperature in the immediate surroundings of that area. Thermal loads associated to the standing flames can cause damage to cables and components (instrumentation, pumps, valves, seals of air locks) [125].

Hydrogen is an extremely volatile substance. As no containment is fully leak tight, it will leak to the surrounding areas, which often have the function of secondary containment. If the containment is bypassed (interfacing system LOCA or steam generator tube rupture) or damaged otherwise, even more H2 can escape to the surrounding building. Hence, there is a certain risk that combustion may occur outside the primary containment. This may lead to combustion loads exerted on the containment from outside. Usually, containments have considerable margin against loads from inside, as they are in principle designed to carry the pressure loads from a large break LOCA. The pressure bearing capability for loads from outside can be substantially less: many primary containments are vulnerable to subatmospheric pressure or, in more general terms, to negative pressures differences with the outside. Hence, it is necessary to analyse such external loads carefully (vaccum breakers may be implemented).

Apart from the damage that the hydrogen combustion may cause to the primary containment, it may also damage the secondary containment and the equipment inside. This can include important systems which are located there, e.g. the emergency core cooling system (ECCS).The Fukushima accident has shown the importance of these issues for Mark I BWRs (huge damage to the secondary building of reactor 1, 2 and 3 due to hydrogen release from wetwell/drywell).
If pressure is relieved through the vent, steam may condense in the long lines or in the filter, and dry hydrogen may appear in vent appurtenances. Consequently, explosions are well possible and may lead to damage of the pipes and also of the filter itself and, hence, to an unfiltered release of radioactive substances.

All possibles pathes for hydrogen have to be identified and then the hydrogen distribution in the secondary building must be calculated. Possibility for hydrogen combustion, flame acceleration and consequences for the plant must be assessed.

4.3.6.4 Impact of recombiners/igniters on atmosphere

Hydrogen mitigation measures can also modify the distribution of gases. Recombiners should have little influence on global convection because they induce only small flow rates. Nevertheless, there are ongoing R&D activities to characterise the impact of recombiners on gas stratification in containment, for example in the OECD SETH2 program (MISTRA and PANDA experiments) [124]. Their major global effect may be the atmosphere heating and the transformation of non-condensable oxygen and hydrogen into steam, whose condensation could slightly reduce the pressure in the long term compared to a situation without recombiners. Further, depending on the recombiners design, they may produce ignition sources if the hydrogen volume fraction is high enough. By triggering combustions, igniters or - inadvertently – recombiners can have a strong influence on the composition of the atmosphere and the distribution of gases (rapid movement of the flame front and displacement of gases). This applies, of course, to all combustions.

4.3.6.5 Relation between atmosphere composition and FP retention

Influence on chemistry in general

FP chemistry depends, among other factors (temperature, pressure, humidity, dose rate…) on reducing (excess of hydrogen) or oxidising (excess of oxygen) atmosphere properties. The chemical species carrying radionuclides are transported to the containment building either as gases or as condensed compounds in form of aerosols, and can react with each other or primary system materials. Retention of FPs, both as reactive vapours and aerosol particles, strongly depends on their physical state and properties (and on their chemical form), but large uncertainties still exist regarding the formation of compounds that can be transported to the containment.

The gas temperature in the containment is not expected to exceed 130-140 °C (403-413 K) prior to VF in most accident sequences. At this quite low temperature, the chemical speciation of gaseous FPs that occurs in the core region and during the transport through the primary circuit is likely to be almost completed. Nevertheless, due to radiative environment some evolution of the chemical speciation may have to be considered: for example, for large dry PWR containment, the worst chemical form of iodine (ICH3) is formed in the containment by reaction between gaseous molecular iodine and paint.

Influence on iodine chemistry

Iodine chemistry in the containment is an important issue affecting the source term uncertainties in accident sequences where only small or filtered flow paths into the environment exist. The formation and the stability of the compounds of this element are conditioned both in the aqueous phase (the sump pH is an important parameter in determining the formation of volatile iodine in the aqueous phase) and in the gaseous phase (complexity of reactions, in particular radiolysis). The steam fraction in the containment atmosphere was found to have influence on the retention of airborne iodine onto the containment structures, as it is responsible for the presence of water on the surfaces [119]. A containment atmosphere permitting combustion processes can promote decomposition of iodides: some experiments showed that recombiners can favour the thermal decomposition of metal iodides in a humid atmosphere [117] but this effect has yet to be confirmed and quantified, in particular when analysing results of other recent representative experiments in the THAI German facility.

On the other hand, it was experimentally found that alkaline aerosols can be an effective sink for gaseous compounds of iodine, like HI and I_2 [121]. For this reason, most of the attention was focused on the iodine removal by aerosol deposition.

Influence on hygroscopic aerosol behaviour

The humidity of containment atmosphere appears as an important parameter due to its influence on growth and deposition of aerosol particles.

The bulk condensation of steam, as well as hygroscopicity of aerosol material, causes the growth of the suspended particles and improves their retention by turbulent inertial deposition and gravitational settling as these phenomena are more effective for large particles. It is usually assumed that steam condenses on particles at supersaturated conditions: however, the presence of water soluble substances, like CsOH and CsI, can initiate the water condensation even at superheated conditions. This effect was experimentally quantified and code benchmarks were set up to validate the aerosol models for this phenomenon (see Fig. 29) [118].

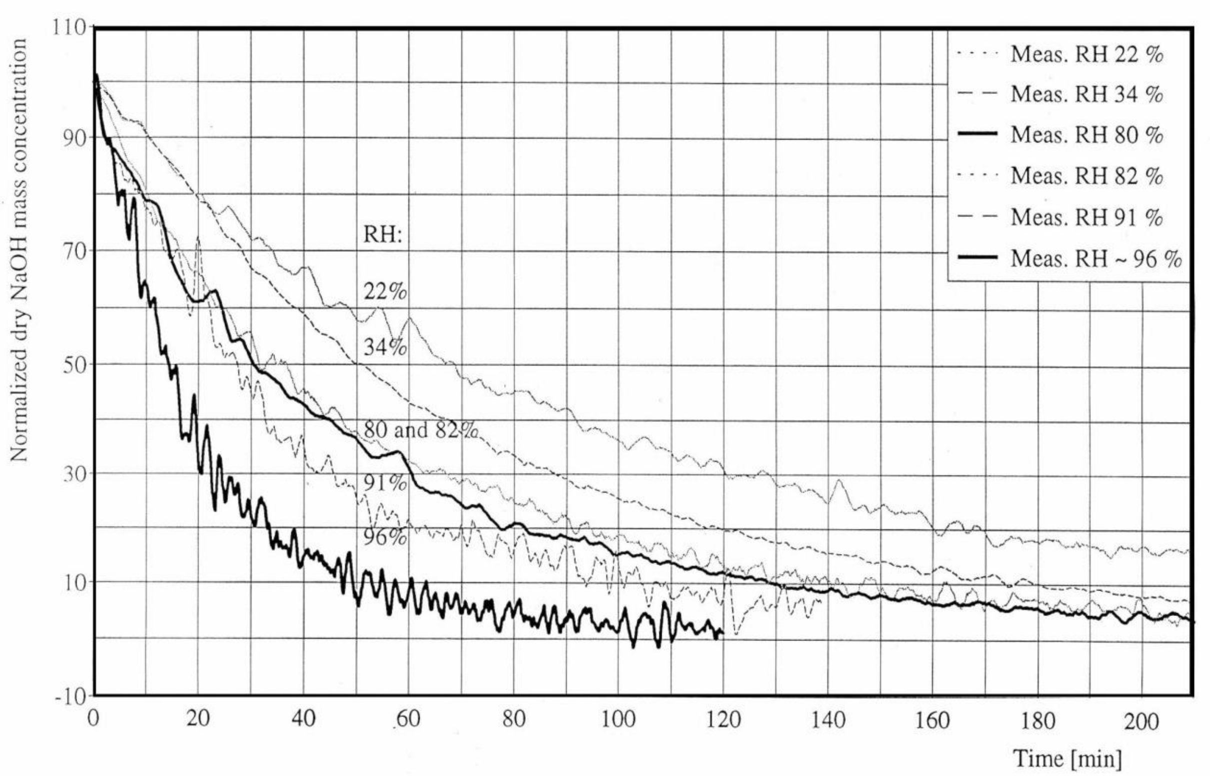

Fig. 29 Effect of aerosol hygroscopicity put in evidence with Finnish experimental facility AHMED, using NaOH as simulant [118] (RH = Relative Humidity).

<u>Influence on aerosol diffusiophoretic deposition onto the containment walls</u>

If steam is supersaturated at the containment wall temperature, a condensation flux toward the cold surface occurs. This gaseous flux is able to remove the particles suspended near that surface.

This important removal mechanism is known as diffusiophoresis. The condensing steam slowly washes out all particles involved in the flow, independent of their size.

The equation typically used to evaluate the deposition velocity associated to this process was deduced by Schmit and Waldmann for stagnant air [120]:

$$v_D = \frac{\sqrt{M_1}}{x_1\sqrt{M_1} + x_2\sqrt{M_2}} \cdot \frac{W}{N_m}$$

where x_1 and M_1 are respectively the molar fraction and molecular weight of steam, x_2 and M_2 are respectively the molar fraction and the mean molecular weight of the stagnant gas mixture, W is the molar flux of condensing steam (kmole/m^2/s), and N_m is the molar concentration of the mixture (kmole/m^3).

The flux W of steam condensing onto the walls is due to the thermal behaviour of the whole nuclear island and, then, of the accident sequence characteristics. A significant generation of non-condensable gases from clad oxidation can imply high containment pressurisation and, consequently, a reduction of the molar concentration N_m, with a reduction of the diffusiophoresis velocity.

Even if considered in the code simulations through the step-by-step adjustment of gas properties, other relationships between aerosol behaviour and containment atmosphere composition appear of minor importance.

4.3.6.6 Application to L2PSA - assessment of containment atmosphere using computer codes

It is clear from the previous subsections that an analysis of the composition of the containment atmosphere, the distribution of gases in the containment and the containment pressure needs to rely on calculations using validated accident simulation codes. Typically, for a L2PSA, integral codes such as MELCOR, ASTEC or MAAP are used. An appropriately detailed nodalisation of the containment is needed to adequately cover all important aspects like pressure difference-dependent opening of flow paths or the localised nature of sources and sinks of gases. If specific phenomena cannot be simulated with enough detail in these codes, the use of more specialised codes should be considered. Computational fluid dynamics codes like GASFLOW, TONUS or ANSYS CFX allow for a more detailed analysis.

These codes are deterministic, i.e. each and every input data and all their results represent only a single value out of a range which has to be assumed in reality. In principle, for a full scope L2PSA, a large set of different code runs with variations of input data – possibly in the framework of a Monte Carlo simulation – could provide a reliable basis for probabilistic assessments. However, limitations of resources will in most cases not allow this to be achieved. It is then recommended, as a minimum, to explore the boundary development of the containment atmosphere by eliminating the ignition in the code calculation. Such an analysis would show the potential threat for combustion if the combustion peak at each time step can be calculated "virtually" (the H_2 burnt is not removed from the containment).

Within the probabilistic framework of the L2PSA, conditional probabilities should then be assigned to the atmospheric conditions between the extremes of no combustion and combustion at the worst time for combustion (see section 4.3.6.1). Assumptions about the activity of ignition sources inside the containment will have to be part of this evaluation.

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4.3.7 Containment venting

4.3.7.1 Description of accident phenomena

Containment venting systems have been implemented on some NPPs to avoid containment failure in the case of an accident inducing a slow containment overpressurisation. Activation of the venting system allows the discharge of the gas from the containment to environment. Such systems are generally linked to a means of fission product filtration to limit as far as possible the radiological impact. Activation of a containment venting system would lead to a controlled (time), filtered release that would be preferred (regarding protection of population) to any uncontrolled, not filtered radioactive release in the case of a containment failure.

In PWRs with large robust containments, early venting in the in-vessel phase is generally not considered as a significant accident management issue, because the mass and energy release from the RCS is not sufficient to threaten containment integrity directly. However, containment venting during the in-vessel core degradation phase may be considered within strict limits. Possible examples:

- Some accident sequences with a long period before core degradation or vessel failure may also consider a failure of the CHRS. Steam inducing a pressure rise in the containment could be produced from in-vessel cooling by injection into the RCS (as well as from ex-vessel cooling by injection in the reactor cavity). Venting would remove energy from the containment and decrease the temperature and pressure, which might improve the likelihood of operating safety equipment. However, if failure of cooling of the water recirculated from the sump is not considered in the L2PSA analysis, containment venting during the in-vessel phase may not be addressed,
- Precautionary venting of the containment might be initiated to lower the baseline containment pressure, so that any subsequent energetic event in the containment e.g. hydrogen combustion or DCH, would start from a lower initial pressure, thus further reducing potential threats to the containment.

If such an action is part of the SAM strategy in a plant, the L2PSA shall take into account its potential benefits i.e. prevention of potential high pressure events and prevention of safety injection failure due to high sump water temperature, as well as its potential negative effect i.e. opening a path for radioactivity releases

In BWRs there may be two reasons for venting the containment in the in-vessel phase:

- If the pressure suppression system is not functional e.g. because there is a potential bypass from the drywell to the wetwell gas space, early and substantial venting of the containment may be needed to prevent overpressure failure of the containment,

- If the accident sequence has a long term failure of heat removal from the condensation pool of the pressure suppression system, this pool may start boiling, thus increasing the containment pressure. Gradual venting of the containment may be needed to prevent overpressure failure or to reduce containment pressure loadings from any subsequent processes e.g. from pressure increase due to hydrogen production.

In the APET, the following branches will typically have to be defined and their probability quantified:

1. The venting system depressurises the containment, and filters minimise the radioactive releases as required.
2. The venting system ddepressurises the containment, but filters do not sufficiently minimise the radioactive releases as required (e.g. due to filter failure).
3. The venting system does not depressurise the containment as required (e.g. due to missing human action).
4. A failure occurs during operation of the venting system which leads to a large unmitigated release (e.g. due to a hydrogen burn inside the venting system).

4.3.7.2 Issues to be addressed in L2PSA

The following issues have to be addressed with regard to containment venting in both the in-vessel and ex-vessel phases:

- If the venting system needs instrumentation, control or human action: Quantification of the probability that these necessary preconditions for successful operation of the system will not be met,
- Quantification of the probability that the capacity of the venting system is not adequate to fulfil its function with regard to protecting the containment under the conditions which prevail during this phase of the accident,
- Quantification of the probability that components of the venting system e.g. valves, piping, filters, will fail during operation, under the conditions which prevail during the accident,
- If the PSA aims at quantifying releases of radionuclides: Quantification of the radionuclide retention efficiency of the venting system, in particular if filters are applied, for the different chemical elements,
- Quantification of the probability of unintended venting in the in-vessel phase e.g. due to human errors or technical failures, and the consequences of such an event.

4.3.7.3 Initiation and control of the venting process

- Opening the venting system when <u>fast</u> action is required: If the venting system is needed to prevent containment pressurisation in the case of bypass of the condensation pool in a BWR the system must open very fast i.e. within seconds in case of a large LOCA. Therefore, such systems do not rely on human actions. They are simply initiated by opening of a rupture disk. In principle it is possible that the disk has not been designed or manufactured properly or that any valves downstream in the system are closed by mistake, so that the venting would not be initiated. Discussion and quantification of such failure probabilities should be part of a L2PSA,

- Opening the venting system when <u>slow</u> action is sufficient: If venting is applied to control gradual containment pressurisation, it may be initiated by simply opening a rupture disk (see above for potential failure probabilities), or by instrumentation and active action – either by automatic systems or by human intervention. In this case the reliability of the action is generally high because there is sufficient time available to perform the action. Standard techniques for determining component failure and human performance can be applied. To quantify the human actions, the specific conditions on the site should be taken into account. These may include time available for the actions to be performed, availability of tools and protective clothing, radiological conditions and accessibility of local controls, power supply, availability of containment pressure measurement and quality of the relevant EOP,
- Controlling and closing a venting system: The following reasons may exist which require closing a venting system in the in-vessel phase:
    - If the initial venting process did not proceed through a filter, as may be the case for large capacity fast acting systems in BWRs, the venting route has to be closed and redirected through a filter to reduce the later, potentially very large, radionuclide releases. The potential consequences of not closing an unfiltered venting system are so high that the closing process must have a very high reliability, including component reliability and human action where appropriate,
    - Filtered slow acting containment venting will gradually reduce the containment pressure and there will be a considerable loss of non condensable constituents, in particular nitrogen, from the containment. If the remaining gases and particularly steam condense later, under-pressure in the containment could develop and threaten the containment structural integrity. Therefore, the venting process has to be interrupted before containment pressure is unacceptably low. In general, this action is performed manually with a considerable margin of time available. Therefore, it can be analysed with standard approaches, and the success probability is expected to be high. The potential sub-atmospheric failure conditions could be assessed relatively easily for large steel structures. This analysis may be more difficult for concrete containments where liners or penetrations could constitute the critical structures.

4.3.7.4 <u>Pressure reducing capacity of the venting system</u>

The thermodynamic analyses for the in-vessel phase which determine the static containment loads are well advanced. Therefore, in a L2PSA it is recommended to apply the same state-of-the-art codes which are normally used for determining the containment atmosphere pressure and conditions. The results of such codes, including modelling the venting system, can be applied to determine whether the pressure reducing capacity of the venting system is sufficient.

Possible examples where the capacity may be insufficient include:

- Containment venting systems in some plants making use of pre-existing containment piping and penetrations which have not been designed for such a purpose. In this case the margin between the calculated containment pressure with the operation of the venting system and the containment load bearing capacity may be small,

- If containment venting is performed without a dedicated filtration system designed for SA conditions, the flow can be discharged in the annulus or in the nuclear auxiliary building before being released at the chimney. In that case, the availability of the internal and extraction ventilations of these rooms should be assessed, as well as the efficiency and loading capacity of the filters facing the FPs carried by the venting flow, and potential hydrogen threats,
- The potential production of steam or hydrogen gases has been increased, or the possible temperature rise in the containment during accident conditions has been increased e.g. due to reactor power increase, major changes in the fuel design or modifications in the containment systems or structures,
- If the spent fuel pool is located inside the containment (as in many PWRs), and if the heat removal from that pool is insufficient (which may have a significant probability in core melt accidents), steam production from that pool may contribute to the pressurisation, increasing the load for the venting system.

4.3.7.5 Failure of the venting system

Containment venting systems are comparatively simple systems. Therefore, the failure probability of the system, including unintended operation, can be assessed with standard techniques. It also has to be taken into account that some components e.g. valves and filters may never have been in operation before they are needed so that proper consideration of test intervals and the issue of unknown or unnoticed degradation of these components has to be taken into account.

The venting system has to perform its function under conditions which are far beyond normal plant operating conditions. Therefore, a key issue is to assess whether the boundary conditions which prevail before or during the venting process might exceed the system capabilities, leading to failures. In particular the following issues have to be addressed:

- Availability of electric power if it is required e.g. for instrumentation or valves,
- Threats inside the venting system, or outside the system where the venting flow is discharged, due to hydrogen combustion. A particular concern is the position downstream of condensing locations like scrubbers and condensation pools, where steam would be depleted, increasing the volume fractions of non-condensable gases like hydrogen. As an example from a PWR, the venting flow behind a venturi scrubber, containing considerable amounts of hydrogen (and carbon monoxide in the ex-vessel phase) is directed into the off-gas system of the plant. This creates a high risk of hydrogen combustion in this system with consequential structural failures,
- If the containment venting system is equipped with a pipe heating system (like French PWRs) to limit the hydrogen combustion risk, then the activation of the system can be required up to one day before the containment venting,
- If human action is foreseen, the human performance should be assessed i.e. time, information, accessibility,

- Filters in the venting system will be heavily loaded with heat producing aerosols and moisture. The likelihood and failure mode due to mechanical or thermal loads (e.g. hot discharged gases on the filter, or water loads caused by rapid condensation in the scrubbing solution) should be assessed.

Potential failures of the venting system are design specific and depend on the accident sequence. Therefore, no general rules can be given how to address these issues.

4.3.7.6 Filtering efficiency of the venting system

Containment venting systems which are designed to operate when the core is already degrading, or has degraded, must have filters to prevent widespread contamination of the environment. There are several filter designs, including venturi scrubbers, sand bed filters, high efficiency particle filters, charcoal filters.

If the L2PSA aims to quantify source terms into the environment, it is essential to take the venting filter efficiency into account in the analyses. When the venting system is designed, there are minimum retention requirements which have to be guaranteed by the filters. However, in practice the actual retention factors achieved may be better, but they may also be dependent on the filter loading, including humidity and the decay heat generation of FPs within the filter.

In general, the retention of FPs in aerosol form can be very good. But, in addition to noble gases, there are gaseous FPs which are not well controlled by particle filters e.g. specific chemical forms of iodine or of ruthenium. Filters specific to the retention of gaseous forms of iodine are made with charcoal, usually impregnated with tri-ethylene-di-amine (TEDA) to increase the trapping efficiency for organic iodide. In such filters the retention capacity depends on the humidity and on the temperature. If such filters are credited for influencing the radionuclide releases in a L2PSA, appropriate care must be taken on demonstrating these conditions and the related retention factors apply. If the long term behaviour of the plant is addressed in the L2PSA, the stability of the FPs in the filter should also be examined.

In practice, complex analysis of filter retention capacity including the related boundary conditions is beyond the scope of most L2PSAs. If such analysis is not feasible, an appropriate range of uncertainty should be assigned to the retention factors of the containment venting system. The requirement for this analysis, and the level of detail of the L2PSA analysis, may vary depending on the overall objectives (see volume 1, chapter 7).

4.3.8 Hydrogen combustion

4.3.8.1 Description of accident phenomena

In severe accidents the risk of losing containment integrity exists as a result of hydrogen combustion. The hydrogen is mainly produced by oxidation of cladding zirconium and fuel element structures during the core degradation phase and by oxidation of metals present in the corium pool or in the basemat during the molten corium-concrete interaction phase (see section 4.5.4). This hydrogen is released into the containment atmosphere (see sections 4.3.6).

Studies of representative accident sequences indicate that despite the installation of mitigating features such as recombiners, it is difficult to prevent, at all times and locations, the formation of a combustible mixture potentially leading to combustion and to local flame acceleration. The RUT project [126] produced criteria which allow determining whether flame acceleration and a deflagration-to-detonation transition (DDT) are possible. The outcomes of the ENACCEF project [127], [128] will further develop these criteria.

Within the SARNET project an attempt was made to propose harmonised methods to assess hydrogen combustion. Reference document [131] introduced several recommendations.

4.3.8.2 Flammability

The flammability of the containment gas mixture depends on its composition, temperature and pressure as well as the ignition mode. However, in practice, the point representing the mixture composition (hydrogen, air, steam) on the Shapiro diagram (Fig. 30) is used to determine whether the mixture is flammable. In this diagram, the zones of ignition and detonation are limited by the exterior and interior curves respectively. These limits are dependent on temperature and pressure but in practice these dependencies are often neglected, e.g. when determining the flammability of an atmosphere with an integral code.

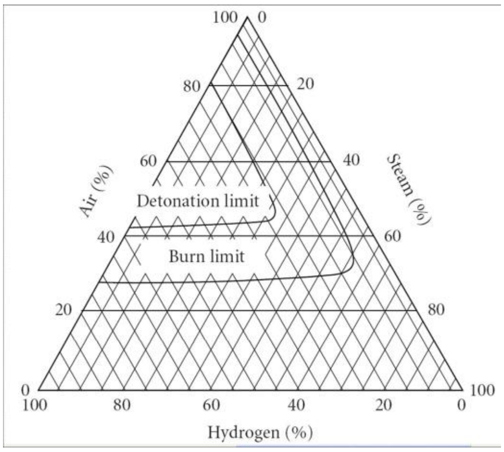

Fig. 30 : The Shapiro diagram (from [129])

The German guidelines on PSA ([132] section 7.5.4.3) provide a simplified necessary condition for estimating whether a gas mixture is combustible:

$$H_2O < 100 - 37{,}3 * \exp(-0{,}007{*}H_2) - 518 * \exp(-0{,}488{*}H_2)$$

H_2O and H_2 in the equation are percentiles of volume fractions for steam and hydrogen.

The previous considerations on flammability are incomplete because the detonation limit is not only a characteristic of the gas mixture but it also depends on the geometry.

The potential effect of ESFs should also be considered, For example, sprays have a de-inerting effect due to condensation of steam on water droplets. This could rapidly change a non-combustible mixture into a combustible one.

4.3.8.3 Ignition sources and ignition probabilities

In a flammable mixture, combustion may be initiated by an energy source of a few mJ. Consequently, in the presence of an ignition source such as electrical power sources, actuator switching or hot spots developing, it appears probable that ignition would occur if the atmosphere is in the combustion zone of the Shapiro diagram. In contrast, a much more powerful energy source (at least 100 kJ) is required to directly trigger a stable detonation. This explains why direct detonation can be ruled out for practical purposes; the only mechanism considered likely to provoke detonation is flame acceleration and the deflagration-to-detonation transition. In fact, under the effect of hydrodynamic instabilities and turbulence (caused primarily by obstacles in the flame's path), an initially laminar deflagration (with a flame velocity around 1 m/s) may accelerate. Fast combustion regimes may also develop, involving rapid deflagration (a few hundred m/s), deflagration-to-detonation transition (DDT) and detonation (over 1000 m/s). These explosive phenomena pose the biggest threat to the mechanical integrity of the containment because they can produce locally very large dynamic loads.

It must be distinguished whether an ignition source is steady (e.g. hot structures) or not (e.g. electric sparks or flame propagation from other rooms). For example, if the hydrogen concentration in a room is increasing, a steady ignition source will lead to combustion as soon as the flammability limit is reached, generating rather moderate containment loads. Significantly higher concentrations and hence the possibility of higher containment loads will not be reached in the presence of steady ignition sources (assuming no temporary de-inertisation by steam condensation).

If the ignition source is not steady higher hydrogen concentrations can develop before the ignition occurs and more severe consequences can no longer be excluded. Within the SARNET project [131], a related recommendation is given as follows:

“The ignition time may be considered at the worst time – physically possible - for simplified assessment (combustion consequences assessed for the highest concentrations values), or as random (as part of uncertainties).” There are two possibilities for modelling the unintentional hydrogen burn ignition due to random sources in dynamic calculations. The ignition may be given a constant possibility during a selected period for a single ignition or the ignition may have a specific frequency. As the ignition source activation is more like a stochastic process, a constant frequency is not a very good choice, but an expectation value between two separate ignitions should be considered. Depending on values of the characteristic parameters (i.e. constant probability or specific frequency) very different ignition times and combustion processes can be obtained for otherwise identical boundary conditions of the containment atmosphere evolution.

Whatever principal approach is selected, the ignition source probability or frequency is very uncertain. A tentative example is given below, but it is recommended to assign a high degree of uncertainty to the assessment.

Due to their hot catalytic sheets, some PARs can be considered as ignition sources under specific conditions. In fact, some of the experimental tests performed on KALI H2 and H2PAR [130] show that PARs could ignite the flammable gas mixture. These experimental results show that ignition induced by the type of recombiners which have been tested occurs for low hydrogen concentrations leading to rather low pressure loads on containment.

The German guidelines on PSA ([132] section 7.5.4.3) provide the following statements on ignition:

- If electric power is not available and if the atmosphere is combustible (but not detonative) the ignition probability is between 0.0 and 1.0 (triangle distribution with maximum at 0.4),
- If electric power is available and if the atmosphere is combustible (but not detonative) the ignition probability is between 0.0 and 1.0 (triangle distribution with maximum at 0.75),
- If the atmosphere is detonative, there is a probability p between 0.8 and 1.0 (homogeneous distribution) for ignition soon after reaching the detonative regime. With complementary probability (1-p) there is an ignition late after reaching the detonative regime (which in principle may lead to combustion far into the detonative regime).

These data take into account the ignition properties of specific recombiners applied in German plants. Care should be applied when making use of these ignition probabilities for other types.

For NPP equipped with active igniter systems, L2PSA should include a specific analysis of the time of activation regarding the evolution of hydrogen concentration in the containment. The case of spurious igniter activation should be considered.

4.3.8.4 Criteria for flame acceleration and deflagration-to-detonation transition

Researchers have developed prerequisite criteria, i.e. conditions required for the various combustion modes. These criteria are a measure for the sensitivity of the gas mixture. There are two criteria which are widely used:

The σ criterion is related to flame acceleration. The σ quantity is the mixture's expansion factor, the ratio of fresh and burnt gas densities at constant pressure. It is an intrinsic property of the mixture in question. The critical value σ^* beyond which flame acceleration is possible depends on initial gas temperature and flame stability. It is based on the results of numerous experiments at various scales and in various geometries. For details, see e.g. [126].

Similarly, prerequisite conditions have been defined for characterising the transition between deflagration and detonation regimes (DDT). They are based on comparing a characteristic geometrical length *L* of an enclosure with the detonation cell width λ. *L* needs to be larger than 7λ for a possible detonation onset (in cubic geometry). Descriptions of how to obtain *L* (there is no unique definition) are given, e.g., in [126] where an interpolating formula for λ can also be found.

These criteria were defined for homogeneous gas mixtures, and understanding the effect of concentration gradients is one of the objectives of the ENACCEF programme [127], [128]. Nonetheless, these criteria, together with the analysis of hydrogen distribution in the containment building and the facility geometry, can be used to identify potentially dangerous situations (i.e. when combustion becomes possible; associated loads must be assessed). To apply these criteria, however, the codes used to calculate hydrogen distribution in the containment must be validated based on situations representative of severe accident conditions. This has been the aim of experimental programmes on hydrogen distribution in recent years.

Several analysis performed in frame of L2PSA [129] has shown that the use of recombiners reduces significantly the risk of flame acceleration and transition to detonation. However it is necessary to understand and evaluate the issue to be able to defend and quantify this reduction on a well founded basis.

4.3.8.5 Assessment of hydrogen combustion loads

A simple method for assessing the quasi-static pressure increase due to combustion is the AICC approach (adiabatic isochoric complete combustion). Factors which either mitigate the pure AICC result (e.g. due to incomplete combustion) or which increase the containment loads (e.g. due to dynamic effects in particular in case of heterogeneous hydrogen distribution) should be applied.

Correction factors from German analyses done by GRS can be found in [131], appendix 1.4 as follows:

- The temperature dependency of the AICC pressure increase is taken into account by multiplying the AICC pressure ratio (pressure after burn / pressure before burn) by the factor 323/TSB where TSB is the containment atmosphere temperature in K before the burn,
- A complete combustion will only occur at initial hydrogen volume fractions above 0.08 to 0.12 (uncertainty represented by homogeneous distribution). Between an initial hydrogen volume fraction of 0.04 and the just mentioned limit for complete combustion the fraction of the burnt hydrogen increases linearly from 0.0 to 1.0,
- The event tree analysis distinguishes three containment zones (not to be confused with the deterministic MELCOR analyses which consider much more nodes). The hydrogen distribution is necessarily inhomogeneous within each of these three rather big zones. A factor between 0.7 and 1.2 (homogeneous distribution) has been chosen to be applied to the above mentioned AICC pressure ratio to take into account inhomogeneous hydrogen distribution,
- For a typical PWR steel containment which has a resonance frequency of about 5Hz to 12Hz the effective static pressure increase of a detonation is approximately twice the AICC pressure increase. This effective static pressure would approximately generate the same containment load as the dynamic pressure history of the detonation. This latter factor of two may be too low, according to more recent analyses results.

The AICC method is applicable for a single volume. In a real containment there will be several connected rooms with different atmospheric conditions. Apart from the issue of flame propagation (which is addressed in the appropriate section) the question arises how to estimate the effective pressure when combustion occurs under such conditions. It may be of particular practical interest if a room with a high hydrogen fraction has a neighbouring room with no combustible atmosphere. Obviously the pressure in the room with combustion tends to be less in this case than without neighbouring room. But there seems to be no simple approach available which could modify the AICC method accordingly.

The AICC method may be appropriate for limited scope PSA, or if there is a considerable margin between containment load estimates and containment failure conditions.

However, if the failure margin is small, or if dynamic issues or phenomena associated with flame propagation in non-uniform atmospheres become important, more sophisticated methods are required. The current integral codes may be suitable as long as flame acceleration or DDT transition does not occur.

Enhanced and validated combustion models are needed to simulate flame propagation in non-uniform environments, especially those with hydrogen gradients enabling transition between flame regimes.

R&D efforts to date have already significantly enhanced understanding of the phenomena governing the distribution of gas mixtures and their potential combustion. In particular, establishing criteria based on experimental data has led to identification of potentially high-risk situations. Regarding computational tools, although they have clearly reached a degree of maturity, their predictive capability must be reinforced by enhanced modelling (multi-compartment calculations, for which the results depend heavily on user expertise, for example) and/or overcoming computing limitations in the case of multidimensional codes, which currently make it impossible to respect all the rules of proper use (e.g. grid refinement).

Since detailed analysis of fast combustion processes cannot be considered state of the art in L2PSA, the following recommendation is given for practical purposes. Each of the following steps could be represented by a branching point in the APET:

- Determine whether flame acceleration or DDT could occur anywhere inside the containment if no ignition source were active. For this purpose screen results of integral code calculations, and multiply maximum hydrogen fractions with appropriate peaking factors (see example above). Use the criteria for flame acceleration or DDT discussed in the pertinent section above,
- Investigate the probability that an ignition source would not be active before the containment atmosphere enters the DDT regime, and that it would become active in the DDT regime (see section on ignition sources),
- If flame acceleration or DDT is possible, identify the location where this event could occur. If it is, for example, in the reactor pit only, challenge to the containment could be negligible. If it is near to the containment wall, apply a multiplication factor to the appropriate AICC pressure in order to estimate the equivalent quasi-static load (the factor is at least 2, but may be higher, see section on dynamic containment loads versus static containment loads). Compare this pressure to the static containment failure pressure,
- Check whether missile generation is possible and whether impact of missile onto containment could challenge the containment. Assign containment failure probability due to missiles based on engineering judgement.

Of course this procedure does not meet perfect scientific standards and it should therefore apply wide uncertainty bands. But taking into account that flame acceleration and DDT is unlikely in most containments (at least if they are equipped with recombiners), this practical approach should be acceptable. In order to be pessimistic it could also be accepted to assume containment failure with probability one if flame acceleration or DDT occurs.

4.3.8.6 Example from a Swedish project group

In parallel to the SARNET L2PSA project a method for assessing the consequences of a hydrogen combustion in a Swedish BWR was developed within the Swedish project group.

The main components of the method are the following.

The thermo hydraulic state and the gas composition in the containment are calculated by MAAP, with the hydrogen combustion turned off. Thus the state is known for any time of a first ignition of the containment gas. The time periods where the gas is combustible and when DDT is possible are illustrated in Fig. 31 below showing a black-out sequence with activated containment sprays (powered by diversified power sources).

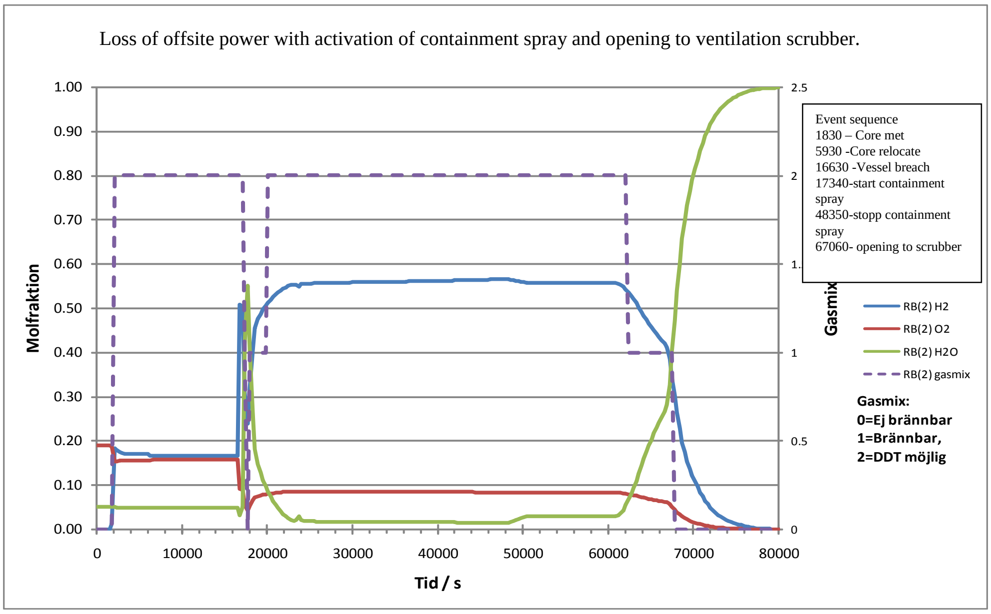

Fig. 31 Example for evolution of containment atmosphere

The solid lines show mole fractions of hydrogen, oxygen and steam and the dotted line “gasmix” show if the gas mix is combustible (value 1) or possible for DDT (value 2).

For the first time period when the gas is combustible the following assumptions are made regarding the ignition of the gas.

- If there is an identified continuous ignition source the gas will ignite as soon as it is combustible,
- If no continuous source but electric power is available the probability is 1 that the gas ignites within the first period,
- If no continuous source and no electric power is available the probability is 0.5 that the gas ignite within the first period.
- The gas will ignite at the vessel melt-through.

The timing of the ignition may be chosen as the worst regarding the consequences for the containment integrity or as a representative time for the sequence or group of sequences it represents. The time chosen is evaluated by running a series of MAAP analyses turning on the combustion at different times.

If DDT is possible during the period the probability for DDT is estimated through the method of Sherman and Berman with classifying the gas mixture and the geometry of the containment volumes.

If DDT occurs the integrity of the containment is assumed to be lost.

The method is then also applied to the following time periods.

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4.3.9 In-vessel steam explosion and consequences

The most threatening consequence of an in-vessel steam explosion would be the α-mode failure of the containment. This failure mode occurs if the RPV upper head is detached from the RPV body due to an energetic steam explosion, subsequently hitting the containment and inducing large containment failure.

All considerations related to in-vessel and ex-vessel steam explosion have been merged in section 4.4.3.

4.4 VESSEL FAILURE PHASE

4.4.1 RPV Bottom failure due to thermal loads

4.4.1.1 Introduction

During a severe accident the reactor pressure vessel (RPV) is subjected to significant thermal loads and, depending on the accident sequence, to pressure loads. If the corium pool is formed on the pressure vessel lower head and cooling is not introduced, the RPV will fail releasing large amounts of molten corium into containment. In the L2PSA, a key issue is the time of RPV failure, if it occurs. Besides the time of RPV failure, the location and size of the crack are also important issues, particularly for high pressure sequences. Realistic melt relocation predictions for the late-phase core degradation process are necessary to determine short term vessel response to corium jets and thermal shocks, while long term issues are governed by the evolution of the debris in the lower plenum.

RPV failure mode and timing is important for other issues such as corium cooling inside the vessel in case of core cooling restoration and RPV external cooling in case of retention of corium inside the vessel (see section 4.3.4). RPV failure mode and time are also important for the subsequent phenomena. Both the pressure vessel failure and phenomena following it, affect the releases to the containment and possible threats to containment integrity. RPV failure time is also related to the timing of releases and generally RPV failure is seen as a milestone in L2PSA. If the releases happen before RPV failure, it is considered to be early release and consecutively releases that happen later are considered to be late releases.

The RPV failure time and mode are plant specific. Two main types of RPVs can be distinguished; vessels with and without penetration nozzles at the bottom.

In principle two different issues have to be addressed when evaluating RPV bottom failure under thermal loads:

8. Heat flux into the vessel wall,
9. Response of the RPV structure to the thermal load.

The assessment of the heat flux is the more complicated issue. It is almost impossible to realistically predict local short term heat fluxes related to the impact of relocating core melt because the boundary conditions of the relocation process are unknown. At best, conservative assumptions can be made. In contrast, long term heat fluxes from fully developed molten pools can be evaluated with existing knowledge even though some uncertainties remain, particularly related to the transient evolution of a steel layer on top of the molten pool which could focus the heat flux. This issue is briefly described in this section and further discussed in section 4.3.4.

Structural mechanics of a RPV subject to transient thermal loads can be analysed with state-of-the art finite element codes. However, in PSA more simplified models typically in integral accident simulation codes are applied.

4.4.1.2 Description of accident phenomena

After the core has melted the melted material and particles from the fuel assembly leave the core region and they flow or fall down into the bottom of the RPV (Fig. 32). The material from structures, control rods and the fuel bundles (for example: Fe, Cr, Ni, B, Zr and their oxides together with the UO_2) arrives into the remaining water at the reactor vessel bottom. In the water the melted material flows down into the lower plenum and the melt jet fragments into solid particles. It forms debris, mixture of metal and oxide particles. The hot debris with their decay power evaporates the steam and start to heat up the RPV wall.

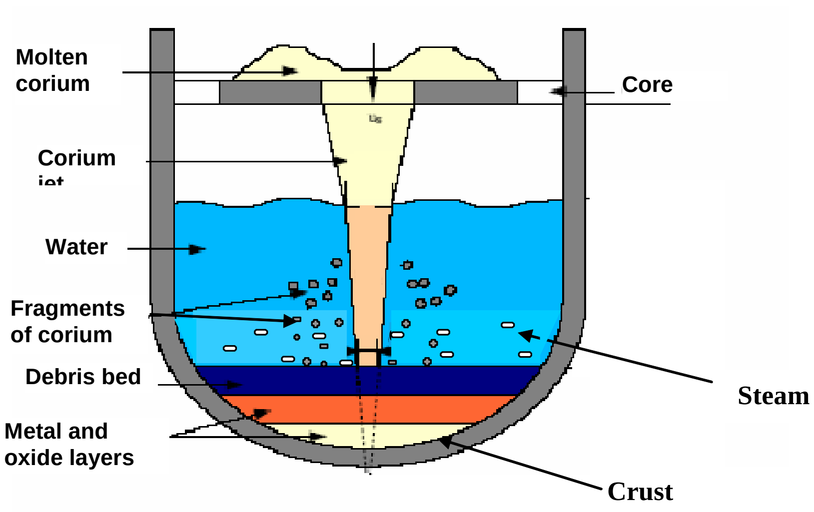


Fig. 32 Corium flows into the lower plenum

The core debris can form different structures at the RPV bottom. The composition of debris determines the type of stratification. It can be homogeneous melt, 2 or 3 layers. Typically it is expected that there will be a layer with heavy oxide materials in the bottom and metal layer on top of it. However, according to the MASCA [137] experiment there are 3 layers: heavy metallic, oxide and light metallic. At the bottom and the top of these layers solid debris (crust) can be found. In the RASPLAV [138] experiments depending on the composition of the melt 2 layers, oxide and metal layers were formed (Fig. 33). At the moment uncertainties in pool configuration (2 or 3 layers) are high. This is further discussed in section 4.3.4

Debris pool configuration is important since the evolved formation of the debris in the lower plenum determines the interaction (heat flux) between debris and RPV wall (Fig. 34). This gives the thermal boundary conditions which are the heat fluxes at the inner vessel wall. The boundary conditions are mainly important when the RPV is cooled from outside (see section 4.3.4) but they also determine the place and mode of the RPV failure.

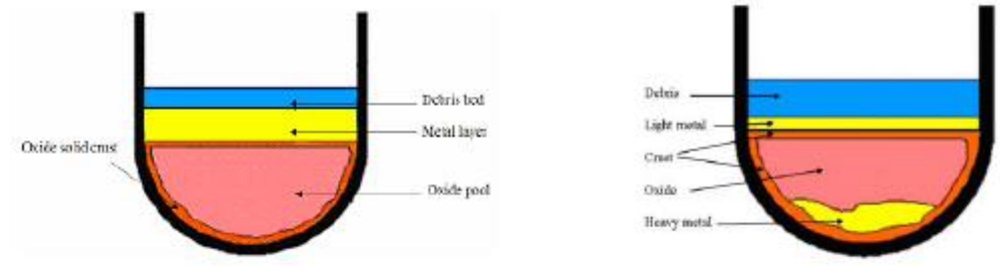


Fig. 33 Two type of layer structures

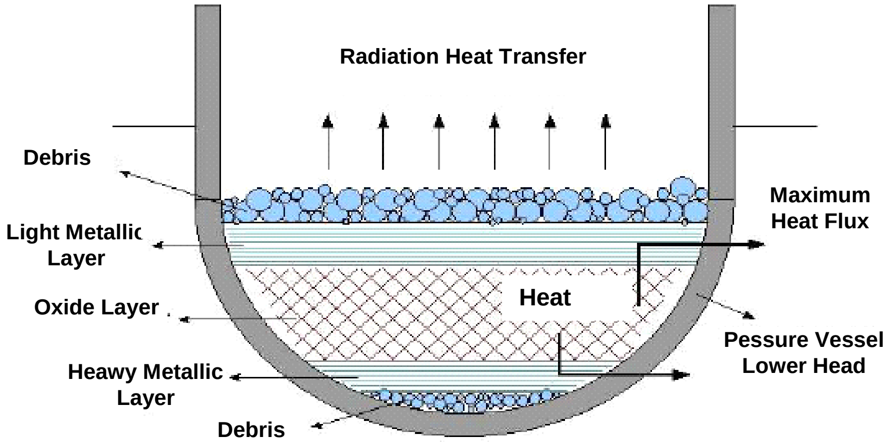


Fig. 34 Corium heat transfer to the reactor pressure vessel wall

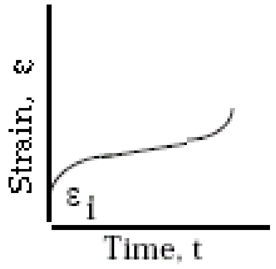
For RPV failure, knowledge of the temperature, stress and strain field in the RPV wall is necessary. The debris melt behaviour in the RPV bottom determines the heat flux to the wall (thermal load), which is the input for the temperature field of the wall. The main integral severe accident codes are able to calculate the heat flux to the wall and temperature distribution in the wall. The precision of the calculation is generally acceptable for L2PSA use. The stress comes from the pressure load in the RPV and from the weight of corium and RPV wall. Thermal stresses caused by the temperature gradient in the wall are considered negligible.

The RPV fails when the load or loads exceed its limits. The load can be local or global, which together with the special feature of the RPV, particularly if there are penetrations in the vessel (nozzles) or not, determines the RPV failure mode. The vessel failure can be:

- A global failure,
- Local failure of the vessel caused by a hot spot or,
- Failure of a nozzle.

The vessel steel has elastic, plastic, creep and thermal expansion properties which determine together with the loads the deformation and RPV failure mode. The vessel failure is caused by creep.

Creep is a time-dependent deformation at temperatures greater than 0.4 to 0.5 of the melting temperature (when the temperature is expressed in degrees Kelvin). Materials will slowly deform under loads which would not cause any plastic deformation at room temperature. The strain, instead of depending only on the stress, also depends on temperature. In creep testing a constant load is applied to a specimen and the specimen's elongation, or strain, is measured. This strain is plotted against time to form a creep curve. This curve usually contains three regimes, after the initial elastic strain. The first is primary creep where the strain rate is initially rapid and then decreases with time. Then the specimen enters into secondary creep, or steady-state creep, in which the creep rate is constant. This constant creep rate is called the steady-state creep rate, or minimum creep rate, since it is the slowest creep rate during the test. Finally, the specimen enters into tertiary creep, in which the creep rate continually increases until the specimen breaks. This event is called creep rupture or creep fracture, and is measured by the time to fracture.



Total Creep Curve

4.4.1.3 Experimental work

There have been huge experimental efforts to resolve the unanswered safety issues for the melt vessel interaction phase of the in-vessel progression of a severe accident.

FOREVER

FOREVER experiments [139] simulated the prototypic severe accident scenario of melt pool convection with the accident management action of vessel depressurisation, *i.e.*, the vessel pressure is maintained at around 25 bar. The vessel was at 1/10th scale *i.e.* its outer diameter was ~400 mm and wall thickness was ~15 mm. Some experiments were designed with eight 1/10th scale Inconel-600 penetrations, which were welded in the lower head of the vessel spanning from 15-55° from the bottom pole of the vessel.

The location of failure was found to be ~70° above the bottom pole of the vessel. The mode of failure was a crack which extended azimuthally from ~62° (for a French Steel) to ~95° (for an American steel). The opening area varied with different steels. The difference in behaviour seems to be due to the actual content of some minor elements than from the steel grade. Not all the melt discharged from the vessel, even at pressure of 25 bars. The FOREVER facility and vessel failure after test are introduced in Figure below (Fig. 35).

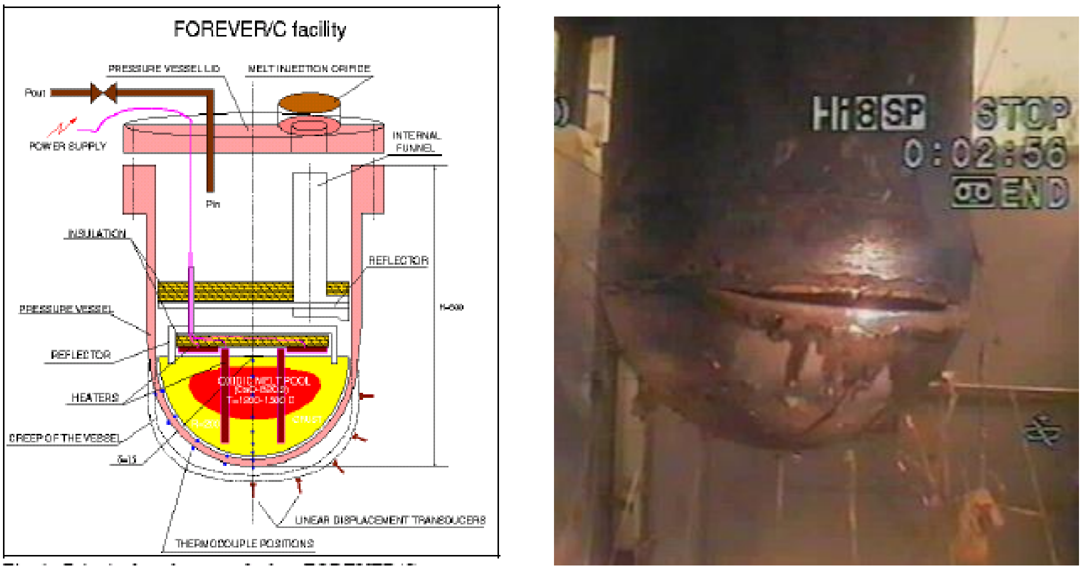


Fig. 35 FOREVER experimental facility and vessel failure after the test

TMI

The Three Mile Island accident involved the collapse of about 20 tonnes of corium into the lower head of the RPV. Despite the presence of water, the lower head reached temperatures of ~1300 K for 30 minutes in an area with an equivalent diameter of 1 m. During this period the reactor cooling system was at 10 MPa.

Although the Three Mile Island RPV did not fail, code analyses conducted in the course of an OECD/NEA TMI-II Vessel Investigation Project (VIP) predicted creep rupture in the prevailing conditions. This implies that during that time the state-of-the art modelling of the lower head failure was not mature because it did not take full account of the effect of the thermal loading. These methodologies have been further developed since the TMI-VIP project to analyse existing and next generation reactors from the perspective of accident assessment, management, and mitigation. To improve and validate structural analysis codes, there is a need for experimental data on lower head deformation and failure phenomena.

The objective of the project was to investigate the timing and size of lower head failure under conditions of low reactor coolant system pressure and large differential temperatures across the lower head wall. This objective was achieved through a series of experiments at the Sandia National Laboratory, USA completed in June 2002.

LHF

The Sandia National Laboratory has completed eight USNRC-sponsored tests on lower head failure (LHF) [140]. These tests were specifically designed to address lower head failure issues with prototypic material and geometry. The following OECD poject extended the USNRC SNL/LHF programme to address issues such as lower RCS pressures (representative of depressurised or partially depressurised conditions) and pressure transients. These tests also represent an improvement over previous tests by simulating a large temperature gradient across RPV of lower head wall. The temperature gradients addressed in these tests are representative of conditions without ex-vessel cooling.

FAILURE MODES ASSOCIATED WITH PENETRATION NOZZLES

Most of past studies devoted to the creep rupture have focused on global deformation and rupture modes. Limited efforts were made on local failure modes associated with penetration nozzles as a part of TMI-2 vessel investigation project (TMI-2 VIP) in 1990s. However, it was based on an excessively simplified shear deformation model. In the present study, the mode of nozzle failure has been investigated using data and nozzle materials from Sandia National Laboratory's lower head failure experiment (SNL-LHF). Crack-like separations were revealed at the nozzle weld metal to RPV interfaces indicating the importance of normal stress component rather than the shear stress in the creep rupture. Creep rupture tests were conducted for nozzle and weld metal materials, respectively, at various temperature and stress levels. Stress distribution in the nozzle region is calculated using elastic–viscoplastic finite element analysis (FEA) using the measured properties. Calculation results are compared with earlier results based on the pure shear model of TMI-2 VIP. It is concluded from both LHF-4 nozzle examination and FEA that normal stress at the nozzle/lower head interface is the dominant driving force for the local failure. From the FEA for the nozzle weld attached in RPV, it is shown that nozzle welds failure occur by displacement controlled fracture of nozzle hole not by load controlled fracture of internal pressure. Considering these characteristics of nozzle weld failure, a new concept of nozzle failure time prediction is proposed.

4.4.1.4 Calculation tools

The detailed mechanistic codes and also integral severe accident codes are calculating the strain field and failure mode. Integral codes can calculate in a simplified way the heat flux into the RPV lower head and also the RPV structural mechanics. For L2PSA purposes, the use of the severe accident integral code calculations can provide order of magnitude, even if they cannot accurately represent the experiments mentioned above. Some more detailed studies (2D or 3D finite element modelling) may be useful if realistic results or uncertainties analysis are required for L2PSA but for a realistic assessment, a good understanding of the corium pool behaviour and structure in the vessel, of the interface (crust) with the vessel wall and all thermal flux are needed.

Integral codes

MAAP4 [142] includes a parametric reactor vessel failure model. An RPV under stress at high temperature undergoes irreversible strain known as material creep. When the strain is large enough, the component can rupture. Failure, rupture of RPV lower head due to material creep is predicted by application of the Larson-Miller parameter method. While this method originally developed for isothermal sections it can be applied to a specimen with temperature gradient under the assumption that equal strain is present at all points.

Larson-Miller parameter: L-M parameter is useful in understanding and quantifying the time versus temperature trade-off for various materials. Its use results in a very effective method for rationalising the time-temperature (and even rate-temperature) effects observed in stress-rupture and creep testing. L-M parameter is specified with respect to [141]: LMP = Tx10-3 (C+logtf) where, C is a coefficient, whose value is dependent on the material chosen. In the Larson-Miller study, data for some 40 materials were evaluated: it was found that the constant C was very close to 20 for all materials [NAE].

ASTEC code includes three different mechanical models (beyond simple user-defined criteria) to describe the behaviour (stresses and strains) of a RPV lower head loaded by thermal, pressure and weight loads under severe conditions: the Oeuf model is applicable to the perfect half spherical initial shape of the lower head whereas the Lohey and Combescure models are applicable to half spherical or half ellipsoidal lower heads (like in VVER). The Oeuf model has been validated on several LHF, OLHF and FOREVER experiments.

The vessel rupture can be obtained using several criteria: temperature, molten fraction of vessel lower head, mechanical stress, instantaneous plastic rupture, creep rupture.

The lower head in the ICARE module is meshed through the thickness (usually around 5 meshes) and axially along the vessel (usually around 10 meshes).

- Combescure model [144]: instantaneous plastic rupture (at each axial level, the applied stress is compared to the ultimate steel stress, calculated for different types of steel, like SA533B or 16MND5 ones); or creep rupture (a rupture time is calculated from a creep deformation velocity and is compared with the current accident time),
- Oeuf model [145] : it assumes that the deformation remains axisymmetric and the final shape looks like an "egg" shell. The mathematical formulation takes into account the non-linearities, large displacement and large strain. This model is based on the solution of equilibrium equations of shells of revolution under symmetric loading. At each time step the condition of equilibrium and compatibility is satisfied for each numerical element. The following criteria are checked for vessel failure: fragile failure, ductile failure, failure due to small thickness, failure due to excessive displacement, failure due to excessive creep velocity, failure due to excessive creep deformation. This model only transfers to ICARE the rupture location and time (and not the vessel deformation). The creep and damage constitutive equations are the Norton-Bailey type creep law and the Lemaitre and Chaboche coupled damage-viscoplasticity model,
- Lohey model: the half-spherical or half-ellipsoidal lower head is considered as a set of symmetric around vessel axis hoop elements deformed independently. The following loads act in hoop elements: temperature loads, meridian and hoop normal forces caused by internal overpressure and weight corium and vessel wall. The hoop element is considered as a multi-layer shell. The layer temperature along radial, hoop and meridian direction is constant. The vessel steel has elastic, plastic, creep and thermal expansion properties. The layer stress state is assumed biaxial. Radial stress is not considered. Two failure criteria are considered: the layer fails if the plastic strain intensity exceeds the ultimate failure strain; or the current damage of each layer induced by creep strains is determined using the correlation for time to rupture at the given stress and temperature. A life fraction rule is used to calculate the cumulative material damage under transient conditions. The lower head global rupture occurs when all layers of the hoop element melt or fail.

Other codes

TOLBIAC code is devoted to the simulation of the behaviour of a corium pool with natural convection within a structure, which may be the RPV or a core catcher, to study the structural behaviour. The following main phenomena can therefore be calculated: metal-oxide stratification, residual power, free surface heat transfer (radiation, or heat transfer with water), wall ablation and crust formation. Its main characteristics are the use of a 3 field equation system, in a 2D cylindrical or rectangular geometry.

There are more sophisticated codes as MVITA (SKH) which calculates the heat flux from the corium to the wall and the deformation and failure of the wall. There are special structural analyses codes as ANSYS code using FEM. The Finite element method (FEM) is a powerful technique originally developed for numerical solution of complex problems in structural mechanics. These codes can recalculate the experimental values. Generally these detailed codes are not necessary for PSA. These codes can be used for specific questions, and also for the validation of integral codes.

4.4.1.5 Status of knowledge and main uncertainties

The local heat flux from the corium to the different parts of the vessel wall is uncertain. It depends mainly to the corium composition and mass in the bottom of the vessel and it is time dependent. The main origin of this uncertainty is the clad oxidation because it determines the composition of the melt pool, and strongly influences the heating and melting of the core. The fuel melt and flow down are also uncertain together with the formation of oxide and metal layers.

The structural response of the reactor vessel can be calculated quite well with FEM codes and generally the knowledge of integral codes (ASTEC, MAAP, MELCOR) are sufficient for PSA purposes. However, the remaining uncertainties have to be taken into account when code calculations are used in L2PSA.

4.4.1.6 References

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4.4.2 Issues related to high pressure RPV failure

At vessel failure, it has to be distinguished in L2PSA whether or not the RPV is pressurised when it fails. A high pressure, RPV failure could lead to immediate catastrophic consequences by containment overpressurisation, while a low pressure failure may only lead to gradual containment challenges (if no other energetic phenomena like ex-vessel steam explosion occurs). T

The following sections are directed to the immediate consequences of vessel failure, in case of high pressure at vessel failure time.

It should be recalled that in many reactors, accident management measures are foreseen to depressurise the RPV. In addition, the reactor coolant loops could develop induced leaks due to high pressure combined with high temperature from the core degradation, thus reducing the primary pressure.

Therefore, high pressure RPV failure may be much less likely than failure at low pressure. Nevertheless, even a remote probability for high pressure sequences necessitates an analysis of the related issues. In particular, the RCS pressure below which no risk is induced for the containment must be estimated.

In an APET, typically the following branching points can be introduced (see following chapters for the details):

1. What is the RPV pressure at bottom failure?
    a. Is this pressure below the calculated threshold for DCH effects for the reactor? (typically between 0.7 and 2 MPa assuming a large bottom leak),
    b. Is this pressure below the calculated threshold for RPV lift up ? (typically above 8 MPa assuming a large bottom leak).
2. What is the RPV bottom leak size,
3. Does RPV lift up occur
    a. Without damaging the containment,
    b. With damaging the containment.
4. Does DCH occur
    a. With DCH pressure below containment load limit,
    b. With DCH pressure above containment load limit.

4.4.2.1 Description of accident phenomena

In the case of a core melt accident, molten core materials relocate into the RPV lower plenum and heat up the RPV lower head until it fails. Two interrelated but different issues have to be considered: direct containment heating (DCH) of the containment and lift up of the RPV. First, DCH is addressed, and then lift up is discussed.

Direct containment heating (DCH):

Melt is ejected into the reactor pit in a jet. Fragmentation and break-up of the jet lead to the creation of a large surface of the melt available for heat transfer and oxidation of the metallic components of the fragmented melt. Further fragmentation can occur when the jet hits the walls of the pit. The RPV bottom leak creates a reaction force on the RPV. In the case of a central leak in the RPV this reaction force is directed upwards. Furthermore the pressure in the pit increases due to the in-flow of steam which can only partially be compensated by the outflow of steam into the upper part of the containment (steam generator and pump rooms in case of a PWR) through the annular gap between the RPV and the pit and along the main coolant lines. The steam that leaves the RPV and the lower part of the containment (the pit in case of a PWR) can entrain fragmented parts of the melt and transport them into the upper parts of the containment via the path dependent on the type of the plant (along the main coolant lines and the SG towers in case of a PWR). The melt droplets transfer part of their stored heat to the containment atmosphere. The corresponding pressure increase may be further pronounced by the oxidation of the remaining zirconium (Zr) either by steam, generating hydrogen or directly by oxidation by oxygen. The latter reaction produces the same amount of energy as the production and combustion of hydrogen and leads to a similar heat-up of the atmosphere. Furthermore, the hot particles may ignite the hydrogen produced before RPV failure in cases where the containment is not inerted. The short term pressure increase in the containment can lead to a large containment failure due to overpressurisation.

RPV lift up:

The RPV bottom leak creates a reaction force on the RPV. In the case of a central leak in the RPV this reaction force is directed upwards. The pressure build-up in the pit leads to another force on the RPV in an upwards direction. The reaction and the pressure force combine to an uplift force that accelerates the RPV upwards. If the uplift force minus the gravitational force exceeds the strength of the main coolant lines the RPV is lifted off. With the RPV moving upwards kinetic energy is transformed into potential energy until, at a certain trajectory height the RPV stops its upward movement. If this height is larger than the height of the top of the containment, the containment integrity might be threatened by the RPV crashing through the containment ceiling thus causing a large containment failure. But even if the RPV does not hit the containment directly, the mechanical impact associated with the movement and fall back of several hundred tons could threaten the containment.

Calculation of DCH-related Issues:

A full deterministic calculation of the melt entrainment, the transport of the particles and the heat transferred to the atmosphere is not usual in the frame of a L2PSA and also not meaningful regarding the large range of possible boundary conditions starting with the failure mode of the RPV, the particles size spectrum and the associated transport. Instead, it is proposed to use simple correlations, e.g. in the frame of Monte Carlo calculations, fitted to experimental finding or to generic calculations with sophisticated codes.

4.4.2.1.1 RPV bottom failure for pressurised conditions

There have been a number of experiments dedicated to the investigation of creep deformation and failure of RPV steels. In the following section some of these experiments the results of which are relevant for DCH and RPV movement issues are described briefly. At the end of this subsection a short summary of the RPV failure issue is given.

The Sandia Lower Head failure (LHF) experiments

Prototypic steel 1:4.85 scaled vessels were employed by the Sandia National Laboratories (SNL) to study creep behaviour. The steel is similar to the one used in the RPVs in US power plants. These RPV lower head failure (LHF) experiments were performed at high primary pressure and the lower head was heated by an arrangement of electrical heaters inside the vessel. There was no melt present in the lower head.

The LHF tests were performed with different types of heating distributions: uniform, edge-peaked and centre peaked. LHF-1, LHF-2, LHF-3, LHF-6, LHF-7 and LHF-8 were experiments without penetrations on RPV bottom. LHF-4 was an experiment with uniformly heated vessel and 40 penetrations. LHF-5 was an experiment with edge-peaked heating pattern and 9 penetrations. With the exception of one experiment with 50 bar (LHF-7) the pressure was 100 bar in all cases.

For uniformly heated vessels the failure location was at the minimum wall thickness. In contrast, for the edge-peaked heating pattern (to approximately simulate the melt pool convection) the failure occurred at the location where the maximum temperature occurred. The vessel failed with a limited straight and distinct circumferential rip following the locus of the peak temperature.

The first three experiments showed fairly consistent temperatures at the onset of creep (T_{in}: 935 K to 980 K) and vessel failure (T_f: 1006 K to 1038 K). The time to failure was typically tens of minutes.

LHF-6 was designed to investigate the effect of weldment on vessel deformation and failure. With uniform heating, the vessel weldment was not challenged significantly. The results of LHF-6 were similar to that of the corresponding experiment without weldment LHF-1 with respect to local deformation history, post-test vessel shape and failure temperatures. The leak size area was smaller than in LHF-1.

For the lower pressure (LHF-7) higher values for the temperatures for the onset of creep (992 K) and vessel failure (1200 K) were observed. There were also significant differences in the failure characteristics: while all LHF experiments with 100 bar resulted in severe necking with a thickness reduction of a factor of ten, the corresponding thickness reduction for LHF-7 was only a factor of two.

The areas of failure for different experiments are contained in the table below. This table (Table 25) also lists the leak sizes obtained when scaling up to EPR™ size (by multiplying by a factor of 4.85^2).

Table 25 Summary of Sandia LHF experimental results (no penetrations)

Test	Heat flux distribution	Pressure (bar)	T_{in}(K)	T_f(K)	Failure area (m²) Scaled *	Failure shape
LHF-1	Uniform	100	935	1038	0.0949 2.2	Oval at bottom
LHF-2	Center peaked	100	958	1010	0.00175 0.041	Oval at bottom
LHF-3	Edge peaked	100	980	1006	0.0135 0.32	Fish mouth not at bottom
LHF-6	Uniform	100	949	1052	0.0138 0.32	Oval at bottom
LHF-7	Uniform	50	992	1200	0.0036 0.085	Latitudinal rip
LHF-8	Edge peaked	100	967	1041	0.0027 0.063	Fish mouth not at bottom

* to EPR™ vessel size

The OECD lower head failure tests (OLHF)

In the OECD lower head failure (OLHF) experiments performed at Sandia National Laboratories. The partly depressurised (20-50 bar) severe accident scenario was modelled and the wall thickness was increased by about 2.4 times to obtain a larger temperature difference across the wall (ΔT_{wall}>200K). The heating was provided by a graphite induction furnace installed inside the vessel.

The results of the OLHF tests are summarised in the table below (Table 26).

Table 26 Summary of Sandia OLHF results

Test	Heat flux distribution	Pressure (bar)	T_{in}(K)	Failure area (m²) Scaled *	Failure shape
OLHF-1	Uniform	50	1450	0.00171 0.04	Latitudinal rip
OLHF-2	Uniform	20	1750	0.00365 0.086	Latitudinal rip
OLHF-3	Uniform	20-50	1380	0.118 2.78	Azimuthal rip

* to EPR™ size

Forever Experiments

The FOREVER experiments were initiated at the Royal Institute of Technology in Stockholm to simulate the prototypic severe accident scenario of melt pool convection in a depressurised vessel (at 25 bar). Outer diameter of the vessel is approximately 400mm and the wall thickness is approximately 15 mm. The scaling factor in geometry is 1:10. The vessel lower head is either made from the 16MND5 steels used in vessels of

French plants or the American steel used in the LHF/OLHF tests explained earlier. The cylindrical part is made of German steel. In contrary to the LHF and OLHF experiments prototypic melt pool convection and heat fluxes are simulated as in a real accident.

Seven experiments have been performed.

It was shown that the time of RPV failure depends primarily on the imposed maximum temperature from the thermal load.

The vessel failure mode is like a fish mouth. The vessel failure occurs 70° from the bottom of the vessel where the hot zone is located. The failure occurs at that location where the wall temperatures are the highest and not where the largest creep displacement occurs. The failure crack travelled circumferentially. The failure size is larger for the American steel than for the French steel.

Summary on RPV failure

The above described experiments provide information about size and location of RPV failure that can be used for DCH and RPV movement considerations.

As first failure mode, a central leak at the bottom of the RPV has to be investigated. Such a RPV failure might occur in a scenario where a local melt jet reaches the bottom of the RPV and heats it until the RPV fails due to the thermal loads. Such an early failure of the RPV at the bottom is very improbable for low pressure RPV failure and can only be expected in case of high RPV pressure because the higher the pressure the lower the failure temperature of the RPV steel and the earlier the RPV failure. In that case the failure size can be inferred from the experiments LHF-1, LHF-2 and LHF-6 to be between 0.04 m^2 and 2.2 m^2 with a best-estimate value of 0.3 m^2.

The second failure mode, a lateral hole in the RPV is the most probable failure mode because when a melt pool develops in the RPV lower head a separation between oxidic and metallic melt occurs with a metallic melt layer on top of the oxidic melt (more information on possible melt pool configuration in section 4.3.4 and 4.4.1). Due to convection heat is transferred from the melt to the RPV wall through the metallic layer until the RPV fails locally at the height of the metallic layer with a lateral hole. The hole can have the shape of a latitudinal rip or a fish mouth. Typical failure areas can be inferred to be between 0.04 m^2 and 0.32 m^2 with a best-estimate of 0.08 m^2 from the experiments LHF-3, LHF-8, OLHF-1, OLHF-2 (lower pressure < 100 bar, uniform heat flux distribution). For an azimuthal rip a larger failure area of 2.7 m^2 is obtained (OLHF-3). Hence the leak size distribution is similar than the one for a central leak.

The third failure mode is a circumferential rupture following a latitudinal rip. In the FOREVER experiments it could be seen that the crack travelled circumferentially even though it did not result in a circumferential rupture. In that case the RPV failure size depends on the vertical location z of the rip. If z is large like in the case of LHF-3 and LHF-8 the leak area can be as large as the RPV diameter, if z is small, on the other hand like in OLHF-1 and OLHF-2 the leak area can be quite small comparable to the leak sizes for the second failure mode.

For the leak size distribution a Gaussian distribution centred at 0.3 m^2 is a reasonable choice. For the 5th percentile the lower limit of 0.04 m^2 and for the 95th percentile the upper limit of 2.2 m^2 is adopted. By choosing a Gaussian distribution the case of a large circumferential rupture is also covered by the upper edge of the distribution.

When considering containment failure due to RPV movement only cases with a central leak are of interest because only then the reaction force will accelerate the RPV upwards in vertical direction.

For DCH both central and lateral leaks are relevant.

4.4.2.1.2 Melt entrainment

At the Karlsruhe Institute of Technology (KIT) an experimental programme is performed for DCH.

The DISCO test facility at KIT was set up to perform scaled experiments that simulate melt ejection scenarios in Severe Accidents in Pressurised Water Reactors. The fluid-dynamic, thermal and chemical processes during melt ejection out of a breach in the lower head of a PWR RPV are investigated at pressures below 20 bar. The main components of the facility are scaled linearly by a ratio of about 1:18 to a specific PWR (EPR™).

DISCO experiments dedicated to the investigation of melt entrainment are described briefly in this section. At the end of this sub-section a short summary of the melt entrainment is given.

<u>The DISCO-C experiments</u>

In the DISCO-C (Dispersion of Simulated Corium Cold) test facility experiments were performed using cold stimulant materials (water or Wood's metal instead of corium and helium instead of steam as driving fluid.

The following failure modes were studied: central holes and three types of lateral branches, lateral holes, horizontal slots and complete circumferential failure of the lower head. The initial RPV pressure and the hole size were varied.

In case of a central leak and water as melt simulator the fraction of melt dispersed from the cavity F_d can be as high as 80 %. In case of a lateral leak or slot this fraction is lower: between 2.5 % and 50 % while it lies between 1 % and 22.5 % in tests with unzipping of the lower head. When plotting the dispersed fraction as a function of pressure saturation can be found because of freezing of melt on the surfaces of the pit. The upper limit found in the studies is 80 % which is consistent with the American studies like the NUREG6338 which state that freezing on cavity surfaces retains about 10 % of the melt.

In case of Wood's metal as simulator the amount of dispersed melt is lower than in case of water.

For a 25 mm hole (0.45 m scaled, area of 0.6 m²) and a pressure of 11 bar the following effects on F_d are observed:

- Moving the hole from the central to the side: F_d is reduced from 0.56 to 0.36,
- Using metal instead of water: F_d is reduced from 0.358 to 0.00005.

<u>The DISCO-H experiments</u>

In the DISCO-H experiments 10.6 kg of iron-aluminia melt (16 m² corium scaled) was used as a corium stimulant with steam as driving fluid and a prototypic containment atmosphere. The diameter of the hole at the centre of the head was chosen to be 56 mm (1 m scaled, area of 3 m²) or 28 mm (area of 0.75 m² scaled).

In the base case experiment DISCO H02 EPR™-like geometry with a direct path from the pit to the containment was simulated (open). The initial pressure in the RPV was 12 bar and the diameter of the hole was 56 mm. The melt was part metal part oxide and the atmosphere in the containment was air, steam and some hydrogen. The initial pressure in the containment was 2 bar, the initial temperature 100°C (373K).

Test H01 had less steam in the RPV and the initial pressure in the RPV was only 8 bar versus 12 bar in the base case. Test H06 had an initial pressure of 22 bar and a smaller hole of 28 mm. In test H04 the hydrogen effect was excluded by using only nitrogen as driving gas instead of steam.

In the two experiments DISCO H03 and DISCO H05 the direct path to the containment was closed to simulate an EPR™-like geometry with no direct path from the pit to the containment (closed). In both experiments H03 and H05 the initial pressure in the RPV was 12 bar and the diameters of the hole were 56 mm and 28 mm, respectively. The melt was part metal part oxide and the atmosphere in the containment was air, steam and some hydrogen. The initial pressure in the containment was 2 bar, the initial temperature 100°C (373K).

The melt fractions dispersed from the pit (f_{leave_pit}), transported to containment (f_{con}) and transported to the sub-compartments (f_{subcom}) are listed in the table below (Table 27).

In the two cases with closed cavity only a small fraction of the melt that reaches the sub-compartments ends up in the containment, while in the other cases with open pit the amount of melt in the containment exceeds the amount of melt in the sub-compartments because melt can enter the containment not only through the sub-compartments but also directly from the pit.

Table 27 Melt transport fractions for the DISCO-H experiments

Experiment	H01(open)	H02(open)	H03(closed)	H04(open)	H05(closed)	H06(open)
p_{ini} (bar)	8	12	12	12	12	22
$a_{RPV\_pit}(m^2)$ scaled	3	3	3	3	0.8	0.8
f_{leave_pit}	0.355	0.605	0.458	0.753	0.377	0.486
f_{con}	0.237	0.503	0.022	0.663	0.018	0.361
f_{subcom}	0.120	0.102	0.435	0.090	0.359	0.125

It was shown that a smaller cross section leads to less dispersion (at the same failure pressure, H05 versus H03).

Higher failure pressure only partly compensates the smaller hole in respect to melt dispersed from the cavity (H06 versus H02).

The experiments show that if there is no direct path from the pit to the containment dome (except the small area along the main-coolant lines) there may be a considerable ejection into the pump and steam generator rooms, but almost nothing into the open space of the containment.

<u>**Summary on melt entrainment**</u>

The DISCO-H experiments provide information on the fraction F_d of melt that leaves the pit in case of a central leak in the RPV in case of low RPV initial pressure representative for low pressure scenarios with RPV failure below 20 bar: between 35 % and 75 %. For higher initial RPV pressure it can be expected that the fraction F_d is higher and can reach up to 100 %.

The DISCO-C experiments with water as stimulant indicate dispersed fractions F_d between 2.5 % and 50 % for a lateral leak in the RPV and between 1 % and 23 % for an unzipping of the lower head.

The comparison of experiment H03 (closed cavity) with the experiment H02 (open cavity) shows an increase in the dispersed fraction from 0.458 to 0.605, i.e. by a factor of 1.3. 75 % are dispersed into the equipment rooms and 5 % into the cavity.

In consistency with American experiments an upper limit for F_d of 90 % corresponding to 10 % freezing of the melt on the pit walls can be derived.

The DISCO-H experiments provide information on the retention capability of the equipment rooms.

From the two experiments with closed cavity H03 and H05 it can be concluded that in the case of low RPV pressure (12 bar) the steam generator and pump rooms possess a high retention factor f_{disco}, only 5 % of the melt that reach the sub-compartments reach the containment dome.

In principle, care should be taken to couple the aspects of RPV lift up (if it occurs) and the dispersal of the melt droplets because the RPV movement creates new paths from the pit to the dome, which is particular of importance for large dry PWRs. In addition, the failure of walls, e.g. pit walls, may influence these paths. However, since the RPV lift up issue is partly a structural mechanics problem, and since coupling of structural mechanics with DCH thermodynamics is not yet state of the art, it is recommended to apply engineering judgement for melt entrainment in this case.

4.4.2.2 Containment pressurisation

While for PWRs the nodalisation of the containment in three to four atmospheric nodes may be sufficient the suppression pool has to be taken into account for BWRs as part of the melt droplets enter the water phase via the downcomers before depositing the stored heat in the atmosphere.

With respect to the heat deposited in the containment atmosphere, the oxidation of Zr with steam and, in case of a non-inerted containment, with oxygen and the combustion of all the hydrogen (generated during DCH and remaining hydrogen from the previous core degradation) has to be considered.

The load distribution can be either calculated within the Monte-Carlo codes by fast-running correlations based either on detailed calculations, e.g. with codes of the SIMMER family, such as AFDM, or with general severe accident codes, such as ASTEC or COCOSYS, or directly based on the experiments described above. Multiphase flow codes like MC3D (figure) also offer a solution able in near future to calculate vessel depressurisation, melt ejection and fragmentation, reactor cavity pressurisation, vessel displacement, containment pressurisation. Although still under research activity, MC3D applications have been recently performed by IRSN to support a French 1300 MWe PWR L2PSA and provide melt ejection correlation and forces applied to the RPV.

To calculate the containment failure probability due to a high pressure breach of the RPV, the fragility curve of the containment and the containment response to mechanical loads (the impact of the RPV) also has to be implemented in the MC code.

To create the distributions for the Monte Carlo code one has to define distributions for the input parameter. Main source of uncertainties are:

- The accident progression (relation to the deterministic codes used in the frame of L2PSA):
    - Mass of melt,
    - Zr-content,
    - Initial hydrogen mass in the containment,

- Pressure in the RPV at time of vessel breach,
- Pressure within the containment before vessel breach.

- The epistemic uncertainties:
  - RPV failure mode,
  - Melt entrainment,
  - Melt droplets deposition on the way from the pit to larger compartments.

With the aid of such MC codes the corresponding branch probability for the APET and also its distribution can be calculated.

As these phenomena are not very plant specific – given all the uncertainties – values from the literature can also be used in a more simplified PSA, possibly adapted to the plant specific fragility of the containment or to other plant specific vulnerabilities.

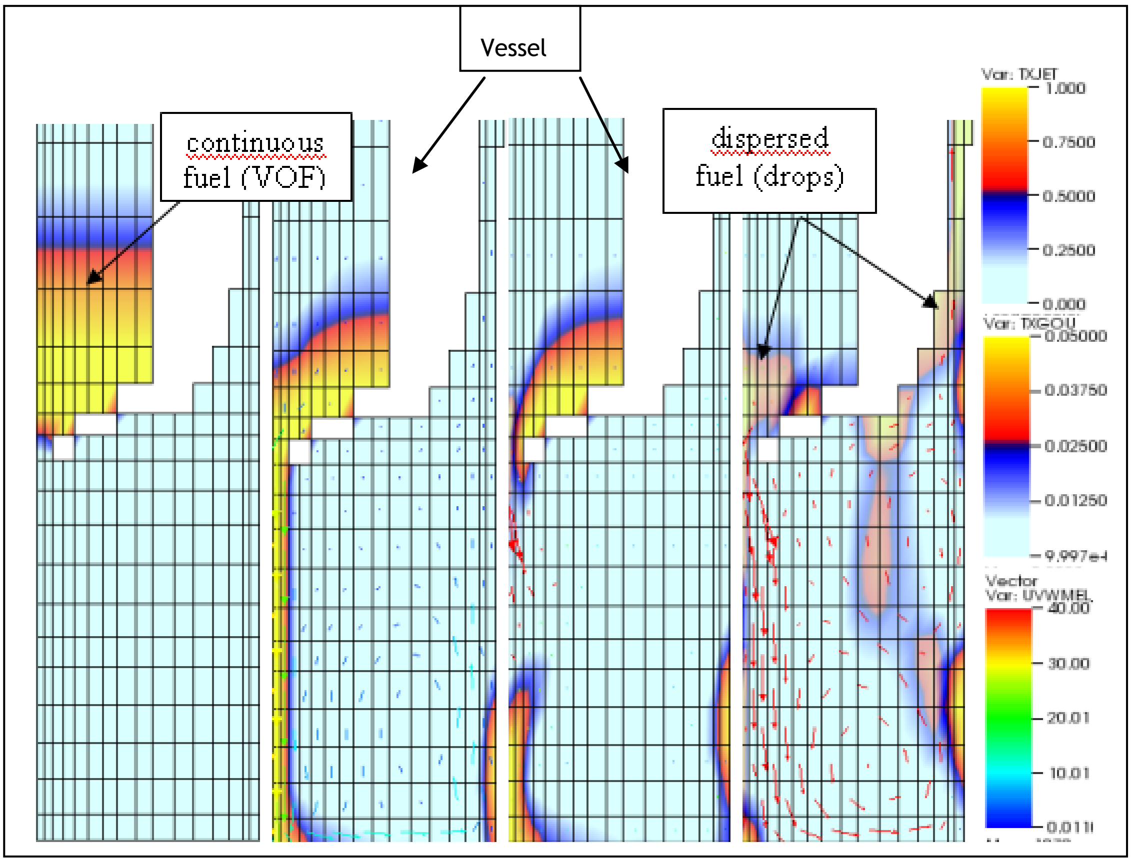

Fig. 36 Visualisation of melt ejection fragmentation and upward dispersion in a typical simulation with rough mesh for PSA-2 evaluations. Legend: top: continuous fuel field volume fraction, middle: fuel drop volume fraction, bottom: drop velocity (m/s) (MC3D from IRSN 1300 MWe L2PSA)

4.4.2.3 Lift up of RPV

Lift up of the RPV depends on the conditions inside the RPV when it fails, the RPV failure size and position, the thermodynamic processes leading to reactor pit pressurisation, and finally on the mechanical resistance of the structures around the RPV. The first three aspects are closely linked to the DCH issue and could be analysed together with it. The mechanical analysis of the forces needed to lift up the RPV is a different topic, and it is very much dependent on the mechanical design of the plant under consideration.

The lift up of the RPV is impeded by at least two aspects:

- The mass of the RPV itself and any directly connected structures (e.g. in a German PWR a heavy concrete ring would have to be lifted up as well),
- The resistance of the main coolant lines, depending on details of the geometry, and taking into account the probably very high temperature of the coolant lines.

In a typical German PWR lift-up of the RPV and subsequent failure of the containment is assumed if the RPV bottom leak is 10 m^2 and if the RPV pressure is at least 8 – 10 MPa [146].
Since precise analysis is very complex it is recommended to first make a screening analysis about the necessary forces for lifting the RPV. If these forces turn out to be in the range of the loads to be expected, a satisfactory analysis delivering consequences and associated probabilities may be hard to obtain.

It can be mentioned that multiphasic codes like MC3D offers a solution to model vessel uplift taking into account the vessel depressurisation, the vessel pit pressurisation and the induced forces on the vessel. It can be used to validate some order of magnitude calculated during a screening analysis.

4.4.2.4 Reference

[146] [Deutsche Risikostudie Kernkraftwerke, Phase B, GRS 1990, ISBN 3-88585-809-6, section 8.5.2.7].

4.4.3 Fuel Coolant Interaction (FCI) and steam explosion

Remark: this chapter presents fuel coolant interaction for both in-vessel and ex-vessel situations.

4.4.3.1 Description of accident phenomena

Fuel coolant interaction (FCI) occurs in different phases of a severe accident. In the in-vessel phase, molten core materials relocate into the water-filled lower plenum of the RPV. This leads to potentially violent thermal interactions between the fuel and coolant that, in the extreme case, might have an explosive nature (in-vessel steam explosion). This energetic event could endanger the containment integrity if the energy released by the in-vessel steam explosion accelerates a liquid slug of core melt towards the RPV head. When the energy is sufficient to lift off the upper head, the upper head is subsequently accelerated towards the containment ceiling and causes a large containment failure (α-mode failure). Lower head failure and the failure of the RCS pipes and steam generator tubes may occur instead. Furthermore the radionuclide release is increased due to the generation of large melt surface areas.
In the ex-vessel phase, FCI may take place in the reactor cavity provided that there is water present. A violent FCI (ex-vessel steam explosion) has the potential to endanger the structural integrity of the reactor cavity

which in turn may endanger containment leaktightness. Even in the case of no cavity/containment failure there is a risk that the functionality of the melt retention capability is reduced.

Furthermore, FCI can also take place in the core catcher (for relevant plants) when the melt is flooded. Whilst this is not a violent FCI there is, nevertheless, a possibility of containment failure due to the pressure spikes from melt quenching.

The term FCI can be used to describe all processes induced by the mixing of a hot molten fuel within a volatile liquid coolant:

- Dynamical mixing between both fluids (premixing),
- Eventually steam explosion (explosive FCI).

Therefore, FCI can be a generic term used to designate all phenomena occurring during the initial mixing of a melt and a coolant, ignoring the issue of debris cooling. FCI involves the “slow” mixing, generally called premixing, and, eventually, a steam explosion (“fast” mixing). The long term interaction of corium debris and water is generally treated separately due to different time-scale but one should remind that the long term coolability of a corium debris bed is strongly influenced by the phenomena that can occur during the initial interaction between corium and water.

Generally, when premixing involves low velocity melt jets in water pools, the steam explosion occurs as a sudden destabilisation of the premixing, i.e. with a very distinct behaviour and scale. In such a case, the driving mixing process is different as it is due to the pressurisation itself which is self-sustaining.

In the past much effort has been dedicated to in-vessel FCI research. It finally lead to a wide consensus that the probability for RPV head rip-off and consequential containment failure due to in-vessel FCI is very low if the vessel is intact. Some parties even consider the issue as “closed”. Consequently, compared to other phenomena, there is little R&D regarding FCI involving only a few teams and very few tools are under development. There is also a shortage of new experimental data for code validation for both premixing and explosion, particularly data which can be used quantitatively.

Currently, the R&D efforts are grouped within two international structures/projects. SERENA-2 is an international OECD project involving two experimental facilities devoted to steam explosion: KROTOS (CEA) and TROI (KAERI). The second international collaborative effort is SARNET which is focussing on modelling.

Some national experimental projects also exist with investigations on particular geometries or phenomena (e.g. AECL, BARC, KTH).

4.4.3.2 Description of issues

4.4.3.2.1 Premixing

The problem involves the dispersion of a corium jet at approximately 2500-3000 K in a pool of water. It has to be mentioned that according to integral accident analyses the temperature of corium relocating into the lower plenum may be significantly lower. The premixing time scale is that of the melt injection, i.e. the time to eject the melt into the water of the lower plenum or the cavity. This is roughly the same time scale for melt fragmentation and cooling processes.

A- Fragmentation

During this time scale, the melt jet will undergo instability and fragmentation.

There are several modes of instability and these phenomena are the most difficult to model in FCI premixing since the theories are undeveloped even for simple cases (e.g. water jet in air).

In the case of gravitational pour, the jet velocity is small and it is likely that the mixing/ fragmentation is primarily driven by the counter flow imposed by the vapour. Fragmentation drives vaporisation that, in turn, drives the fragmentation itself. Experiments have confirmed that the general characteristics of fragmentation can finally be somewhat independent of the ambient conditions. If the fragmentation is complete and the pool sufficiently large, the characteristic final drop diameter is relatively independent of conditions, of the order of some millimetres (1-4 mm mean for dense melts like corium) but with a rather extended size spectrum (starting from tens of microns).

However, the fragmentation is limited by a specific time scale which is the time for the melt to flow in the coolant before reaching the basemat and, depending on the jet diameter, fragmentation might not be complete. A very simple formula was proposed by Meignen to evaluate the length scale necessary for jet fragmentation as $L/D = 5 \times V_{jet}$, valid only for corium jets in water flowing under gravitational conditions (with low velocities of up to 3-4 $m.s^{-1}$). This is however a maximum as the jet might be destabilised by large scale phenomenon or may not flow vertically (side break). Additionally, for long-lasting pours, the conditions for fragmentation might change with time if a high vaporisation occurs and/or if the water is expelled out of the cavity.

The case of high melt velocities has been far less investigated theoretically and experimentally, particularly in the frame of steam explosion. Although this case might be simpler, as the fragmentation might be more strictly related to the jet velocity, there is a real lack of data. Note that in the ex-vessel situation, it is likely that a pressure difference of several bar exists (between vessel and cavity) and this ensures high velocities.

Thus, it is difficult to use a simple classification and characterisation scheme for fragmentation.

B- Void generation

The second important aspect is the generation of voids (i.e. gases in mixture) which has several impacts.

Firstly, although the impact is not quantitatively clear, voiding can suppress the explosion or strongly limit its strength. This is the major reason why steam explosion is difficult to trigger in conditions with low sub-cooling and spontaneous explosions have been reported only in sub-cooled cases. Therefore, for premixing, voiding could generally be considered as a secondary parameter when compared to debris formation. However, a high void will probably entrain a smaller fragmentation so that the melt might form a cake at the basemat with low possibilities of cooling.

Secondly, there is the possibility of entrainment of part of the melt out of the cavity which would have a positive effect regarding debris bed formation and a negative effect for DCH.

Currently, no simple model can be given for the evaluation of the void in a mixture.

C - Melt solidification

Solidification of the melt is a positive point for:

- Explosion: a partially solid melt limits the fine fragmentation implied by explosion,
- Debris bed formation: the bed will be made of particles that are more easily coolable.

Given the fragmentation characteristics and the voiding (i.e. the ambient condition around the drop), the heat transfer can be characterised and thus solidification aspects can be qualitatively characterised. However, several difficulties are encountered in both theory and modelling. First, the real material in the reactor situation is a very complex mixture containing a large part of the periodic table and the actual properties for both the solidification process and subsequent behaviour are unclear. Second, it is not clear (even for simple materials used in experiments) that the situation evolves under equilibrium conditions due to the rapidity and intensity of the cooling.

The situation might be complicated by eventual changes in material properties due to oxidation as well as the eventual heat of reaction, which can potentially be of the same order of magnitude as the initial enthalpy.

4.4.3.2.2 Steam explosion

Steam explosion is a complex phenomenon induced by the very fast transfer of heat from a hot fluid (melt) and vaporisation of a volatile second fluid (the coolant). A significant amount of energy can also come from the oxidation of metallic components that have a strong reaction heat, such as zirconium. The pressurisation can occur if the heat transfer (and energy increase in case of oxidation) process is very fast, through two distinct processes:

- A fine fragmentation of the melt,
- The heat transfer related to the fine fragment and the associated vaporisation of coolant.

Although the global phenomenology is well understood, there are still some important unknowns.

<u>The fine fragmentation</u> can occur with the help of two main classes of processes: thermal or hydrodynamic. Thermal fragmentation is a phenomenon which involves the total fine fragmentation of a hot liquid drop, under the action of a weak destabilisation of the vapour film around the drop that would entrain local high heat transfers, and pressurisation that would in turn destabilise the droplet.

It is generally agreed that thermal fragmentation can only have an impact at the beginning of the process and very rapidly the hydrodynamical fragmentation process is overwhelming. The main codes IDEMO (IKE/GRS), MC3D (IRSN/CEA/EDF) and JASMINE (JNES) use a fine fragmentation model based on hydrodynamic fragmentation, although some codes (TEXAS, ESPROSE) have been used to model thermal fragmentation in more detail.

<u>Regarding the heat transfer and pressurisation processes</u>, there is still uncertainty and a lack of consensus which is reflected in the code. Some codes are based on the "micro-interaction model", which states that the pressurisation is due to the heat-up and dilation of a fraction of the water which is entrained and in equilibrium with the fragments (ESPROSE, IDEMO). The second category of codes (e.g. MC3D, TEXAS) hypothesises a strong non-equilibrium between the melt and water, which directly creates a void (e.g. film boiling at the fragment surface). This void is responsible for the pressurisation and the water heat-up has a low impact.

If the difficulties can be considered as important for the evaluation of the pressure loads, a more significant problem comes from the high sensitivity of the phenomenon to the initial conditions.

<u>The triggering of an explosion still appears as a stochastic process,</u> at least from the point of view of applicability to real situations. The thermal fragmentation phenomenon, that might be one of the triggering processes, is a good example of such apparent stochasticity. Although it has been possible to characterise the phenomenon at the laboratory scale for certain conditions, the difference between the explosion and non-explosive situation is so small that it is unclear how to characterise the triggering at reactor scale.

The strength of the explosion is likely to be linked to the explosivity, i.e. the ability for a mixture to be triggered[3]. Thus, a conservative way to handle the problem is to consider that triggering is related to the strength of the potential explosion. Therefore, evaluation of the strength for a given situation (by numerical simulation) could give an evaluation of the explosivity, although this implies triggering in all cases which is extremely conservative.

Given these intrinsic difficulties, the modelling of steam explosion is complex due to solidification and oxidation. The stability of a drop with a crust submitted to a shock wave has been the subject of some investigations but the approaches for applications are still quite rudimentary. Oxidation during an explosion has been investigated experimentally [151] and it has been shown that, at least for zirconium, the oxidation can be nearly complete during the time scale of the explosion. The impact on the explosion loads is only related to the extra energy that is introduced by the oxidation reaction. The oxidation of iron can also be relatively important but the impact on the energy has not been quantified to date. However, the reaction heat of oxidation of iron is approximately 10 times lower than that for zirconium and the impact on the explosion energy is believed to be small.

4.4.3.3 Modelling of FCI for L2PSA

A – Simulation tools

The FCI problem is now studied through the development of dedicated complex multidimensional fluid dynamics (CMFD) tools, aiming to overcome the lack of experimental data.

CMFD tools have the ability to give qualitative pictures, although they are difficult to build because the many models required for the numerous interactions between fluids can lead to numerical instability and misleading conclusions.

Some simplified tools already exist, particularly for the explosion phase although these have the drawback of large uncertainty regarding the premixing, in particular void and solidification.

B - Expert judgment and arbitrary global probability

This method is generally followed when no tool is available, or for simplified L2PSA. However, this method should be used with caution, ensuring that the investigation is appropriate for the judged situation. For example, a group of experts (SERG[4]) [152] examined the question of containment failure through the α-mode. In this scenario, an in-vessel steam explosion could lead to the detachment of the vessel head which would itself hit the containment. The overall conditional probability for this failure mode was estimated to be between 10^{-4} to 10^{-2}. However, this value (proposed only for a specific event) was quickly used for estimating the probability of occurrence of a steam explosion regardless of the situation.

C – Methodology for the use of best-estimate CMFD codes

The use of CMFD codes leads in general to a best-estimate evaluation with some restrictions dictated by the feasibility i.e. capabilities of the codes regarding the models and CPU possibilities. The qualification of the

[3] This is however not totally clear as very high explosion strengths were recorded in nearly saturated conditions whereas spontaneous explosion was never obtained in this condition.

[4] Steam Explosion Review Group, A review of current understanding of the potential for containment failure arising from in-vessel steam explosion. NUREG-I 116, 1985.

code must be established to estimate the uncertainty and accuracy. In any case, a specific methodology is needed for the application of CMFD codes for L2PSA. Two examples are provided:

a/ One possibility, to screen all the involved phenomena is to use the ROAAM (Risk Oriented Accident Analysis Methodology)[5] [153] which is described in more detail in § 3. In this method, the phenomenon is decomposed into sub-phenomena that can be solved independently. An example is the study of lower head failure by Theofanous et al.[6] [154] (see figure below). At each step, a Probability Density Function (PDF) for the input is combined with the “Causal Relationship” (CR), derived from the evaluation for each sub-phenomena, to get another PDF as an output (input for subsequent sub-phenomena). Note that in this particular example, the ROAAM process was not completed since it was estimated that in any case the lower head would stand the loads, thus no output probability was necessary (although an arbitrary probability of 10^{-3} was assigned corresponding to an event judged “physically unreasonable”).

b/ A second possibility was utilised in particular at IRSN for the analysis of the ex-vessel steam explosion analysis for French 900 MWe reactors. This method is simpler as it consists of a full integration of the module as shown in Fig. 38. With a set of initial conditions at the time of vessel failure, the module outputs only the final result e.g. the risk of containment failure. Some intermediate outputs can also be given. This method avoids the difficult task to characterising probabilistically each of the sub-phenomena. The integration of the module in the global probabilistic framework is also simplified. The disadvantage is that the method is less modular, each step being fully dependent on the preceding ones. In ROAAM methodology, each step is independent and can be re-evaluated.

[5] Recent developments in level 2 psa and severe accident management , NEA/CSNI/R(2007)16 http://www.nea.fr/nsd/docs/2007/csni-r2007-16.pdf

[6] T. G. Theofanous, W. W. Yuen, S. Angelini, J. J. Sienicki, K. Freeman, X. Chen, T. Salmassi, Lower head integrity under steam explosion loads, Nuclear Engineering and Design, Volume 189, Issues 1-3, 11 May 1999, Pages 7-57,

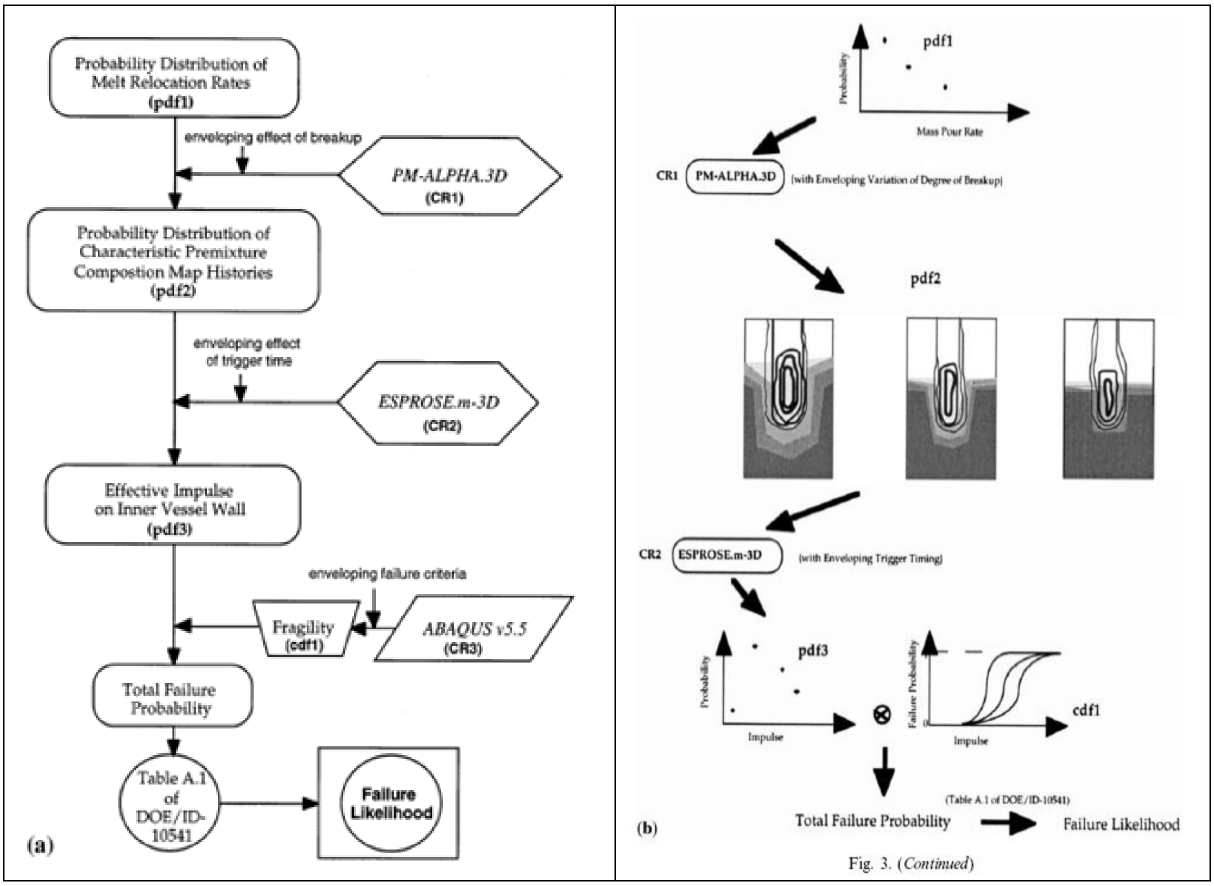

Fig. 37 : Schematic diagram of probabilistic framework used in the analysis of lower head failure by Theofanous et al.

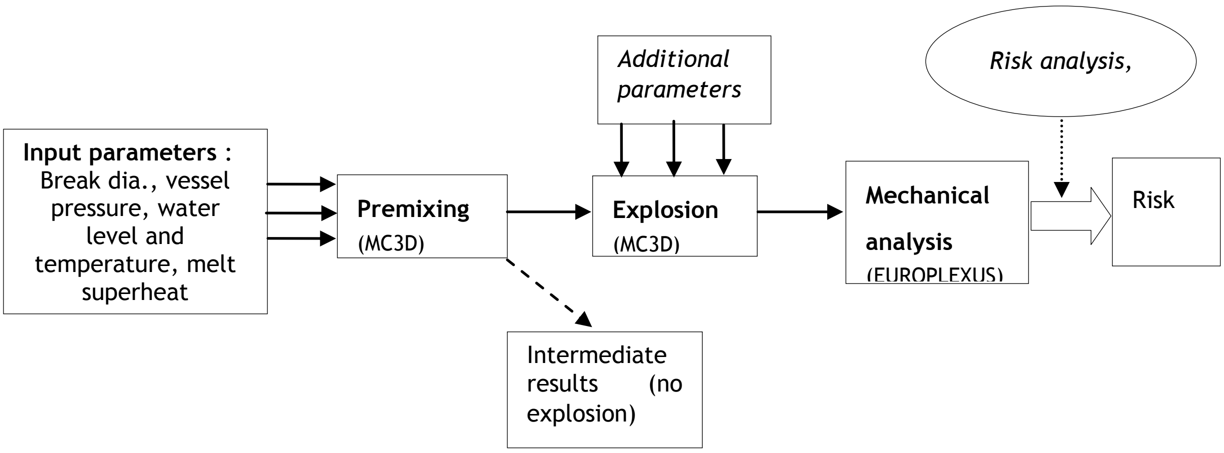

Fig. 38 : Diagram of the integrated analysis of ex-vessel steam explosion at IRSN for the French 900 MWe reactor

4.4.3.4 Application to in-vessel situation

4.4.3.4.1 Potential impact of a steam explosion in the lower plenum

A potentially energetic steam explosion can occur during the relocation of the melt to the lower plenum.

The most threatening consequence of an in-vessel steam explosion would be the α-mode failure of the containment (Fig. 39). This failure mode occurs if the vessel head is detached from the vessel body due to an energetic steam explosion, subsequently hitting the containment and inducing large containment failure. The event is quite complex: the explosion in the lower plenum would push the lower plate, remaining liquid/solid core and the internal structures which would in turn impact the upper head. The bolts might fail and the upper head would be ejected under the effect of the pressure to directly hit the containment.

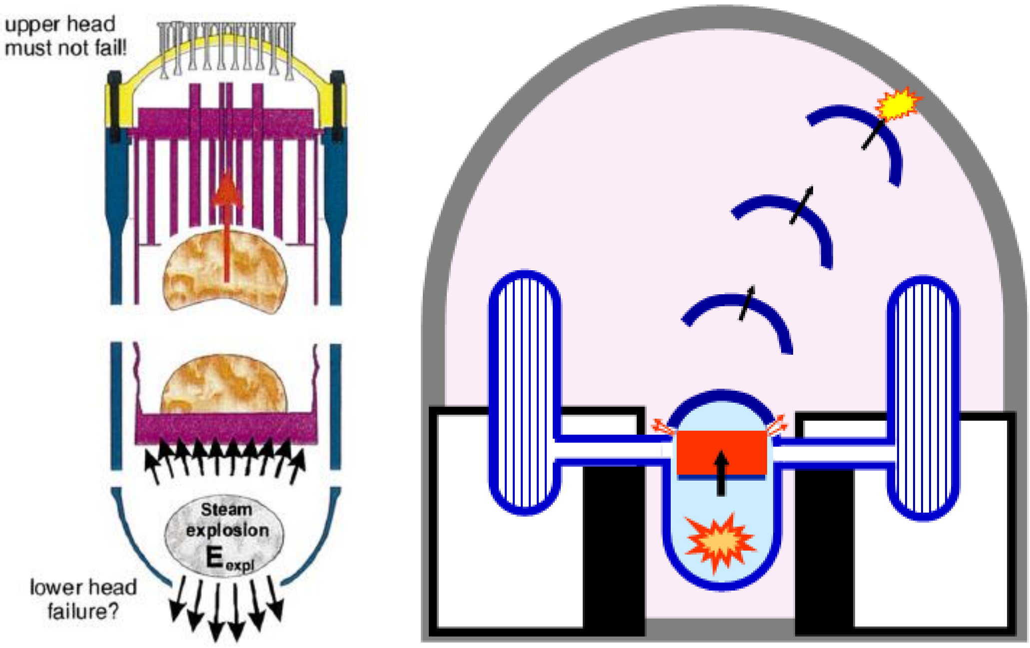

Fig. 39 Potential effects on the reactor vessel of a steam explosion in the lower head plenum (left drawing from [156] [8])

This catastrophic scenario motivated numerous analyses worldwide for many years, particularly in the US. Important studies were made in particular by Theofanous and co-worker (see e.g. [7]). Although the studies were done with some questionable assumptions, the probability for α-mode was found small if not "physically unreasonable".

[7] T. G. Theofanous, W. W. Yuen, The probability of alpha-mode containment failure, Nuclear Engineering and Design, Volume 155, Issues 1-2, 2 April 1995, Pages 459-473

More recently, an analytical program was conducted in Germany (BERDA[8]) to investigate and quantify the mechanical effects and the necessary energy.

Although the probability for α-mode failure is low, even a low probability may be risk relevant due to the potentially large consequences. Therefore in-vessel steam explosion analysis should be taken into account in L2PSA.

Apart from the α-mode failure, there are additional issues to be considered due to rapid interaction between corium and residual water in the lower plenum:

- The RPV lower head could be threatened and fail by mechanical impact,
- A pressure surge could develop and threaten already weakened primary piping.

In addition to the direct consequences of a steam explosion, the melt relocation and eventual explosion might require precise evaluation in principle since:

- The properties (particle fraction and size, porosity) of debris in the lower plenum will depend on the FCI whether explosive or not. A fine particle layer may be formed from an explosion, resulting in an uncoolable debris bed. In contrast, an energetic interaction may also lead to a dispersion of melt;,
- Hydrogen will be produced by the interaction, particularly in an explosive phase.

4.4.3.4.1.1 Specifics of in-vessel FCI

FCI in the vessel should occur at the time of relocation of the melt in the lower plenum. This event is particularly complex and uncertain[9] [157]. This may occur either from the bottom centre of the melt pool (Fig. 40), or, as in TMI-2, laterally (Fig. 41). In this case, the melt might flow directly through the internals of the core with a very complex path, either through the core barrel or via the down-comer.

Unless the melt comes from the downcomer, it has to flow through various supports below the core which is complex to evaluate. The OECD SERENA project[10] [158] recognised that the simulation of lateral flow is complex and requires 3D modelling, therefore a 2D situation with central flow was selected for code comparison. The maximum pressure on the vessel ranged from 10 to 120 MPa, highlighting the high uncertainty attached to these evaluations. It was however recognised that such load levels would not challenge the integrity of the vessel in the absence of pre-existing thermal loads due to the very short duration of the pressure peak. The SERENA Phase 1 members concluded that the safety margins for in-vessel FCI are large enough to encompass any possible underestimation of the predictions.

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[8] R. Krieg et al. 'Load carrying capacity of a reactor vessel head under a corium slug impact from a postulated in-vessel steam explosion', Nuclear Engineering and Design, Volume 202, Issues 2-3, 1 December 2000, Pages 179-196

[9] N. Reinke et al., 'Formation, characterisation and cooling of debris: Scenario discussion with emphasis on TMI-2',

[10] SERENA, OECD Research Programme on Fuel Coolant Interaction, Final report, NEA/CSNI/R(2007)11

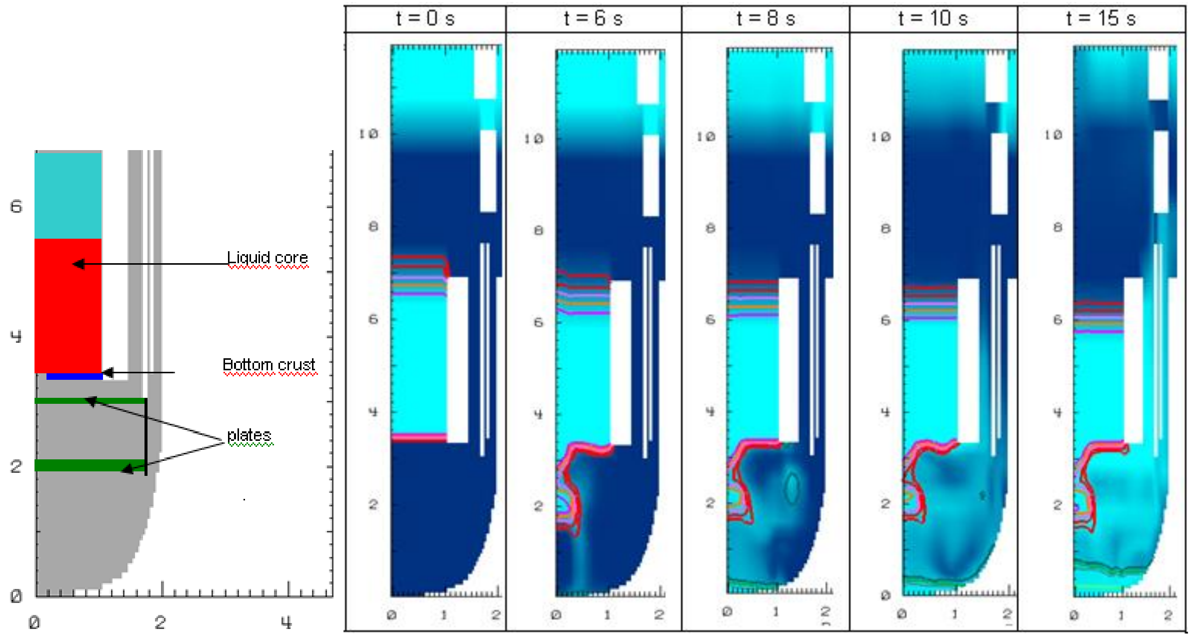

Fig. 40 Example of 2-D evaluation of a central melt down with the MC3D code.

The background colour indicates the liquid volume fraction (darker means more water), the coloured lines indicate the volume fraction of the melt. The core is supposed to be reflooded just before vessel failure.

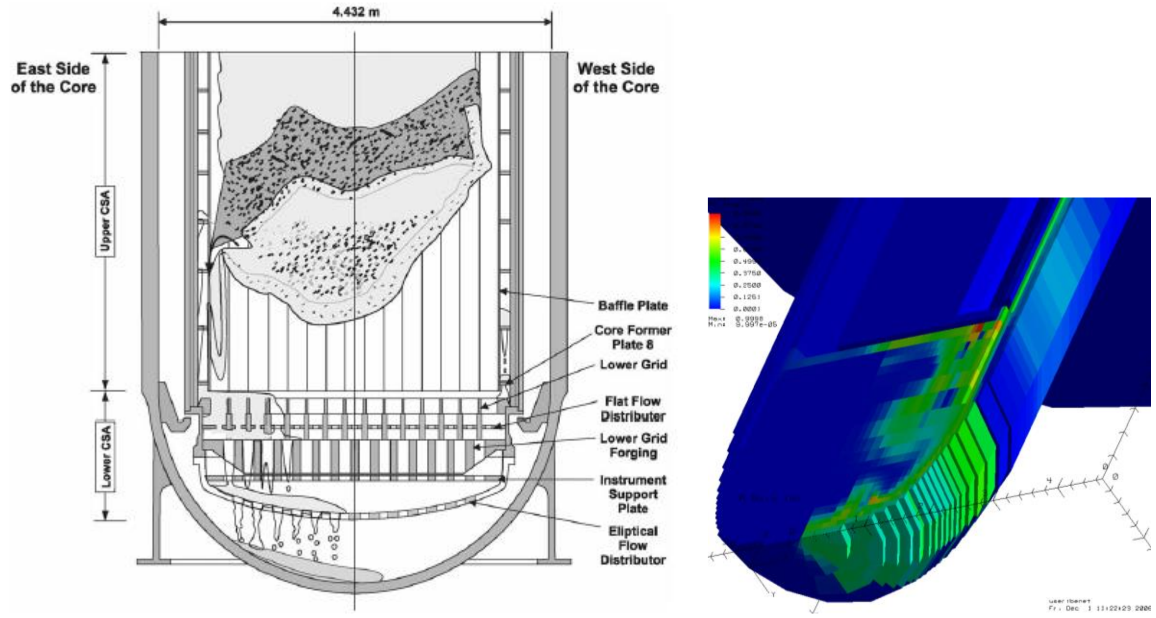

Fig. 41 Left: postulated scenario of TMI-2 corium relocation onto the lower head (from [157][9]). Right: example of 3D calculation of a lateral melt relocation with MC3D. Background colour indicates the volume fraction of the melt.

However, the SERENA project examined only one potential situation, which involved a specific pre-dispersed melt. Also, the codes used were at various levels of sophistication and qualification which makes a direct comparison misleading and the user effect (interpretation of how to handle the problem) may be an important issue.

It is clear that a high uncertainty should be attached to the SERENA evaluations and conclusions.

4.4.3.4.2 α-mode failure

A typical event tree for the α-mode failure is given in Fig. 42. It is seen to be very complex and contains a large number of large uncertainties.

The Steam Explosion Review Group (SERG) provided assessment for the conditional probability of containment failure due to a steam explosion[4] [152]. The SERG consensus is that the occurrence of a steam explosion of sufficient energy which could lead to α-mode containment failure has a low probability (evaluated from 10^{-2} to 10^{-4}). However, there is little support for quantitative assessment which is required in a PSA.

The BERDA facility was a scaled reproduction of the upper part of the vessel [8] [156]. The objective was to identify the admissible energy the vessel head would withstand from the impact of a slug, depending the presence and state of upper internals. In the absence of internals (all melted), the admissible energy was estimated to range between 0.1 and 1 GJ and with intact upper internal structures, the admissible energy estimated to range between 1 and 7 GJ.

A recent SARNET survey[11] [159] provides a little information on the approach of some SARNET partners. Partly based on this survey, Table 28 provides a short synthesis of methods for available recent analyses.

Table 28 Methods for in-vessel fuel coolant interaction

Organisation	Treatment of the phenomenon in the accident progression event tree
NRC[7] [12]	L2PSA studies for PWRs and BWRs. Analyses based on a ROAAM methodology with expert judgment and the supporting use of PM-ALPHA and CHYMES codes for the premixing and ESPROSE.m for explosion in [7] [155] and TEXAS-V in [160] [12]. An important idea is that “an explosion energetic enough to produce an upper-head-threatening missile should be able to fail the lower head that contained it in the first place”. The lower head failure was estimated to occur for a energy yield of 1 GJ. Such failure provides a significant mitigation of energy for the upward missile.
AEA/NNC [13]	Expert judgement based on an extension of the ROAAM model similar to those for NRC (Fig. 42).
IRSN	Steam explosion inside the RPV of 900 MWe PWR, with different consequences: lower head rupture, vessel head rupture, primary system rupture due to overpressurisation or water plugs propagation. First study with simplified tools. Second study with mechanistic code MC3D.

[11] SARNET-PSA2-P08 Revision 0, «Comparison of partners methodologies for level 2 PSA development”, 2005

[12] H. Esmaili et al., ‘An assessment of steam explosions-induced containment failure for Beznau and Leibstadt nuclear power plants’, ERI/HSK 95-302

[13] B.D. Turland et al., ‘Quantification of the probability of containment failure caused by an in-vessel steam explosion for the Sizewell B PWR’, Nuclear Engineering and Design, Volume 155, Issues 1-2, 2 April 1995, Pages 445-458

	Mechanical consequences assessed using specific simplified modelling based on an extrapolation to reactor size of BERDA experiments results (see above). Upper head failure predicted only in the absence of upper internals (high temperature upper head). The conservative assumption was made that a failure of the upper head would automatically lead an impact of some object on the containment wall.
FRAMATOME	Expert judgment (following SERG2 review) or application of FZK results: comparing generated loads (based on the extent of coarse fragmentation, ECO experiments) with the load bearing capacity of the RPV (based on BERDA experiments) applying MC techniques. As consequence of a steam explosion exceeding the load bearing capacity of the RPV a large (1 m^2) containment failure is assumed (due to impact of the vessel head).
VEIKI	Assessment of FCI on the basis of experimental knowledge and engineering judgment; RPV failure cannot be excluded in case of ECCS recovery or ex-vessel cooling (which is part of the accident management strategy); steam explosion not credited

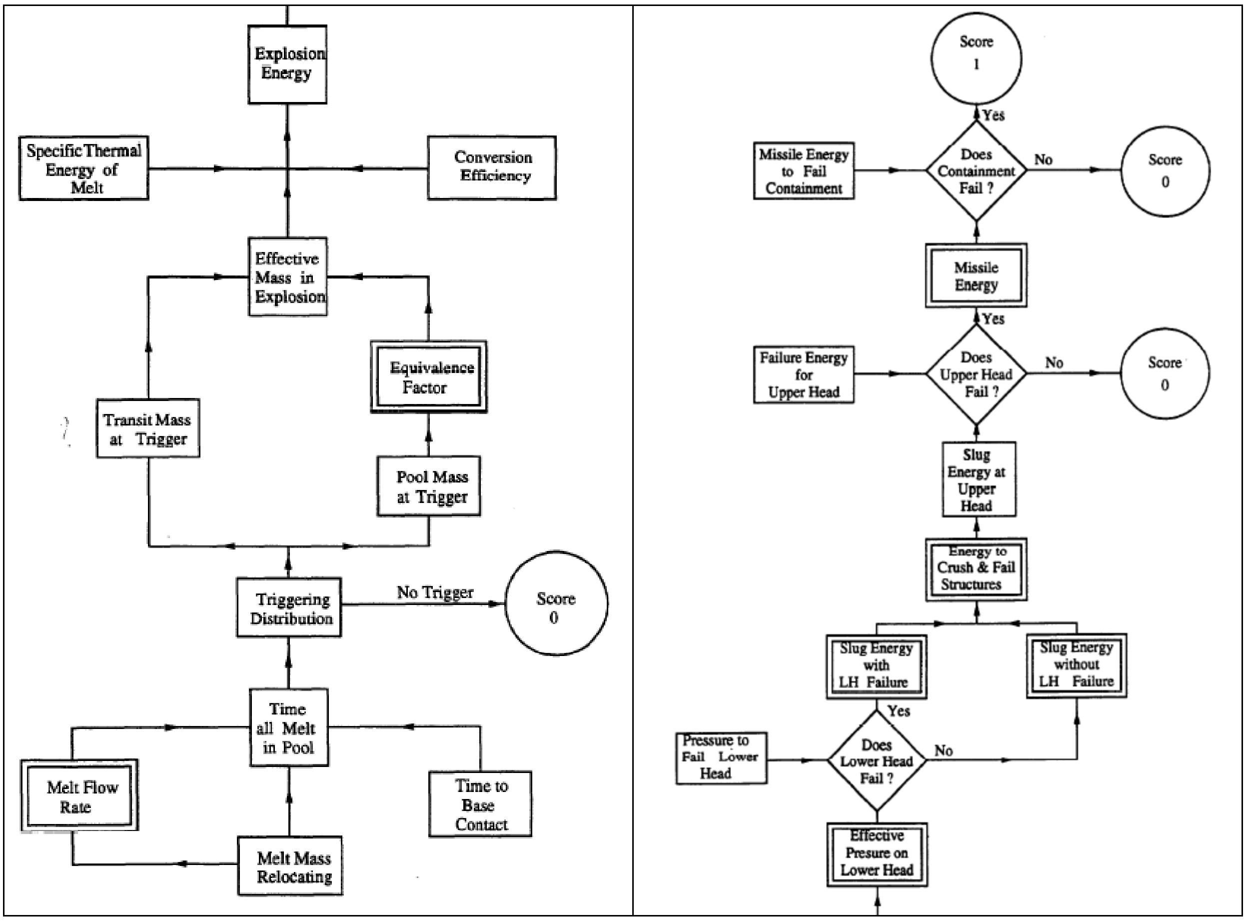

Fig. 42 Typical vent tree for the a-mode failure (from [161] [13])

4.4.3.4.3 Vessel lower head failure

Regarding lower head failure, a large number of tools for evaluating the vessel deformation from a given load are available and general confidence can be considered as good. However, the knowledge of the characteristics of the vessel itself (steel properties) is not always clear. For example, some scenarios may lead to a hot part of the vessel with low resistance but perhaps a higher strain capacity. The impact of aging on steel properties should also be examined when assessing the vessel behaviour.

In the case of an intact vessel (no thermal load, no impact of aging), various evaluations show that the vessel should withstand (i.e. high deformation but no rupture) peak pressures of about 100 MPa (for example the maximum in SERENA calculations was 120 MPa).

4.4.3.4.4 Other potential effects of in-vessel FCI

The impact of an explosion on the pipes or steam generators seems very difficult to handle, as FCI codes cannot easily be used due to limitations in geometrical representations which would require high CPU capabilities. This was attempted in the IRSN L2PSA study for 900 MWe French PWRs but the results were inconclusive. With the development of computer capabilities, investigation of more complex geometries should be possible in the near future. However, in the current frame of a PSA, it may be better to use expert judgment based on the loads propagating the vessel.

Regarding the debris bed formation, the code capability is adequate although, only scarce information can be obtained for the formation of the debris bed itself[14]. In general, evaluations with MC3D predict only a limited fraction of solid dispersed debris. As seen in Fig. 40, even when the initial amount of water is high due to the long pour, there is a tendency to produce high void after some time, and the melt finally flows with a limited fragmentation (this was the case of TMI-2, since the average porosity was only 18 %). A priori, only a small melt relocation flux can lead to a coolable situation. Although there are still some uncertainties, an adequate geometrical representation of the problem should show reasonable agreement between the premixing codes, although no specific comparison of codes currently exists. The behaviour shown for example by MC3D (Fig. 40) is driven primarily by the high vaporisation and water entrainment in the long term, thus minimising the impact of potential uncertainties.

4.4.3.4.4.1 PSA approaches

Although the steam explosion issue has been studied for decades numerous uncertainties remain showing the difficulties to model and predict this phenomenon. However, the last international comparison exercise undertaken during Phase 1 of SERENA [147] concludes that the safety margins for in-vessel FCI are large enough to encompass any possible underestimation of the predictions. Therefore it is possible to consider the in-vessel

[14] A status of the problem and capabilities of IKEMIX and JASMINE codes can be found in a special issue of Nuclear Eng. And Design on debris bed coolability, Volume 236, Issues 19-21, Pages 1937-2328 (October 2006)

steam explosion issue based on the final conclusion of the SERENA phase 1 program. For example in Belgium L2PSA (Tractebel), in case of pre-existing thermal loads, the assessment of the in-vessel steam explosion risk is made by expert judgment using the latest literature available. The consequence of an in-vessel steam explosion is a large vessel failure and α-mode failure is not credited.

Alternatively there are specific simulation codes for steam explosion analysis which could be applied in principle for PSA purposes. However, even for state of the art codes which can model reactor specific conditions, their application within a PSA is difficult in practise. The codes require initial and boundary conditions from preceding phases of the accident, but the existing models for core relocation into the lower plenum are rough and the possible variations due to slightly different accident sequences are immense. The application of simulation codes is therefore focussed on exploring upper limits for steam explosion energy rather than on determining a realistic distribution of various sequences. However, if such analysis shows that even under pessimistic assumptions there is no threat due to steam explosions, further efforts within a PSA will not be necessary.

Regarding the specific α-mode failure there is a consensus for a very small probability ("physically unreasonable") and some PSA simply assign a low probability to α-mode containment failure without further analysis. Such an approach may only be justified if the plant under consideration is subject to a high probability for large early releases due to other mechanisms, e.g. containment bypass or hydrogen combustion.

However, the most likely situation to be encountered is that a threat (vessel failure, lower or upper head) due to in-vessel steam explosion cannot be ruled out completely, and that there is a need to at least roughly determine the associated probability. There is also a need for analysis to assess the impact of these failures on the containment itself.

However, considering:

- The high difficulty of the problem,
- The lack of availability of dedicated FCI tools and the expertise that they require,
- Uncertainty at various stages of the analysis, from the evaluation of the flow rate to the determination of load bearing capacity of the structures (including effects of temperature and/or aging);

it seems difficult to propose a general method.

Besides a simplistic approach with a fairly arbitrarily small probability, a potential second level approach would be the one followed by AEA for Sizewell-B using simplified tools or expert judgement. However, a precise evaluation with a discretisation of events as shown in Fig. 44 can then be quite costly and will inevitably lead to large uncertainties.

With the development of computer codes, it is now possible to integrate parts of the event tree to reduce as much as possible the different steps. As an example, a "missile" module was developed in MC3D to directly evaluate the load yielded by a slug on the vessel head. In such a case, the event tree is very simplified (Fig. 43).

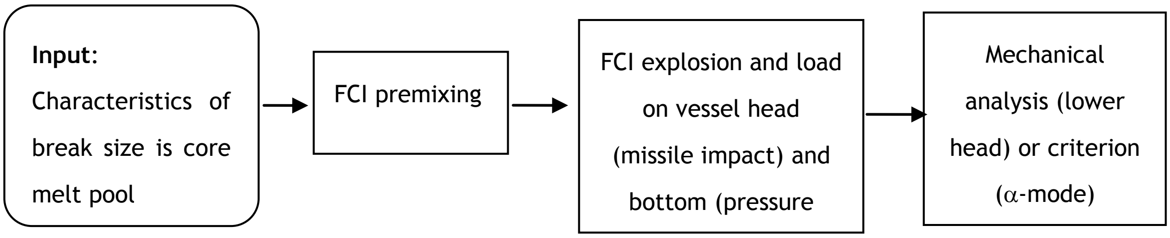

Fig. 43 Simplified event tree used in the last IRSN studies

4.4.3.4.5 Example from L2PSA for PWR by GRS

GRS has performed a L2PSA for a modern German PWR [148]. Within this PSA a simplified model has been integrated into the event tree analysis. The model is based on the following principles:

- The mass flow of corium through the grid plate is based on MELCOR core degradation calculations and on pessimistic assumptions about the failure of the grid plate,
- Breakup of corium jet and mass of reactive corium in the lower plenum were based on a parametric model. Parameters for this model (including uncertainty bands) have been provided by a steam explosion expert,
- The maximum theoretical isentropic work potential of a mixture of corium and water is 840 J/g if it could expand without any obstacle down to 1 bar final pressure. It has a weak dependency on the corium/water volume fraction in the mixing zone [149].
- If the steam explosion is extremely energetic so that it fails the RPV lower head, only a weak force remains for loading the upper head.
- The ratio of the real isentropic work potential for an expansion down to 1 bar final pressure to the maximum theoretical isentropic work potential is between 0.05 and 0.4. (This value is not to be confused with experimental data for steam explosion “efficiency” which are mostly related to the thermal content of the corium),
- Since the RPV volume is limited, the expansion process cannot proceed down to 1 bar. The expansion process is limited by the RPV volume, so that the real isentropic work potential cannot be realised.
- It is necessary to assume the acceleration of a massive slug above the expanding reaction zone to be able to create significant loads to the RPV head. At the same time the gas volume above the slug is being compressed by the slug, so that the slug impact on the RPV head gets smoother,
- The coupled expansion in the reaction zone and the compression in the upper RPV zone is calculated assuming polytropic gas law. The polytropic coefficient is in a range between 1.4 (adiabatic) and 1.0 (isothermal),
- The maximum energy of the slug (expansion force from bottom minus deceleration from top) is compared to the mechanical load capacity of the RPV head,

- The mechanical load capacity of the RPV head is based on experimental results [150]. It depends on the effects of the upper core structures (control rod drives etc.) and their temperature. A typical value is 1 GJ of mechanical energy for a large PWR vessel.

With this model, a zero probability has been calculated for RPV head failure in all Monte Carlo samples. Consequently α-mode failure has been excluded for this reactor. RPV lower head failure occurred with a conditional mean probability of less than 1% and induced failure of hot leg was at about 2% for cases with high initial RPV pressure.

4.4.3.5 Application to ex-vessel situation

4.4.3.5.1 Events to be considered in the ex-vessel situation

FCI occurs if the vessel fails and delivers melt into the reactor cavity which contains water.

Fig. 44 summarises the situation. The coloured parts of the diagram are considered to be part of the FCI problem. Blue concerns the premixing, red the explosion and green is for both. It is seen that FCI includes the mechanical aspects regarding the structure behaviour. The black/grey parts are the entrance and exit of the FCI problem.

In contrast to FCI, due to the larger time scale, the impact of quasi static containment pressurisation can be deduced from other studies (e.g. for global containment pressurisation, combustion) through containment fragility curves. However, the impact on the internal structures due an explosion is specific to FCI and must be treated accordingly (although methods could be common to other issues).

Slow mixing, generally called premixing, has mainly three impacts:

- Slow global pressurisation of containment (slow means at a rate of some seconds) through:
  - Vaporisation of the coolant,
  - Oxidation and hydrogen production,
  - Probable combustion,
  - Probable flashing of the water present in the primary circuit and ejected through the break.
- Dispersion of part of the melt out of the cavity,
- Debris bed formation.

The question of pressurisation is clearly related to the DCH phenomena in presence of water.

The occurrence of steam explosion, i.e. its triggering, cannot reasonably be predicted, although it is considered that there are situations more explosive than others. One conservative method is to consider that an explosion can be triggered at any time.

The steam explosion has several effects:

- Strong, short loads on internal structures (cavity wall) that might lead to a loss of structural integrity, disequilibrium and displacement of heavy materials (such as SG and vessel itself) with direct impact on the containment wall or walls,
- Contribution to global pressurisation of containment,

- Fine fragmentation and impact on debris bed characteristics,
- Stronger dispersion of melt and water outside the cavity.

Although a steam explosion might lead to a resuspension, it is unlikely that a steam explosion leads directly to an additional fission product release (in the time scale of explosion).

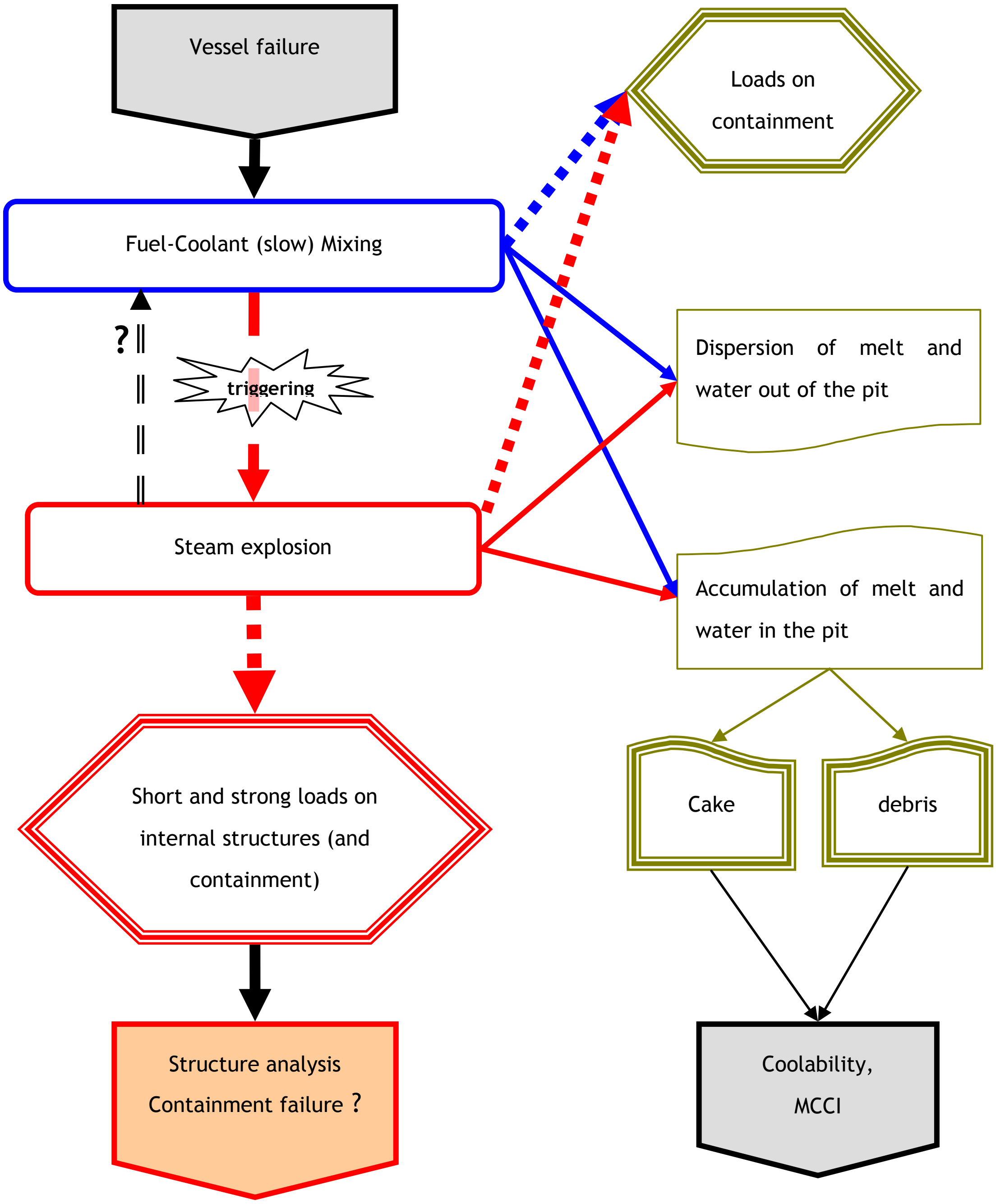

Fig. 44 Diagram of events and consequences in an ex-vessel FCI.

Some experiments show occurrences of double steam explosions (FITS [162][15]) and one could consider successive events of explosion. Of course the potential single mass which is available for reaction in one of the successive events is smaller than the potential mass if only a single event occurs. However, no tool is currently able to calculate a premixing from an explosion and only a one-way path from mixing to explosion can be considered (dashed arrow with question mark in graph).

The direct effects of an explosion, or even of the premixing if the vessel pressure is sufficiently high, on the containment, are various and strongly depend on the reactor itself. Generic potential effects include:

- potential vessel detachment under the effect of high pressure in the cavity,
- damage to the internal concrete structure.

In both cases, the primary circuit is displaced and this can lead to direct damage to the containment.

In case of strong explosion, the internal concrete structures (wall, floors) will be damaged and lose their structural integrity:

- the floors can hit the containment wall,
- the heavy materials (e.g. steam generators) can be de-stabilised and hit the containment.

4.4.3.5.2 Code capabilities

A - Inputs

The general input conditions for the calculation of FCI loads are:

- In vessel conditions:
  - Pressure,
  - Melt mass and composition (initially liquid part):
    - Oxidic,
    - Various metallic,
  - Melt energy (potentially several layers),
  - Amount and temperature of water,
  - Gas composition and temperature,
- Break characteristics: section, position,
- Cavity geometry, level of constraint (availability of exit paths from the cavity),
- Water level and temperature,
- Containment pressure,

Regarding the code capabilities and qualification, the ex-vessel studies require design specifics which make calculations quite uncertain, particularly due to the general knowledge and availability of experimental validation for the following issues:

- Possibility of very large break,
- Possibility of pressurised ejection (even a few bars are important),

[15] E.g. N. A. Evans, D. 8. Mitchell, L. S. Nelson, M.L. Corradini, “Recent results prom the SANDIA steam-explosion program” SAND8202269C,

- Non central position of break (probably), not strictly vertical jet,
- High amount of metallic components with high energy of oxidation,
- Ejection of gas if the vessel is pressurised,
- Possibility of water over the melt in the vessel (TMI-2 situation).

It can be emphasised that if the premixing involves a high velocity melt jet, as might be the case under ex-vessel conditions, fine fragmentation might occur during this stage. The difference between mixing and explosion is in the timescale of the event, driven by melt injection in one case and pressure wave propagation in the other. However, if the mixing is constrained by geometry (none or small venting paths), a high pressurisation can occur and there is high uncertainty regarding the behaviour of mixing and triggering of steam explosion in such conditions.

B - Code functional limitations

Several input conditions are very uncertain; particularly break location and size with experiments showing that all sizes might still be envisaged. However, it is unlikely that the break will be precisely round and axisymetric, which is the (non-conservative) case for all available FCI experiments. A side break would involve a higher ratio (interfacial area)/(break section) for the jet, i.e. an increase of the coarse fragmentation and then of mass in mixture. However, 3D evaluations are impractical as a L2PSA should envisage many different situations. Increases in computational performance should help to make 3D calculations achievable but then the question of validation will still be open whilst no experimental data are available.

Still concerning the question of geometry, the BWR case has the additional complexity of steel plates below the vessel. Similarly to the in-vessel case, the melt relocation occurs though some complex paths which, unless using very fine mesh, might be difficult to simulate.

A further difficulty comes from the use of representative corium in calculations. The tendency for the experiments to date has been to use “prototypical” corium (purely oxidic UO_2/ZrO_2 which is not necessarily representative or conservative), although the SERENA-2 project plans some experiments with alternative corium.

The presence of non-oxidised metals is very important, particularly zirconium for which the energy of oxidation is very high. The ZrEx/ZrSS experiments [151],[152] have clearly shown that zirconium has the ability to be almost totally oxidised during the explosion timescale. The high input of additional energy increases considerably the explosion energy. Regarding the specific ex-vessel situation, iron might not be strongly oxidised and may not be significant due to the low energy release of the iron oxidation.

However, the code capabilities regarding oxidation are currently rather limited. Among the 11 codes involved in SERENA-1 project, only 3 codes were able to handle oxidation: IFCI 6, TEXAS-V, and MC3D. MC3D includes a parametric model regarding oxidation itself but is able to qualitatively reproduce the mechanical effect obtained in the ZrEx experiments.

C - Code qualification

One ex-vessel exercise was also performed in the SERENA project using a simplified situation to allow all involved codes to participate:

- central break (diameter = 50 cm),
- gravitational pour,
- partial flooding of the cavity (water level 1 m below vessel),
- no oxidation.

There were issues for the modelling of a 50 cm diameter jet with codes not having the capabilities of representing a jet (but only drops). The large discrepancy obtained in the results highlighted again the differences among codes, the user effect and the effect of uncertainty of some parameters particularly regarding fragmentation. Additionally it was difficult to find a rational in the results, for example the two extreme calculations regarding the loads involved the highest void during the premixing, with one code (MC3D) suppressing the explosion as a result.

4.4.3.5.3 Mechanical issues

Depending on the reactor design, the major potential impact of ex-vessel FCI is the loss of integrity of the concrete structures surrounding the vessel. The potential effects are numerous, from the direct impact on containment wall to the loss of strength for the support of large heavy elements as steam generators.

One of the major problems is the evaluation of behaviour of these structures due to the specific nature of the reinforced concrete. The concrete itself has a very low capacity of deformation whereas steel irons have a significant elasticity. Some studies were performed by using an "equivalent concrete", mixing properties of concrete and steel [163] [16]. In this case, the concrete behaves like an elasto-plastic medium which can withstand relatively large strains. In contrast, IRSN, based on preliminary studies by CEA/EDF, performed a study using a mechanical model which considered the concrete and iron structures separately. A fragile Drücker-Prägger model was used for the concrete behaviour, and the study included analysis of the impact of a visco-plastic regularisation so that the behaviour after the strain limit is more precise. The two types of models lead to very different behaviour but this was found to have a low impact on final risk quantification because the visco-plasticity only acts on the dynamics and not on the final result.

4.4.3.5.4 Examples of recent L2PSA studies

4.4.3.5.4.1 IRSN L2PSA for French PWR 900 MWe reactors

[16] E.g. :

- Cizelj, Končara and Leskovar, 'Vulnerability of a partially flooded PWR reactor cavity to a steam explosion', Nuclear Engineering and Design, Volume 236, Issues 14-16, August 2006, Pages 1617-1627.

- Almström, Sundel, Frid, Engelbrektson, 'Significance of fluid-structure interaction phenomena for containment response to ex-vessel steam explosions', Nuclear Engineering and Design, Volume 189, Issues 1-3, 11 May 1999, Pages 405-422

The study for French 900 MWe reactors was conducted in 2003-2004 with the general integrated method depicted in Fig. 45, using the MC3D code for the FCI modelling and EUROPLEXUS for the mechanical calculations[17]. For these particular reactors, the risk was related to the displacement of the floors attached to the reactor cavity, since a small gap (10 cm) exists between the floors and the containment wall. The risk was established with an index from 0 to 4 and a failure probability with a criterion related to the floor displacement (Table 29). Fig. 46 is an attempt to summarise the developed methodology. Within a pre-defined spectrum of initial conditions (Table 30), 16 cases were selected which led to 16 premixing calculations. For each premixing, between 20 and 40 explosion calculations were performed, each with a different triggering time (regular sampling up to the time of melt ejection of presence of water in the cavity). Each explosion calculation is followed by a structural mechanics calculation to establish the risk for this particular case and time for triggering. The coupling between thermalhydraulics calculation and mechanical calculation allows avoiding the difficult discussion on available energy transmitted to the wall. Finally, a global probability of failure is estimated for each case.

The major simplifying assumptions made for the evaluation were:

- 2-D evaluations with central break and no modelling of the access corridor to the cavity,
- Absence of impact of the vessel insulator,
- No oxidation of the melt,
- Pure UO_2/ZrO_2 mixture properties,
- Parameter setting based on the best representation of available experimental results (FARO, KROTOS).

Table 29 Quantification of risk for containment failure

Risk	Criterion : floor displacement	Meaning	Equivalent conditional probability of failure
0	d < 1 mm	No danger	0
1	1 mm < d < 1 cm	Low danger	10 %
2	1 cm < d < 10 cm	High danger	50 %
3	10 cm < d < 20 cm	Very high danger	90 %
4	20 cm < d	Unacceptable	100 %

[17] R. Meignen, J. Dupas, B. Chaumont, First evaluations of Ex-Vessel Fuel-Coolant Interaction with MC3D, The 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea, October 5-9, 2003

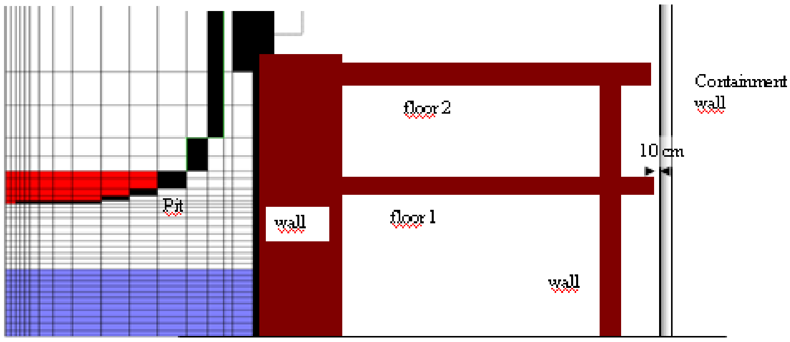

Fig. 45 : Representation of geometry for French PWR 900 MWe reactor analysis

The major conclusions of the study were:

- The exclusion of risk was difficult to demonstrate: despite rather mild interactions in general, the risk is mainly due to the relatively low strength of the structures.
- The water level has a strong impact, the total cavity flooding leading to high constraints and higher loads. In contrast, a low amount of water (up to half of vessel height) represents an acceptable situation.
- The vessel pressure has two compensating effects. Stronger explosions occur at low vessel pressure due to lower melt velocity inducing larger drops and thus lower void and solidification. At high pressure, the explosions are weaker but the global pressurisation during premixing is higher.
- The global melt flow is quite unstable at high water level with a high pressurisation event during the premixing. These events might easily trigger an explosion.

The global L2PSA results have shown the importance of this issue.

For this type of reactor, the activation of the spray system fills the reactor cavity within less than 2 hours and the probability of this penalising situation cannot be easily reduced. The conditional containment failure in case of ex-vessel steam explosion was found to be 1.8 % (fractile 5%), 9.2 % (fractile 50 %) or 18 % (Fractile 95 %). In that case, the containment failure was associated to a small break (~5 cm^2) induced by the global displacement of the structure, including the floor mentioned above.

IRSN was aware of the large uncertainties associated to these results, and considers that specific effort is still needed to solve this issue with implementation (or not) of plant design modification.

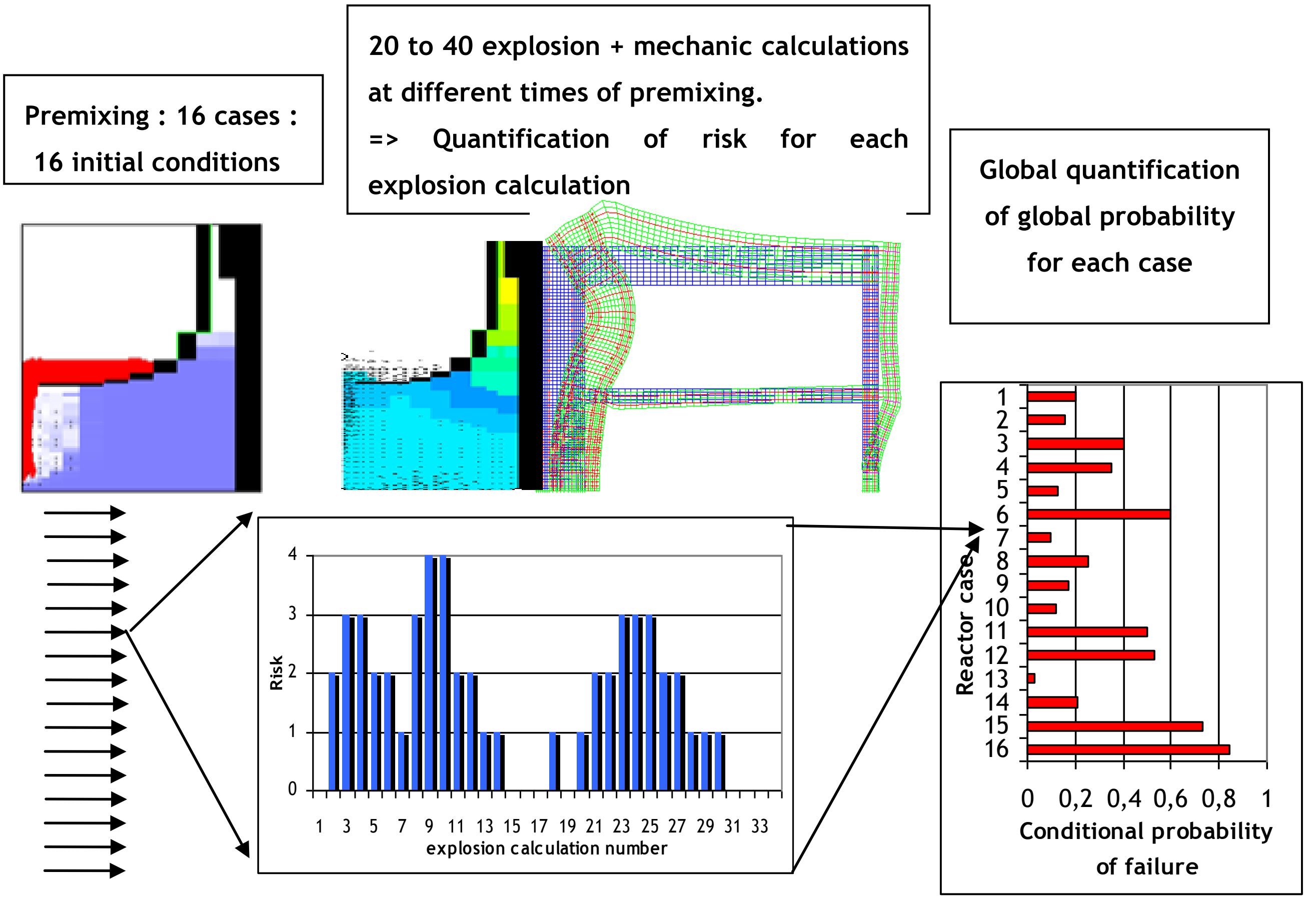

Fig. 46 Methodology for evaluation of risk (PSA-2 900 MWe PWR)

Table 30 Spectrum of initial conditions (PSA-2 900 MWe PWR)

		1	2	3	4
Pvessel (bar) [DP]	Vessel pressure [(Pvessel - Pcontainment]	2 [0]	4 [2]	10 [8]	50 [48]
Djet (m)	Jet Diameter	0.2	0.4	0.7	1.
Dtsat (K)	Subccoling	10	30	50	70
Heau (m)	Water level	2	3	4	5
DTcor (K)	Corium overheat	50	100	200	400

4.4.3.5.4.2 L2PSA for BWR 860 MWe reactors

The Lower Drywell (LDW) compartment below the RPV of BWR was originally designed to remain a dry compartment under normal operation and during design basis accidents. A severe accident management strategy was developed and necessary plant modifications were carried out in the 1980s. The starting point of the SAM is that RPV integrity cannot be guaranteed during core melt accidents. The cooling and stabilisation of core melt is targeted to be reached in the LDW, where a much larger and deeper sub-cooled water pool can be arranged. Additionally, the heat transfer area of the core debris would be larger after discharge into containment. The LDW was back-fitted with a water flooding line between the Suppression Pool and the LDW. The valve closing the flood line will be opened by operator action according to SAM guidelines prior to the anticipated pressure vessel failure. Typically the LDW flooding is initiated if the reactor scram has been unsuccessful and the water level in the RPV downcomer has been lower than 0.7 m above the top of active fuel. The successful flooding of the LDW to a pool depth of 7.8 m takes approximately 1500 s. The flooding water sub-cooling is 40 – 100 degrees and the anticipated pool depth in the LDW is of the order 5 – 10 m at the time of RPV failure. In the benign situation the core melt jet entering the LDW pool will be fragmented and quenched before arrival on the LDW floor. However, the possibility and consequences of energetic fuel coolant interaction has to be accounted for in the L2PSA.

In the two BWR units, steam explosion pressure and impulse loads were investigated with separate effect computer codes (PM-ALPHA for premixing, ESPROSE.m for propagation assessments). A number of calculations were performed to investigate the effects of melt pouring rate (kg/s), water pool depth, water sub-cooling and pressure in the LDW. The baseline assumption for the melt mass and melt jet diameter participating in steam explosion is that RPV failure and melt discharge take place through a lower head penetration. Based on the analyses of steam explosion loads and calculated fragility curve of the LDW walls and boundaries, the equipment door located at the bottom of the LDW was reinforced to withstand estimated loads with good margins. The steam explosion analyses performed assumed that the explosion triggering took place near the bottom of the LDW either by a pulse caused by melt hitting the floor or by rapid condensation of a large steam globule in the pre-mixture that has been accumulating while melt droplets are passing through the water pool. However, the ex-vessel steam explosion issue is not considered fully closed and research in the area is required, particularly for situations where steam explosion would be triggered at higher elevation in the LDW which may affect L2PSA results.

For situations where steam explosion loads do not lead to LDW failure, the melt fragmentation and consequent thermal hydraulic effects were assessed with MELCOR code. The current MELCOR code version is not able to model ex-vessel melt fragmentation, so the Control Function Package in MELCOR is applied to incorporate anticipated non-condensable gas and steam sources originating from fragmentation. The base assumption was that 15 % of the melt energy is consumed for vaporisation of LDW water, 35 % of the melt energy goes to heating of the bulk of the LDW pool over a short period of time (over 1 minute) and the rest of the melt heat is transferred to water over 4 minutes. The production of additional hydrogen during the FCI and premixing is taken into account. It is assumed that all remaining Zr metal in the melt jet would oxidise over a 1-minute interval. The contribution of steam and non-condensable gases from the non-propagating ex-vessel FCI to the containment pressurisation was calculated with MELCOR 1.8.6 code using only the stand-alone containment nodalisation model.

In the L2PSA of the BWR units the steam explosion is modelled in the CET as follows. If the LDW has not been flooded the steam explosion is not possible, but the seal of the personnel access door and the penetrations at the lower part of the LDW will melt, which leads to large leakage to the reactor building. If the LDW has been flooded, the probability of the early survival of the containment is estimated based on following randomised functions:

- Probability of a steam explosion followed by vessel melt through is modelled as a function of the flow rate of the melt jet. The flow rate depends on the number of penetrations simultaneously melting through. However, due to large uncertainty in the flow rate, the probability is modelled with a non-informative distribution [0;1] . If the primary pressure is high, the probability of steam explosion is multiplied by 2,
- Load of impulse due to the steam explosion is modelled with a lognormal distribution with mean 20 kPa·s and error factor 2, based on the calculations performed with ESPROSE-M/PM-alpha code. If the primary pressure is high, the impulse load is divided by 2,
- Strength of the LDW is modelled with a lognormal distribution with mean 54 kPa·s and error factor 1.7 that is based on structural analysis performed with FEM computer code. After strengthening the LDW door structure, the mean value of the impulse load is well below the strength of the structures. The mean impulse strength of the LDW door before strengthening was estimated at only 6.3 kPa·s according to the structural analysis. The weakest point was the door frame that might have caused the displacement of the whole door structure at loads exceeding 6.3 kPa·s. A wide steel collar was welded on the door frame supporting itself on the inner surface of Lower Drywell concrete wall in case of pressure load. With this relatively simple backfitting, the structural strength of the door was enhanced to at least 54 kPa·s.

If the simulation gives a "true" value for the steam explosion and the impulse load exceeds the strength of the LDW, a large leak area from the LDW is set for the gas flow from the containment to the reactor building and the transportation of radionuclides to the environment is considered.

The strengthening of the LDW door decreased the frequency of unfiltered release by 17%. However, the most effective way to decrease the LER frequency was decreasing the probability of the human error to flood the LDW. The frequency of unfiltered release was decreased by 54 % after training of the operators.

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4.5 EX-VESSEL PHASE (MCCI)

4.5.1 Introduction

In case of vessel melt-through, the core debris (corium) and any water, steam and non-condensable gases will be released from the vessel to the containment building. The mass of corium relocated into the containment will depend on the previous degradation process in-vessel, the nature of the vessel breach and the primary system pressure at the vessel failure time. This mass of corium may fall into the reactor cavity or be swept out of the cavity region into the containment building. In both cases, the initial corium configuration in the containment will depend not only on these previous issues but also on other relevant issues related with design characteristics or phenomenological processes in the containment such as the cavity geometry and the nature of the debris interaction with water which may be present in the reactor cavity.

At high pressure of the Reactor Coolant System (RCS), ejection of the molten core debris from the RPV could occur in jet form, causing fragmentation into small particles. Thinly fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes due to Direct Containment Heating (DCH). In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the containment pressurisation loads. The hydrogen in the containment could ignite, contributing to the loads on the containment.

The RPV failure in presence of water within the reactor cavity, either at high or low RCS pressure, could lead to interactions between fuel and coolant with the potential for rapid steam generation or steam explosions.

The contact between molten core debris and the reactor cavity concrete basemat leads to Molten Corium Concrete Interaction (MCCI) if the corium is not adequately cooled, with the consequent decomposition of concrete and challenge to the containment integrity by various mechanisms, including the following:

- Pressurisation to the point of containment rupture as a result of steam and non-condensable gases generation,
- Transport of high-temperature gases and aerosols into the containment leading to the failure of the containment seals and penetrations,
- Containment liner melt-through.
- Reactor support structures melt-through leading to the relocation of the reactor vessel and damage of containment penetrations,
- The production of combustible gases such as hydrogen and carbon monoxide.

Many factors affect the development of the MCCI phenomenon such as the availability of water in the reactor cavity, the containment geometry, the composition, size distribution and amount of the core debris, the thermodynamic condition of the core debris and the type of the concrete used for the basemat construction.

To investigate most of these phenomena, a significant number of experiments worldwide were done in the eighties and late nineties (BETA, SANDIA Large Scale, ACE/MACE experiments) increasing understanding of most of the MCCI phenomenology and assessing thermal hydraulic codes like CORCON and WECHSL. A summary of all

these activities is available in a CEC report in a framework of the MCCI Programme (Alsmeyer et al. 1995), and was used in the development of this guideline item.
In the years 2000, new experiments have been launched among which CCI [167] and VULCANO [168] with the view of studying the 2D ablation effects and the role of the metallic layer[169].

An APET should typically address the following branching points in the ex-vessel phase:

1. Is the corium coolable early after RPV meltthrough? (A typical situation would be the corium spread and water addition in a core catcher)
2. Is the corium coolable late after RPV failure before the MCCI destroys viable structures? (A typical situation would be progress of MCCI in a dry cavity until the corium reaches the sump water, and then further MCCI is prevented due to sump water ingress)
3. When would the pressure limit be reached
    a. for initiation of containment venting
    b. for containment overpressure failure
4. When would containment function be lost by melt attack
    a. by penetration of the containment bottom
    b. by destruction of containment penetrations (sump suction lines, instrumentation ducts, …)

4.5.1.1 Vessel failure mode issue

The pressure in the RPV at the time of vessel failure may have significant impact on the initial corium configuration in containment.
A high pressure melt ejection (HPME) can occur in case of a non-depressurised primary system. The pressurised ejection of molten corium into the reactor cavity, associated with violent gas discharge, will result in dispersal of a significant fraction of the core debris inside the containment and in many cases with a high fragmentation. Both a higher spreading area in the containment and a higher debris fragmentation will increase the probability to have a coolable corium configuration.
Other fast containment pressurisation processes related with the vessel failure time (e.g. steam explosion, direct containment heating and hydrogen combustion) will have a similar impact on the corium distribution in the containment as HPME phenomenon.
With a low-pressure vessel failure the fragmented or molten mass is collected at the bottom of the reactor cavity, through a gravity-driven process of relocation.

4.5.1.2 Mass and properties of corium relocated issue

The mass of corium that would be relocated at vessel failure and its properties will depend on the in-vessel meltdown process, mainly:

- Release of fission products from degraded core reducing the decay heat source remaining in the corium mass (about 25% of the decay heat is lost from the corium as gaseous species, as iodine compounds and nobles gases),
- Timing of thermal-hydraulic phenomena related to the accident conditions, and influence on the structures mechanical behaviour and their failure mode (a short time failure may increase the contribution of solid state in the corium composition, which has a lower thermal conductivity and density by higher void rates than the equivalent liquid state),
- Randomness in molten corium relocation pathways by fluid-dynamic behaviour in complex geometries (different pathways imply different corium flow rates),
- Melting of large structures (i.e. PWR core barrel and lower plate, BWR shroud and control rod housings, and vessel bottom head),
- Degree of oxidation of both fuel cladding and other metallic structures (a lower fraction of zirconium oxidised in the corium will increase the hydrogen and heat production in the corium relocated),
- The presence of water may affect significantly the spreading area of the corium (for example, in the EPR concept the corium spreading phase should occur without any water interaction to maximise the spreading).

All these processes will determine the total mass of corium relocated and its properties, primarily the temperature and corium composition, which can be grouped at:

- T_{corium} < 1700K: corium and structural metals are mostly in the solid phase.

  This situation is likely to be associated with an early vessel failure caused by mechanical stresses and not with the RPV lower head melting (e.g. vessel failure in a high-pressure scenario); low corium temperatures could also be associated with corium quenching promoted by the presence of water within the cavity.
- 1700K < T_{corium} < 2850K: metals are molten, and oxidic corium is solid.

  This situation is likely to occur when the corium has been previously quenched with the residual water in the vessel lower plenum (e.g. station blackout or small LOCA with corium not located in the lower plenum); in BWRs, in particular, control rod penetration can result in an earlier vessel failure by permitting corium discharge before the steel of the vessel melts. In addition, this situation can be associated with an early vessel failure caused mostly by mechanical stresses, and not with a massive lower head melting (e.g. vessel failure in high-pressure scenario); in this case, moderate corium temperatures could be obtained by corium quenching promoted by the presence of some water within the cavity,
- T_{corium} > 2850K: eutectic UO_2-ZrO_2 and structural metals are in liquid phase,
- T_{corium} > 3120K: whole oxidic and metallic fractions are in liquid phase,

  This situation implies that corium has not been quenched, and can be mostly associated with those accidents caused by significant loss of coolant (e.g. large LOCAs); unsuccessful debris quenching within the vessel lower plenum could also permit the formation of a molten pool with overlying water and then, the discharge of corium mostly in liquid phase.

4.5.1.3 Containment geometry issue

At vessel failure time, the corium is dispersed through the available containment area depending on the RPV failure pressure. Based on the total mass relocated and the pressure values that govern the dispersion process, the corium is spread on an effective containment area (the maximum being the available area). The most influential geometric parameter is of course the cavity area available to contain the corium. A sufficient containment area for a coolable corium, relying solely by conduction, is required to maintain the heat flux density significantly below the critical heat flux (CHF). EPRI [166], [170], suggests a reference value of 0.02m^2/MWth (that represents 50% of CHF, assuming cooling only at the top of a flat corium surface by overlying water and the MWth referring to the nominal reactor power). This translates to a necessary cavity area of approximately 60 m^2 for a 1000 MWe reactor. Typical average loads in the cavity due to corium are given in Table 31.

For many existing plants, the available cavity area is not so large. Therefore, the corium cooling could only be possible assuming debris fragmentation - if at all. Moreover, the reference value of 0.02 m^2/MWth could not be sufficient to guarantee the corium cooling in the case of robust crust formation (with insufficient contact with the bulk corium) not permitting sufficient heat exchange within the corium mass below.

Plant type	Corium bed height (m)	Specific power in UO_2 in the reactor cavity at early MCCI (1% rate power) (W/kg)
Traditional PWR 1000 MWe	0.2 – 0.3	~250
Large Advanced PWR (EPR)	0.1 – 0.2	~210
Traditional BWR 1000 MWe	0.2 – 0.7	150-170

Notes:
The range of debris height has been estimated assuming uniform spreading of 75% of in-vessel mass, 50% of bottom head mass, and taking into account two possible material densities: the apparent density of solid corium with 50% of voids, and the full density of molten corium. Large spreading area, satisfying the requirement 0.02m^2/MWth have been assumed (smaller cavities of some existing plants would imply corium beds higher by a factor of 2 or 3.
The percentage of rated power are referred to the total decay power available in the whole fission products inventory for an irradiation time of 3 years, with a power lost of 25% due to volatile fission products, released during core meltdown.

Table 31 Typical average cavity loads from corium

The above considerations are assuming ideal homogeneous situations. Such assumptions lead to the flattest corium levels. Additionally, the PSA has to check the potential for (locally) higher debris accumulation and other obstacles for coolability, e.g. due to:

- Inhomogeneous corium deposition: note that the release from the RPV will probably not be homogeneous, that the released corium is partly solid, and that debris may tend to collect at walls, or build up agglomerations around the remains of the RPV lower head,

- In the bottom of the cavity there may be a sump or other inhomogenities. For the assessment of coolability these local conditions may be crucial. If such a location were not coolable, a local molten pool could develop and eventually spread to areas which otherwise had been coolable,
- Impeded water access: Part of the lower RPV would not be molten and would remain as a deformed structure inside the cavity. Further, debris of RPV or wall insulation may be present in the cavity. It seems to be very difficult to show with absolute certainty that there will be sufficient water access to each part of the corium,

To ensure continuous coolability, it is necessary that steam can be removed and water can be replenished. This of course is very plant specific. However, it has to be taken into account that significant damage to any structure in the cavity could occur either due to thermal effects and /or mechanical impact.

4.5.1.4 Corium fragmentation by water in cavity at vessel failure

Fragmented corium will be more readily coolable than dense corium. The fragmentation of the debris can occur by hydrodynamic forces as it flows from the RPV into the cavity or by the occurrence of a molten debris coolant interaction (e.g. steam explosion). Work performed at the Argonne National Laboratory [174], [175] indicates that the ratio of the water pool depth to the debris stream diameter (L/D) is a critical parameter in assessing fragmentation and quenching of the debris stream. For L/D ratios in excess of approximately 50, substantial fragmentation can be expected and for L/D in excess of approximately 75 essentially complete fragmentation could be expected.

Note that fragmentation will generate a wide distribution of particle sizes. With regard to coolability, a debris bed with inhomogeneous particle sizes is unfavourable. The worst situation (given a certain debris mass per area) is a debris bed with large particles at the bottom and small ones at the top. Unfortunately such a situation is likely to develop as large particles are likely to settle first, with the smaller ones gradually being added on top.

4.5.1.5 Coolability of corium

To answer this question is needed information about the spreading area, melt fragmentation, pressure peak and the associated erosion mechanism, issues previously discussed.

A further necessary precondition is the continuous availability of water and the removal of steam.

The assessment of corium coolability has to take into account the situation in the cavity as realistically as possible. This means considering inhomogeneous corium distribution, unmolten debris from the RPV bottom head, remains of RPV or cavity wall insulation or ventilation ducts. If there is only a part of the debris uncoolable, it may be possible that a small molten pool develops and spreads to areas which initially had been coolable. Therefore, coolability in the ex-vessel phase is subject to large uncertainties, even if the average corium configuration seems to be coolable. For this reason, users should avoid any "simplified" conclusion on this phenomenon and a bad treatment into the existing codes.

As an example, the geometry of French PWR reactor cavity bottom consists of a circular cylinder of inner radius 2.6 m, sided by a rectangular area facing the In-core Instrumentation System Room, whose dimensions are approximately 3.0 m X 2.25 m for 1300 MWe reactors, and 4.0 m X 2.6 m for 900 MWe reactors, and by the cavity access corridor. For 900 MWe reactors, the corridor is approximately 90 cm above the bottom of the cavity floor. For 1300 MWe reactors, it is located at the same level than cavity floor, and connected to the rest of the cavity by a narrow path. This location increases the corium spreading area. The distance between reactor vessel and cavity bottom is between 4 and 5 metres. The cavity volume (available for water) is around 150 m^3.

		1300 MWe	900 MWe
Cylinder	Bottom Surface (m^2)	≅21.2	≅21.2
Rectangular Area (Facing the In-Core Instrumentation Room)	Bottom Surface (m^2)	≅6.8	≅10.4
Access Corridor	Bottom Surface (m^2)	≅9.0	-
Total area	Bottom Surface (m^2)	≅ 37	≅31.6

Table 32 Vessel pit surfaces for French PWRs

Referring to the indicative figure of 0.02 m^2/MWth this translates to a necessary area of approximately 80 m^2 for the 1300 MWe reactor and to approximately 55 m^2 for the 900 MWe reactor. Consequently, for both reactor types coolability in the cavity is unlikely.

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4.5.2 Basemat lateral and axial erosion

The following considerations apply to homogeneous concrete cavities. Note that there are reactors with a steel containment bottom or with steel parts or penetrations in an otherwise concrete structure which are not covered by the following considerations. The decomposition of the concrete primarily by thermal loads from the corium is similar to the melting process in many aspects. This leads to downward and sideward erosion of the structures with the possibility of basemat penetration or loss of important static structures influencing the containment integrity.

4.5.2.1 Description of accident phenomena

The corium contacts the concrete cavity at an initially high temperature when it leaves the RPV. The melt consists of a metal phase, mainly steel (Fe, Cr, Ni) with typically 20 tons of non-oxidised zircalloy from the fuel cladding dissolved in the steel, and an oxidic phase of approximately 100 tons. The dominating constituent in the oxide melt is, at this stage of the accident, the UO_2 from the fuel which is mixed up with the ZrO_2 from the in-vessel cladding oxidation. The oxidic melt will gradually absorb the decomposition products of the concrete and become less dense. Depending on the relative densities of the oxidic and metallic melts, the metallic melt may be on the bottom or on the top or, in case of intensive agitation of the melts, dispersed into the oxidic phase. The decay heat as well as the fission products are distributed partly in the metallic, mostly in the oxidic melt and impose a heating rate prescribed by the slowly decreasing decay heat level. The typical height of ideally dense melts at start of concrete erosion is around 50 cm, for large cavities the height may be 20 cm.

A great deal may be understood about the role of molten core/concrete interaction from a very simple picture. The attack of core debris on concrete is largely thermal. In the short period while metallic Zr is still present, a combined thermal and chemical attack of the silicates in the concrete take place. Decay heat and some heat from chemical reactions are generated in the pool and may be lost either through its top surface or to the melting concrete. After early cool down of the melt, the situation rapidly approaches a quasi-steady state where the heat losses balance the internal sources. The partition of internally generated heat between concrete and surface is determined by the ratio of the thermal resistances of the corresponding paths. In this simple view, pool behaviour is dominated by conservation of mass and energy, with heat transfer relations providing the most important constitutive relations.

Under most circumstances, the heat flux to the concrete is sufficient to decompose it, releasing water vapour (adsorbed from hydroxides) and carbon dioxide (from carbonates), and to melt the residual oxides. The surface of the concrete is ablated at a rate which is typically, after transient cool down of the melt, several centimetres per hour. The molten oxides and molten steel from reinforcing bars in the concrete are added to the pool while temperature exceeds steel melting point. The gases are strongly oxidising at pool temperatures and will be reduced, primarily to hydrogen and carbon monoxide, on contact with metals in the pool. Ultimately the reacted and unreacted gases enter the atmosphere above the pool.

During the early interaction phase, metallic Zr as part of the steel melt would chemically interact with the silica of the concrete. Due to its high affinity to oxygen, Zr is able to reduce the melting SiO_2 releasing a substantial amount of chemical energy in a relatively short time period because of the abundance of silicates. Gas released at the bottom or sidewalls of the pool, rises through it as bubbles. The presence of gas bubbles swells the pool level, increasing its depth and its interfacial area with concrete. These rising gas bubbles may also contribute to the production of aerosols.

The liquid decomposition products from the concrete are easily miscible in the oxidic melt. Therefore, the oxidic melt mass is continuously increasing with a decrease of the internal heat source density. The integral decay heat in the melt remains unaffected and follows the slowly decreasing decay heat level. The properties of the oxide layer are more and more dominated by the decomposition products of the concrete, forming in the long term a glassy melt, if the concrete contains a high concentration of silica.

As time progresses, substantial freezing of the metallic phase at the bottom of the cavity will occur. However, this does not exclude continuation of downward concrete erosion, as the “melting point” of concrete (typically around 1300 °C (1573K)) is below the solidification temperature of the steel melt. Depending on the type of concrete and the resulting properties of the oxidic melt, some thin crusts may exist at the oxide pool/concrete interface, or the oxide may remain totally liquid. The concrete erosion could stop after days, if the surface of the melt has increased sufficiently to remove the decay heat in the ground respectively basemat by heat conduction only, without further melting processes. Interaction with water, if any can help stopping the corium progression.

One of the important parameters influencing the interaction of melt and concrete is the composition of the concrete. In most cases, structural concrete used in NPP can be grouped into one of three categories, namely siliceous, limestone/common sand and pure limestone concrete, with their dominant species as listed in next table.

Table 33 Basemat of NPPs : Types of concrete and compositions

Species	Type of concrete		
(components after decomposition)	Siliceous	Limestone/common sand	Limestone
SiO_2	76.6	35.8	3.6
CaO	9.2	33.1	51.6
MgO	5.3	5.2	3.2
Al_2O_3, MgO, Fe_2O_3, K_2O,…	2.9	21.2	35.7
H_2O_{bound}	1.8	2.0	2.0
H_2O_{free}	4.2	2.7	3.9
H_2O_{total}	6.0	4.7	5.9

The main differences in the three types are the ratio of SiO_2 to CaO. The latter originally exits mainly in the form of limestone ($CaCO_3$). At temperatures above 800°C limestone decomposes into CaO and gaseous CO_2. Therefore, concrete with high limestone content generally produces high gas rates, which exceed the gas rates of pure siliceous concrete by a factor of 2 to 3. The gas rates have considerable influence on the dynamics and heat transfer of the melt pool, e.g. by agitating the melt.

Another aspect of the concrete composition is the constitution of the molten slug which forms during the decomposition process, and its interaction with the melt. It has been recognised since the beginning of MCCI studies that the dissolution of the concrete slug in the oxidic corium phase will strongly influence the properties of the resulting oxidic melt, mainly with respect to its freezing temperature and its viscosity behaviour.

4.5.2.2 Main experimental results

A significant number of experimental investigations (BETA, SANDIA Large Scale, ACE/MACE experiments), have increased understanding of the MCCI phenomenology and to assess thermal hydraulic codes like CORCON and WECHSL. The main results may be summarised as follow:

- The gas release during basemat erosion is determined by the type of concrete,
- If metallic zirconium is present in the melt as expected at RPV failure, zirconium may undergo condensed phase chemical interactions by reduction of silica,
- Metallic melts have a very high heat transfer rate to the concrete. Consequently, temperatures of metallic melts rapidly approach their freezing temperature close to 1500°C (1773K),
- The cavity shape during concrete erosion is determined by the ratio of downward to sideward heat transfer. For melts dominated by metallic melt behaviour, the tests show higher downward erosion

which is especially pronounced before onset of interfacial crust formation. Other tests, with UO_2 rich oxide and typical decay heat density, melts also show downward erosion some 30% higher than sideward.

One of the objectives of the OECD-MCCI experimental program (2002-2005) was to address the uncertainties related to long-term 2D core-concrete interaction under both dry and wet cavity conditions. As a part of the project, 3 large-scale, 2D CCI experiments were conducted in specially-designed two-dimensional concrete test sections [181]. These tests employed ~400kg of prototypical, fully oxidised (homogeneous) PWR core melts containing 8wt% of the concrete decomposition products in the melt. The concrete decomposition products are added to simulate a late phase of CCI when some concrete had already been eroded and dissolved in the melt. The input power to the tests (by DEH-Direct Electrical Heating) was selected to be in the range of 150-200kW/m^2 for the initial heat flux to the concrete surfaces.

The most eagerly awaited outcome of the large CCI tests was the determination of the power split ratio, radial-to-axial, i.e. how much heat from the melt pool is going sidewards as compared to heat going downwards. As the concrete ablation in a given direction is directly proportional to heat flux in this direction (when heat conduction in concrete can be neglected and for temperatures above a certain threshold), the power split ratio is determined by the (maximum) erosion depth in lateral direction to the erosion depth axially, downwards.

Reliable estimates of this power split ratio could be very important: it would decide at an accident whether, primarily, the containment structures are threatened by the radial erosion of concrete or whether the axial erosion of the basemat is more pronounced, leading possibly to ground contamination or producing leak paths to underlying compartments in some reactor designs.

As opposed to coolability issues, the 2D ablation behaviour in CCI experiments of the OECD-MCCI project was found to be closely linked to the type of concrete, i.e. its chemical composition (Siliceous versus Limestone (or LCS), as defined above) and its gas content. Two of the 3 CCI tests in this project were conducted with siliceous concrete, CCI-1 and CCI-3, and the third one, CCI-2, with LCS concrete. The CCI-1 was a US-specific siliceous concrete with very low gas content, ~1wt% of equivalent CO_2 from carbonates, CCI-3 was conducted with a French siliceous concrete of about 10wt% of equivalent CO_2. The LCS concrete in CCI-2 had around 30wt% of CO_2.

In both tests with the siliceous concrete the maximum lateral erosion depths significantly exceeded those of the downward erosion. In case of the first experiment, CCI-1, the lateral erosion itself was highly asymmetric, with ablation rapidly proceeding in one lateral direction and much less in the opposite lateral direction. This could cause the usual stochastically-run ablation to proceed too fast into one direction without having time enough (during the experiment) to equalise on the opposite side. All following tests (CCI-2, CCI-3 and also the new CCI tests of the recent MCCI-2 project) were conducted with lower input power. Test CCI-3 then exhibited fairly symmetrical lateral erosion, its maximum depth being distinctly more pronounced than the maximum depth of the downward erosion. In this respect, the results of both the experiments with the siliceous concrete, CCI-1 and CCI-3, are consistent [182]. This has been confirmed by VULCANO tests in a different geometry and with a different heating technique [C. Journeau et al., *2009,* [183]]. In contrast, the lateral-to-axial surface heat flux ratio estimated from the results of the LCS test, CCI-2, is undoubtedly 1:1. This could mean that heavy mixing of the melt by sparging gases from limestone (or LCS) concrete ensures spatially homogeneous heat distribution, whereas their absence in siliceous concrete makes the heat transfer in the pool more like the in-vessel case, driven just by the natural convection. That is, natural convection in a volumetrically heated pool, with almost no additional mixing, where more heat must go sidewards and up than downwards (see, for example, chapter 4 of the thesis [184] with an application to CCI modelling). There are nevertheless arguments indicating that the gas superficial velocity may not be the cause of this effect, such as the fact that heat transfer coefficients to a siliceous sidewall are larger, at the same gas superficial velocity than to a limestone sidewall [T. Sevon, 2008, [185]] and the anisotropic ablation of a specially designed concrete during VULCANO VB-ES-U2 although the gas superficial velocities were almost equal to those of a limestone concrete ablation test [C. Journeau et al., *2010*].

However, a certain controversy still exists about the interpretation of the 2D CCI test results, about those with siliceous concrete in particular [187]. These questions have been again addressed in the project MCCI-2, the follow-up to the OECD-MCCI project.

As already mentioned, an important thing to know is that at an accident a stratified melt pool in the cavity is thought to be more likely to form than a homogeneous pool. This makes the question of radial-to-axial power split more complicated. Experimental evidence for the associated phenomenology is not sufficient for reliable conclusions. Computer models in integral codes address this issue, but cannot be considered validated for this type of problem. The same statement can be made for the question of which melt configurations will develop, as well as for the heat transfer in stratified conditions.

First protopypic experiments with oxides and metals in VULCANO [Journeau et al., Oxide-Metal corium – concrete interaction tests in the VULCANO facility, OECD MCCI Seminar, Cadarache, 15-17 November 2010] have also shown that other melt configurations are possible, launching further experimental R&D.

4.5.2.3 Example for ASTEC application (IRSN)

The basemat of the reactor building is the lower part of the reactor building. It is vital to the containment function and supports structures and components located inside the reactor building.

It is generally made of reinforced concrete or in some cases in pre-stressed concrete. Its thickness under the reactor cavity is approximately 3.5m for French PWR in operation. In some cases, it supports an additional reinforced concrete layer inside the reactor building. The thickness of this “internal basemat” is approximately 1m. Variations may exist from one reactor to the other.

The composition of the concrete varies from one site to another. For theGen II French PWRs, it may include a large part of silica for siliceous concrete, or a mixture of silica and limestone for limestone sand concrete.

For the 1300 MWe PWR L2PSA, the basemat erosion has been assessed through ASTEC sensitivity calculations , using the MEDICIS [189] and CPA modules with the objective to determine the delay before basemat vertical penetration (loss of containment leaktightness), radial penetration (contact with water in the containment bottom) in different cases and to assess the containment atmosphere composition evolution. A minimum thickness below which the integrity of the basemat is no longer ensured has to be considered due to possible concrete cracking below the erosion zone. Values between 0,2 and 0,5 m are used. The calculations are performed for a time window of 15 days, considering that after 15 days the situation is stable.

The following parameters have been used in the sensitivity calculations:

- The composition and the density of the corium involved,
- The configuration of the corium pool in the calculation (homogeneous, stratified or evolving with time),
- The decay power,
- The presence of water in the reactor vessel cavity before the vessel failure,
- The reactor cavity flooding during MCCI phase.

For the uncertainties assessment, it is considered that a calculation with homogeneous configuration provides a maximal value of the delay before basemat penetration and a calculation with a stratified configuration the minimal value. Results obtained show that for some situations (low decay heat, low corium mass, homogeneous configuration …) the basemat is not vertically penetrated but these are only boundary situations. A conclusion is that the basemat vertical penetration after the vessel rupture for Gen II reactors should occur in numerous scenarios; this encourages work on features able to stabilise the corium in most situations, in particular those leading to an early melt-through in absence of additional mitigation procedures.

4.5.2.4 References

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4.5.3 Impact of water injection after onset of MCCI

4.5.3.1 Identification of cooling mechanisms during top water injection

The situation considered here is quenching of an existing corium pool with top water injection started after the initial onset of MCCI. The issue of the long term cooling of a quenched debris-bed above a concrete basemat is not addressed here.

The main cooling mechanisms identified during the MACE program [190] involving 1D ablation and to be retained for discussion in the reactor case are the following ones: the bulk cooling at first contact of the corium pool with the injected water prior the crust build-up, the water ingression into the crust and the melt-eruption through this crust (see 0).However the crust anchorage, that is enhanced in the MACE experiments because of the 1D ablation and of its relatively small scale, has influenced these mechanisms. The bulk cooling process can be ignored because it will play a role only in the transient crust build-up phase. The possible reactor case application would be the scenario of successive phases of crust anchorage, crust break-up, water penetration through the breach followed by a bulk cooling phase leading to a new crust ; however it was demonstrated from crust strength measurements in the SWICCS and CCI programs [192] that in the reactor case the crust strength is not capable of withstanding its own weight.

In the analysis of MCCI phenomena at the reactor scale, it is useful to distinguish between the short term (the very first few hours with a concrete content below 25%), the mid (around 10 hours with a concrete content around 40%) and the long term MCCI phases (around and beyond 20 hours with a concrete content above 60%).

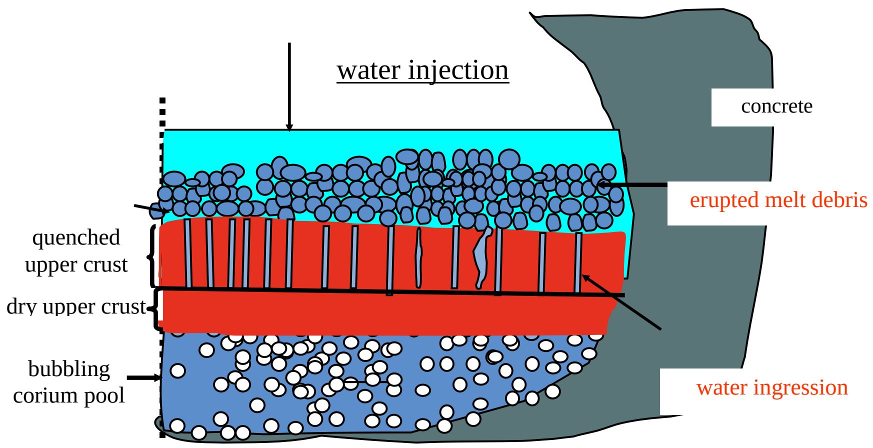

Fig. 47 Representation of the quenching of a corium pool during MCCI by water injection

4.5.3.2 Ingression phenomenon

The dependence of the heat flux extracted by water ingression versus the fraction of concrete in the melt, the pressure and the concrete composition (siliceous or LCS) obtained from SWICCS tests [193] is the following:
The dry-out heat flux is a characteristic measure for assessing the ability of water to cool the underlying corium. When this heat flux is exceeded, the water will no longer be able to access the hot corium so that cooling is no longer achieved. The dry-out heat flux is high (beyond 200 kW/m^2) only for a very low concrete content (8%) ; it decreases below 50 to 100 kW/m^2 for a concrete content around 25% and continues to decrease slowly for a still increasing concrete content corresponding to the mid and long term MCCI phases. The dry-out heat flux depends weakly on the ambient pressure and is slightly lower for a siliceous concrete than LCS concrete (thus < 50 $kW/m^2$for a late phase).
These experimental data show that the dry-out heat flux is smaller than the heat flux out of the corium pool after only a few hours and continues to decrease with concrete ablation. Water ingression is thus not an efficient cooling mechanism in the long term. On the contrary, the CCI-6 test [190] indicates that early water flooding may be efficient. However, this unique experimental result must be confirmed by further experiments.
The SWICCS experimental data [193] are also consistent with the crust permeability measurements performed on SWICCS crust samples: this means that the dry-out heat flux obtained from the classical Darcy's law using these crust permeability data is very similar to the SWICCS experimental dry-out heat flux. This model approach based on Darcy's law is used in the ASTEC code for evaluating the heat flux extracted by water ingression in the upper crust and the recommended value for the crust permeability in the ASTEC code takes into account SWICCS data (a minimum and conservative estimate of the permeability is approximately 3 10^{-11} m^2 at a 1 bar pressure and 10^{-11} m^2 at a 4 bars pressure).

4.5.3.3 Melt-eruption phenomenon

The eruption of melt above the mean corium level and through an existing crust has been observed in several MCCI experiments. This phenomenon is relevant because the erupted material will primarily be quenched by the overlying water, building a pile of debris immersed in water which is coolable as long as the dimensions remain below coolability limits. In addition, it reduces the material available for core concrete interaction.
The analysis of melt eruption phenomena is not mature enough to be applied routinely in L2PSA but general trends on the impact of these phenomena can be outlined as described below. Mostly the reactor cavity can be so small that even if temporary eruption occurs the overall corium load, including uncertainties about the corium distribution would be too high to demonstrate coolability at least in the general case. A more relevant application may be possible if the melt spreads further, e.g. into the sump in case of large enough lateral ablation. Then it is worthwhile discussing the coolability potential due to the melt eruption phenomena.
The average entrained melt volume rate can be deduced from the measured entrained mass above the corium pool during real material tests assuming that the volumetric entrained melt rate is proportional to the injected

gas volume rate as proposed by Epstein [196]. The value of the ratio of the volumetric entrained melt rate to the injected gas volume rate (entrainment ratio) $K_e = \dfrac{Q_{cor,entr}}{Q_g}$ deduced both from MACE tests with 1D ablation [194] and CCI tests with 2D ablation [195] is about 0.1%. The value of the entrainment rate can be correctly evaluated at least in order of magnitude using the Ricou-Spalding correlation [197]:

$$K_e = E_0 \cdot \left(\frac{\rho_g}{\rho_m} \right)^{1/2}$$

as used in the CORQUENCH code [198] and the ASTEC code [199] with a coefficient value of E_0 equal to 0.08 as recommended by Epstein [196] and also in ASTEC. In particular it was shown that the measured entrained mass above the corium pool observed in the CCI2 test is correctly reproduced with an overestimation by a factor of 2 by ASTEC using this simple model [199]. The melt eruption mechanism is much more efficient for cooling than the water ingression mechanism in particular in the mid and long term phases provided that the gas content of the concrete is sufficient, which is true for the LCS concrete but not for the siliceous concrete.

A detailed modelling of melt eruption but with a consistent evaluation of hole size and hole density derived from constraints on hole plugging by freezing and gas flow through holes was proposed by Farmer [201] and introduced as an option in the CORQUENCH code [202]. This modelling effort with an attempt at a mechanistic evaluation of melt eruption is continued at Wisconsin University [203] and will lead to further improvements of the CORQUENCH code; this type of model is interesting because it combines a complete set of mechanistic physical models for the melt eruption. However due the complexity of the modelled phenomenology the model will need additional validation against experimental data before being applied with some confidence to the reactor case.

Therefore, in the present state of knowledge, the use of the melt eruption model based on the Ricou-Spalding correlation and an E_0 coefficient value near that proposed above can be considered as both validated against experiments and also reasonably conservative, i.e. leading to slightly pessimistic results on the impact of water injection on the ablation kinetics during MCCI. Limitations of the model, labelled 'non conservative" if decreasing the conservativeness or "conservative" if increasing it, should nevertheless be kept in mind:

- This model was validated against experiments where crust anchorage occurs at some time period and might limit the melt eruption phenomenon leading to a possible underestimation of the entrainment rate by the model (conservative),
- The assumption of an entrainment ratio independent of the volume gas rate is reasonable but open to discussion: this ratio might decrease with increasing volume gas rate as shown by the PERCOLA experiments and their interpretation leading to a possible overestimation of the entrainment rate by the model at high gas volume rate (non conservative),
- The model has no limitation of the melt eruption rate due to the accumulation of ejected debris above the pool upper crust (non conservative).

4.5.3.4 Application of MCCI codes to reactor cases with water injection

Capabilities for most currently used MCCI codes can be summarised as follows [198], [204] :

CORQUENCH [202]:

The CORQUENCH code permits only a simplified modelling for the MCCI in dry conditions compared to other codes, since only the case of an homogeneous pool configuration is described; however several optional coolability models are implemented in the code for water ingression, melt eruption and crust anchorage and break-up with some new features (crust cracking model for the description of water ingression, evaluation of hole size and hole density for the detailed treatment of melt eruption). This code focussed on coolability aspects permits to perform easily parametric studies on the coolability models but with a fixed homogeneous pool configuration.

TOLBIAC [206] :

The TOLBIAC code has the same modelling level for the MCCI in dry conditions as ASTEC and CORCON codes, in particular with the treatment of different pool configurations and of the switch between these configurations ; moreover it contains coolability models including a detailed model of melt eruption derived from the analysis of the PERCOLA experiments [200] ; as already mentioned the lack of this approach for the melt eruption description is the need of additional models or data for determining the hole size and hole density in the melt eruption model.

MELCOR/CORCON [205]:

The CORCON code has roughly the same modelling level for the MCCI in dry conditions as ASTEC and TOLBIAC codes, in particular with the treatment of different pool configurations and of the switch between these configurations; however no coolability model is implemented in the code.

ASTEC/MEDICIS [207]

The ASTEC/MEDICIS code has the same modelling level for the MCCI in dry conditions as TOLBIAC and CORCON codes, in particular with the treatment of different pool configurations and of the switch between these configurations. Besides it includes simplified coolability models similar to those present in the older version of CORQUENCH code [198] with only one parameter for each mechanism (permeability in the Darcy's law for the water ingression model and coefficient in the Ricou-Spalding entrainment correlation for the melt-eruption model). Values recommended in ASTEC for theses 2 parameters are consistent with experimental data as explained above. This code also permits a large flexibility in the choice of heat transfer, pool/crust interface models and pool configurations and switching criteria. Therefore the ASTEC/MEDICIS code permits easily to perform parametric reactor calculations on MCCI with a realistic configuration evolution and taking into account the coolability aspects.

Some results of MCCI calculation on the concrete ablation stabilisation, obtained from some recent studies are briefly mentioned as examples:

- The parametrical study performed by M. Farmer [208] with the CORQUENCH code in case of an homogeneous pool configuration: parameters of the study are the corium inventory, water injection onset time and concrete type. This study leads to the following results: the risk of axial melt-through increases fast with increasing corium inventory, time delay of water injection onset and is much higher with siliceous concrete than with LCS concrete,

- The parametric study performed with the ASTEC code [209] (using recommended coolability model parameters mentioned above): parameters of the study are the pool configuration assumptions and concrete type on the impact of water injection ; results show the large impact of early water injection for all realistic pool configuration evolutions in case of LCS concrete, e.g. the large increase of the melt-through time by at least ten days for a thick concrete LCS basemat thickness (4m) and the limited impact of the water injection by at most 1 to 2 days for a thick concrete basemat thickness (4m) in case of siliceous concrete,
- The parametric study performed by K. Robb [210] with the CORQUENCH code in case of an homogeneous pool configuration investigates a much larger range of parameters ; it confirms results obtained by the previous study with the same code [208] mentioned above and completes it as far as the melt eruption is concerned: this study shows that the melt eruption is the prevailing cooling mechanism and that an ablation stabilisation is likely only for an average melt entrainment rate around or above 0.1%.

This short review of parametric reactor applications on the coolability aspects shows a consistency between studies at least on main trends in spite of different codes used and boundary and initial conditions. The prevailing cooling mechanism during the water injection is the melt eruption mechanism and the higher efficiency of the water injection in case of LCS concrete and to a less extent in case of an early onset time. Studies differ mainly by conclusions on the threshold entrainment rate required to get an ablation stabilisation due to melt eruption (roughly between 0.1% and 0.5%).

4.5.3.5 Conclusions and applications to PSA studies

Experimental data on the corium coolability during MCCI deduced from both analytical tests (PERCOLA and SWICCS) and integral tests (MACE and CCI) permitted to identify the major cooling mechanisms: the water ingression mechanism, which leads to a large extracted heat flux (beyond 100 kW/m^2) only in the early MCCI phase (first hours); the melt eruption, which is efficient during the whole MCCI phase and might lead to a more or less extended quenching of the corium pool. However the ablation stabilisation with a quenching of most of the corium pool was not demonstrated in available real material experiments, except for some favourable early flooding cases.

More precisely, the applications to the reactor case confirm that the melt eruption is the prevailing cooling mechanism. The melt eruption will delay the axial basemat melt-through significantly if the entrainment rate has the same order of magnitude as observed in real material tests (0.1%) and with a much larger efficiency in case of LCS concrete. The efficiencies of the water ingression and also, to a lesser extent, the melt eruption are enhanced if the water injection occurs in the early MCCI phase.

The melt eruption has good chances to prevent the axial basemat melt-through only if the entrainment rate is several times larger than the experimental value. Large uncertainties remain however on the validity of present melt eruption models in the long term MCCI phase involving the build-up of a thick debris bed above the corium pool. This conclusion is similar to that made by B.R. Sehgal in a review paper [211], but a little more pessimistic.

In practice conclusions for the PSA studies are the following:

- Approximate but conservative estimates of the impact of water injection on the ablation kinetics during MCCI can be obtained with some of the existing codes (CORQUENCH, ASTEC, TOLBIAC) provided available coolability models are used with parameters fitted on the real material experiments (best-estimate models),
- The impact of water injection on the ablation kinetics evaluated from these "best-estimate models" is significant and increased if the water injection occurs early, but anyway much more pronounced in case of LCS concrete ; the uncertainty of coolability predictions increases in the longer term MCCI phase because of the complexity of the corium pool/debris bed configuration,
- These "best-estimate models" do not predict the ablation stabilisation in the reactor case at least in the case of large initial corium inventories, except if increasing the entrainment rate level well beyond the values observed in real material experiments,

  Uncertainties are remaining on the melt eruption process in particular on the question if the process will go on even for a thick accumulated debris bed with "active" volcanoes scattered across this debris bed.

To reduce these uncertainties, it will be necessary:

- First to analyse the experimental database from existing analytical tests, e.g. on the formation and quenching of debris bed such as the DEFOR program [212] (although the scenario addressed by this program is more appropriate for the issue of BWR coolability issue), and to analyse the scarce data from real material experiments (including the last CCI6 test of the MCCI2 program),
- Later to build models validated against the experimental database and capable to take into account size scale effects for the debris bed formation and cooling during MCCI.

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4.5.4 Production of steam and incondensable gases

4.5.4.1 Description of accident phenomena

As a consequence of the RPV failure, a mixture of molten materials consisting of the fuel, cladding and structural elements ("corium") will be released from the vessel into the reactor cavity.

Two issues should be distinguished: steam production from water in a wet cavity, and steam and gas production from the decomposition of concrete in a CCI process.

Water can be available in the cavity due to the accident progression. The water presence in the cavity at the RPV failure is design dependent and may be the result of some accident management strategy or the result of spray system operation or of overflow of the sump water. Cavity overpressurisation can occur the produced steam from the corium water interaction could not leave fast enough the cavity. The cavity fails if the structure could not withstand the dynamic loading (see chapter 4.4.3). In other cases the cavity can survive the fast pressurisation and the debris is quenched (see chapter 4.5). Then slow overpressurisation from the evaporating water occurs.

Concrete erosion generates gas release into the containment atmosphere. Whereas the production of steam and carbon dioxide contributes to the pressure increase in the containment, the release of hydrogen and carbon monoxide may eventually lead in addition to the formation of combustible gas mixtures in the containment atmosphere. Both these types of hazards resulting from MCCI are usually taken into account by dedicated model approaches in PSA studies.

The gas production is a strongly design dependent issue, according to the type of concrete composition. The following gases can be produced from the concrete: H_2O, CO_2, CO, H_2.

The ablation velocity of concrete is governed by the heat flux between the corium pool and the concrete wall, and the gas release velocity strongly depends on the ablation rate. Concrete types are defined by the type of the aggregates used in the concrete. The chemical composition of the concrete determines the gas release during molten corium concrete interaction. The composition of the two main types, siliceous and limestone common sand concrete can be seen in Table 34.

Table 34 General concrete composition[Firnhaber 1992] for siliceous concrete and [Farmer 1997] for LCS

Component/wt%	Siliceous Concrete	LCS concrete	Heat of formation [kJ/kg]
$Ca(OH)_2$	11.92	12.79	-13322
$Al_2O_3.3H_2O$	2.40	2.4	-16593
$CaCO_3$	6.48	30.83	-12066
$MgCO_3$	0.68	16.48	-13000
Fe_2O_3	1.7	1.7	-5146
CaO	0	0	-11339
SiO_2	70.0	29.28	-15180
MgO	0.0	2.08	-14917
Al_2O_3	3.2	2.03	-16424
Water (free)	3.62	2.41	-15880
Specific enthalpy of hardened concrete (kJ/kg)	-14656.0	-13511.0	

Basaltic concrete, which is used in some nuclear power plants, is similar to siliceous concrete but with more impurities like MgO, Al_2O_3, Fe_2O_3, K_2O and CaO.

Water in the concrete pores and capillary pores is called evaporable or physically bound water, in contrast to the chemically bound water in the hydrates and hydroxides.

Prime contributors to concrete decomposition are endothermic degradation reactions, which lead to a successive destabilisation of the concrete structure. These reactions play a dominant role for the decomposition of hardened cement and involve the evaporation of free water at ~100°C, the degradation of the Portlandite ($Ca(OH)_2$) to CaO and H_2O between 450°C (723K)and 550°C (823K), and, to a minor extent at a temperature level of ~800°C, the decarbonisation of the limestone fraction in the cement. The decarbonisation of limestone also occurs in the aggregates. If the aggregates consist to a large part of limestone, this reaction has a considerable influence on the release of CO_2. In this context, it is worth noting that limestone concrete releases ~40wt% of its weight as CO_2 [Roche 1994]. In contrast, siliceous concrete typically liberates 5wt% as CO_2. This relatively small fraction is attributed to lime impurities in the siliceous aggregate.

The concrete is decomposed at around 1300°C (1573K). Chemical reactions are of significance for the released mass composition and temperature.

The first task at hand is to examine the stability of the relevant MCCI oxides in the presence of reducing metals and, given this precondition, to formulate condensed phase reactions. The second task is to evaluate the reactions between the gaseous concrete decomposition products and metals.

The most important chemical reaction in the pool is the oxidation of metals by the concrete decomposition. Due to the oxidation the following simplified chemical reactions modify the gas source from MCCI.

In general, metals undergo an oxidation by H_2O or CO_2 of the form [Frohberg 1994]:

$$MT + n_G H_2O \leftrightarrow n_{Ox} MT_x O_y + n_G H_2$$

$$MT + n_G CO_2 \leftrightarrow n_{Ox} MT_x O_y + n_G CO$$

MT = metal in the corium from core (Zr) and from structural materials (Cr, Fe, Ni)

Therefore the released gases are: CO, CO2, H2, H2O (Fig. 48)

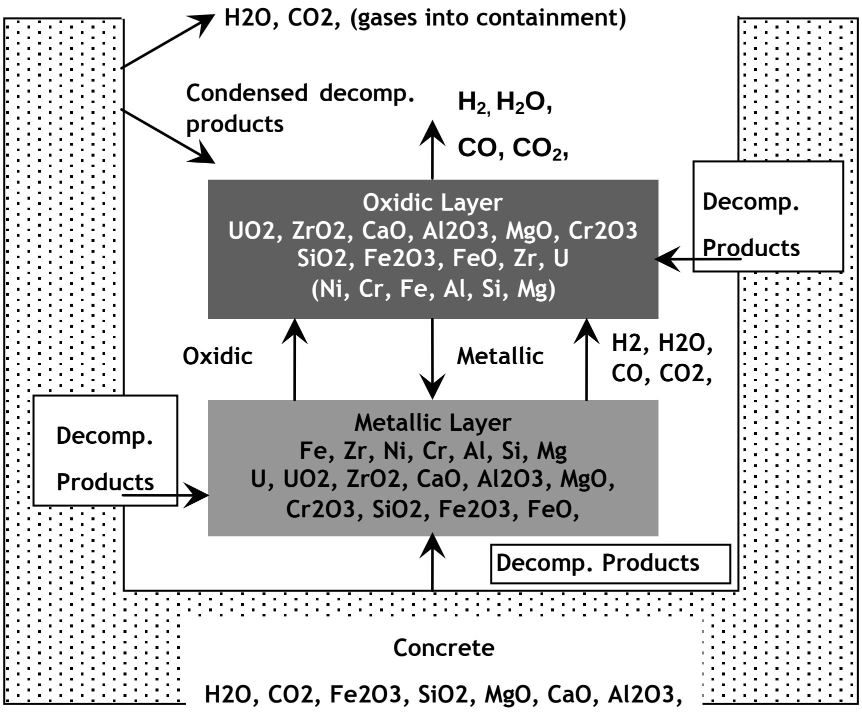

Fig. 48 A melt pool and mass transfer

The release rate depends on ablation velocity of the concrete (see in detailed in chapter 4.4.2). The current practice in L2PSA is to use codes such as MELCOR-CORCON, MAAP-DECOMP, ASTEC-MEDICIS or WECHSL to calculate the released gases from MCCI. The main quantities to be calculated by the MCCI model are the radial/axial erosion depths and the gas release rates as function of time. The latter ones (gas release rates) are derived from the former ones (erosion depths) in combination with the concrete composition. It must be stressed that the total gas release rate is almost independent of the power split between lateral and axial ablation.

4.5.5 Evolution of containment atmosphere composition and long-term pressurisation

4.5.5.1 Description of accident phenomena

The key issues for the containment atmosphere evolution are its flammability and the containment long term pressurisation. Processes which are particularly relevant in the long-term are:

- Continuous generation of decay heat,
- Heat transfer from the containment atmosphere to structures and the associated steam condensation,
- Steam production from residual water if corium-water contact occurs,

- Containment leak rate (design leak rate or enhanced leak rate).

Depending on the plant design and on potential accident management procedures, the following issues may be relevant:

- Gas production from MCCI,
- Availability of water in the containment,
- Operation of sprays or fans,
- Deliberate depressurisation (venting) of the containment,
- Effects of hydrogen recombiners,
- Effects of dedicated heat removal systems,
- Chemical species being produced by long-term corrosion of containment internals.

It is obviously not possible to track the combination of all these phenomena by simple estimates. Therefore, the containment atmosphere evolution should be analysed with integral codes or with dedicated containment codes. A particular issue is the relation between atmospheric composition and fission product retention e.g. behaviour of hygroscopic aerosols, Iodine chemistry. Many of the issues related to this have been discussed in section §7.

The results of such analyses should be transferred into the probabilistic framework (APET) of the L2PSA. One method for integrating the analysis results into the APET is to introduce a simplified physical model of the atmosphere into the event tree analysis which should be validated against the underlying results. An example of such an approach by IRSN is given in section §4.5.5.3 below. GRS has also selected this kind of methodology [213] however, some computer codes for APET evaluation do not allow such linking of user-defined models, so that branching probabilities for the APET often have to be determined outside of the APET and then be transferred into the APET in a suitable format.

Whatever the methodology, the following trends have been observed in various PSAs:

- Depending on the availability of flow paths inside the containment, the containment atmosphere is assumed to be more or less homogeneous. The analysis of inhomogeneous containment atmospheres requires a fine nodalisation of the computer model and hence a significant effort. Inhomogeneous conditions may be prone to local unfavourable atmospheric conditions, therefore it is recommended that sophisticated analyses should only be undertaken if simpler approaches cannot exclude critical situations,
- If a continuous MCCI occurs, the production of CO and H_2 is practically unlimited (valid for large dry PWR containments),
- PARs are primarily designed to cope with the hydrogen generation during the in-vessel phase. The generation rate of combustible gases due to MCCI in the ex-vessel phase is probably lower on average and the source is located at a very low position inside the containment. Both arguments support the expectation that the recombiners will be able to cope with the generation rate in the ex-vessel phase (additional information related to the necessary availability of oxygen is given below). This expectation should be justified,

- In the case of continuous production of CO and H_2, hydrogen recombiners will cease to function when the oxygen inside the containment has been used up. The consequences are continuously rising H_2 and CO volume fractions with no oxygen present. The possibility of late O_2 ingression in the containment, after containment failure, and its consequences should be examined in the L2PSA if this is relevant,
- If the containment leak rate is considerable, long term pressurisation may not be an issue, but in this case the assessment of the leak rate in terms of releases is important (see § 5).
- Water injection onto the corium – either deliberately by SAM procedures or due to accidental phenomena like ingress of sump water to the corium - may accelerate the containment pressurisation rate,
- If the spent fuel pool is located inside the containment, and if the cooling systems or the water level of this pool is affected by the accident sequence (e.g. in station blackout sequences), there may be additional contributions from this pool (steam, heat) to the containment atmosphere evolution,
- The containment venting system efficiency has to take into account the atmospheric composition, possibly including considerable hydrogen and CO fractions, and pressurisation rates; depending on the plant design, a L2PSA may include specific sequences where the containment venting system is not sufficient to depressurise the containment,
- The emission of fission products associated to MCCI can be calculated but lead to weak mass (~several tens of kg) in comparison with total emission of fission products from the core (several tons), but it may have a large impact of external radiological consequences if this emission follows a containment failure (at vessel rupture for example),
- The emission of concrete aerosol during MCCI phase has an impact on the kinetics of deposition of other radioactive particulates in containment (accelerate deposition).

4.5.5.2 Hydrogen and CO combustion

The evolution of the containment atmosphere in the in-vessel phase has been covered in section §4.3.6. The key issues of convection and flammability also apply in the ex-vessel phase. However, if there is a concrete containment bottom and if MCCI occurs in the ex-vessel phase, several additional aspects have to be considered.

After the beginning of the ex-vessel phase with MCCI, hydrogen, CO and CO_2 are released from the melt pool, coming from the concrete composition. The rate of CO and CO_2 release depends strongly on the type of concrete and the water bound in the concrete.

CO is a burnable gas with a relatively high density but with much less reaction energy per kg, in case of combustion compared to hydrogen. It is distributed with the convection flow and in case of combustion it is burned simultaneously with hydrogen. It is recombined by the recombiners as well. Thus, it contributes to the oxygen consumption and influences the duration of oxygen availability and flammability. A general difficulty in the treatment of combustion of CO or a hydrogen/CO mixture is that not all methods developed in the context of hydrogen combustion are readily available e.g. criteria for FA or DDT. Within a L2PSA it is necessary to consider the effects of CO and CO_2 on combustions. A possible approach could be as follows:

- The atmospheric composition after vessel rupture should be assessed, including the impact of potential energetic phenomena such as DCH and steam explosion, which might impact the number of moles of H_2, O_2 and steam,
- Evolution of the containment atmosphere, including CO and CO_2 from MCCI, should be analysed by an integral code or by specific codes for tracking the containment atmosphere (see example below),
- Flammability of the containment atmosphere should be determined by applying integral codes. The presence of CO_2 and CO can modify the atmosphere flammability limits and, if relevant, some specific flammability limits should be used,
- For assessment of combustion, flame acceleration and detonation in atmospheres containing considerable amounts of CO, there is presently no method suitable for L2PSA application. A simple judgement is therefore recommended, replacing the CO volume fraction by H_2 and applying the hydrogen related knowledge.

4.5.5.3 Example of pressurisation and H_2 and CO combustion modelling in APET during ex-vessel phase (IRSN)

In IRSN L2PSA, the containment atmosphere flammability during the ex-vessel phase is assessed via a fast running code, OSCAR, which is operated during the APET quantification with KANT.

OSCAR performs a calculation of the containment atmosphere flammability at each instant of the ex-vessel phase and calculates composition, pressure and temperature at each time step. The calculation of the atmospheric composition depends on:

- The initial conditions at vessel rupture time. As mentioned before (see § 4.5.1), they depend on the previous events that have occurred during the accident sequence including: reflooding, H_2 combustion during the in-vessel phase, DCH, etc. The relevant variables initiating the containment state during ex-vessel phase are: N_2, H_2, O_2 and steam mass, pressure and temperature at vessel rupture,
- The MCCI induced gas production, which are tabulated (from ASTEC sensitivity calculations) as a function of the concrete type, the total corium mass interacting with the concrete, the quantities of non oxidised steel and Zircaloy in the corium;
- The recombiner action, which has been modelled taking into account the results of the H2PAR and KALI H2 experiments; the recombination rate depends on the pressure, the amount of oxygen, the amount of hydrogen and hardware characteristics,
- The containment spray system status i.e. running or not, and, if running, its time of activation. The modelling of the spray system is quite simple: in the IRSN 900 MW L2PSA APET for release 4, the spray system is supposed to completely and instantly dry out the containment atmosphere.

The containment is modelled as a single volume in which, at each instant, a mass balance of non-condensable gases is performed. Use of a single volume may overestimate recombiner efficiency unless very efficient convection takes place. If the convection is not sufficient, there will be different atmospheric conditions in different parts of the containment. Under such circumstances a hydrogen rich layer may develop and only those recombiners which are located there will be able to act. The assumption of a single volume has been qualitatively justified by the open internal geometry of the 900 MWe PWR containment and confirmed with more detailed ASTEC calculations.

To begin hydrogen/carbon monoxide combustion flammable gases, oxygen, and an energy source are needed. The combustion may be initiated in various ways e.g. short-circuit, hot particles, sparks, high temperature on recombiner catalyst, occurring in a highly stochastic manner except for recombiners where experimental evidence has been collected. Therefore, at each time step during the ex-vessel phase, the gas flammability is checked. Regarding the issue of flame acceleration, it has been established that the gas ignition criteria, by the recombiners, is below the flame acceleration criteria. Thus the assumption has been made that flame acceleration is not possible [214] and that the evaluation, at each time of the ex-vessel phase, of the adiabatic isochoric pressure combustion may be sufficient. The fraction of gas to be burnt during combustion is assumed, in a pessimistic manner, to be equal to 1.

The maximum combustion pressure, over the entire ex-vessel phase, is then used as an input to a containment fragility model (see section § 5.2), which gives the containment failure probability.

This tool returns its results within seconds.

A containment pressurisation model (number of moles of steam) is also implemented in the APET (with KANT) based on linear correlations; these correlations have been established using the ASTEC/MEDICIS code.

Pressurisation (number of moles of steam) slopes have been assessed considering:

- The initial amount of water in the reactor cavity bottom, according to the spray system and the water injection system running or not,
- The appearance of a lateral breach in the reactor cavity bottom, or in the access corridor, due to the MCCI phenomenon, which may cause reflooding of the corium by the water lying in the bottom of the reactor building,
- The dry out of the reactor cavity bottom after vaporisation,
- The total mass of corium in the reactor cavity bottom,
- The type of concrete.

The containment pressurisation model is coupled with OSCAR described above.

Some future improvements of the modelling in the APET may come from these points:

- Verification with more detailed ASTEC (or CFD) calculation that modelling with a single containment volume is sufficient , i.e. justification that hydrogen stratification in the upper part of the containment cannot modify the assessment of recombiner efficiency,
- Better modelling of thermal effects and phenomenon regarding steam condensation or vaporisation,
- Regarding recombination, hydrogen and carbon monoxide are supposed to behave exactly in the same way. The ongoing experimental work (IRSN and CNRS) may help to generate more precise information about this matter,
- Confirmation that gas combustion by the recombiners prevents any possibility of flame acceleration.

4.5.5.4 References

[213] Assessment of the accidental risk of advanced pressurized water reactors in Germany, GRS-184, April 2002.

[214] A. Bentaib, C. Caroli, B. Chaumont, K. Chevalier-Jabet, “Evaluation of the impact that PARs have on the hydrogen risk in the reactor containment : Methodology and application to PSA2”, Science and Technology of Nuclear Installation, Volume 2010

4.5.6 Containment venting in the ex-vessel-phase

4.5.6.1 Introduction

Many of the generic issues related to containment venting have already been covered in section §[125] related to venting during the in-vessel phase. A significant difference between containment conditions for in-vessel and ex-vessel phase containment venting exists if MCCI occurs. MCCI will create significant amounts of condensable and noncondensable gases, thus increasing the containment pressure. Therefore, this present section considers only those aspects of containment venting which are related to MCCI.
There are reactor types which do not have concrete in the containment structure below the RPV, e.g. the German BWR-69 type. The following considerations do not apply to such types of reactors.

4.5.6.2 Issues to be addressed in PSA

The same issues have to be addressed, with regard to containment venting in the ex-vessel phase, as considered in section § [125] on in-vessel venting.

4.5.6.3 Initiation and control of the venting process

The pressure increase in the containment due to MCCI after RPV failure is slow. Immediately after RPV failure into a dry reactor cavity there may even be a temporary pressure decrease in the containment. Of course, the containment pressure history depends on plant properties and on the accident sequence. As an example, in a PWR with large dry containment of German design there is at least one full day available between RPV failure and the need to vent the containment. This slow pressurisation rate ensures plenty of time for preparing and initiating the venting process. Therefore, human error due to inadequate time to perform actions will probably not be significant. Nevertheless, some accident conditions may cause additional difficulties for the operators e.g. failure of the pressure measurement system in the containment or degraded radiological conditions in the vicinity of the valves to be operated (if manual).

4.5.6.4 Pressure reducing capacity of the venting system

The thermodynamic analyses for the containment atmosphere in the ex-vessel phase are well advanced. The remaining most uncertain issue with regard to the venting system capacity is the gas and steam generation rate from the MCCI and the condensation inside the containment. Considerable uncertainty should be allowed for these parameters when assessing the venting system. Furthermore, for some reactors, heat and steam from the spent fuel storage pool (in extreme cases also gases from spent fuel degradation) could contribute to the containment load if an extended loss of cooling exists. If the spent fuel pool cooling is functional, the cool surface of this water pool will act as a heat sink and could decrease the containment pressurisation.

4.5.6.5 Failure of the venting system

Containment venting systems are comparatively simple systems. Therefore, the failure probability of the system can be assessed with standard techniques as discussed in section 4.3.7.5. Containment venting in the ex-vessel phase occurs late in the accident, so that battery-supported power will probably no longer be available.

4.5.6.6 Filtering efficiency of the venting system

This issue has been addressed in section 4.3.7.6.

4.5.7 Pool scrubbing

4.5.7.1 Introduction

This phenomenon involves the retention of radionuclides in any water volume existing in the release path of the contaminated gas flow. Therefore, pool scrubbing could play an important role in the mitigation of severe accidents. The empirical data shows that pool scrubbing of chemical forms involving some important elements such as iodine, caesium and tellurium is an efficient mechanism, thus reducing the radionuclide releases to the environment. Fission product scrubbing by water pools during the in-vessel phase includes aerosol and vapour retention in the quench tank or pressurizer in PWR plants and suppression pools in BWRs. The pool scrubbing phenomenon is also applicable in the ex-vessel phase in cases where the molten materials are covered by an overlying water pool. It is expected that a substantial fraction of the aerosol would be retained by the pool and therefore not be present in the atmosphere.

The scrubbing efficiency of pools, measured through the decontamination factor (DF)[18], depends on many parameters, including the type of particulate species to be trapped, the fraction of non-condensables in the bubbled through gas, pool depth and pool temperature.

Other example of PWR scenarios with pool scrubbing are:

- Loss of off-site power with loss of RCS heat removal: scrubbing of fission products through the water retained in the pressurizer,

[18] Ratio of activity prior to and after the decontamination of radioactively contaminated objects, waste water, air, etc.

- SGTR sequence: retention of fission products on the secondary side, if the tube rupture remains covered with water,

Other example of BWR scenarios with pool scrubbing are:

- Large pipe break with failure of the standby core cooling system: radionuclides released from the RCS to the drywell are scrubbed as they pass through the suppression pool,
- Anticipated transient with failure of reactor shutdown system, or failure of residual heat removal or failure of all makeup water (the TQUV sequence): in all these accident scenarios, retention of radionuclides could occur when radionuclides are released through the safety valves into the suppression pool.

If the water pool is expected to evaporate, or significantly reduce later in the accident sequence, the possibility of re-mobilisation of radionuclides trapped in the pool should be considered. In this case it may be sufficient to just take into account a time delay (consistent with the vaporisation) in the releases. Specific calculations with codes like ASTEC may be useful.

4.5.7.2 Description of accident phenomena

The combined effect of all the possible phenomena leading to decontamination in the pool is a complicated function of several parameters, including the values of thermal-hydraulic variables in the pool such as: temperature, the non-condensable fraction in the steam-gas mixture, fluid dynamic parameters and particle/bubble sizes. Vapours and aerosols within gas bubbles can diffuse, undergo sedimentation or inertially impact the gas-water interface. Surface tension and *Van der Waals* forces ensure that when a vapour or aerosol particle reaches the gas-water interface, it will be trapped in the aqueous phase.

Inertial impaction of aerosol particles with the bubble walls occurs because gases within a rising bubble circulate. The nature of the gas circulation depends on the bubble size and the purity of the system. When gas circulation does occur, excessively large aerosol particles cannot follow the stream lines of the gas flow. Due to inertia, these particles will cross the streamlines and impact the bubble walls. Particles larger than about 0.5 µm are mostly affected by inertial impaction.

Diffusion of aerosol particles within the bubbles is the result of Brownian motion brought on by the stochastic nature of gas molecule collisions with the particles. Diffusion significantly affects only very small particles, less than about 0.1 µm.

Sedimentation is the gravitational settling of particles within bubbles. Typically, sedimentation is important only for particles larger than 1 µm.

The factors with a strong influence on the overall decontamination factor are:

- *Aerosol particle* size. Large particles are retained in water pools more easily than small particles. Stable, dispersible aerosols coalesce rapidly when their concentration is high; on the other hand, low density aerosols increase their effective density rapidly in the presence of water vapour, serving as condensation nuclei. The other effect resulting in aerosol particle enlargement is agglomeration. All these processes lead to an increase in aerosol particle size and their easier capture by the water pool,

- Steam fraction in the carrier gas. The higher the steam fraction, the higher the DF. This dependence results clearly from the above, as steam condenses onto aerosol particles making them larger their retention increases,
- Submergence of the injection point. In general, experiments show that the DF increases with the depth of water above the injection point, both for aerosols and vapours. For very massive particles this effect is less important, since the majority of such particles are immediately trapped close to the injection point.

Other secondary factors with a potential impact on the DF are:

- Solubility of aerosol material: higher DF values are seen for soluble species,
- Pool sub-cooling: lower DF values are seen with lower sub-cooling,
- Dependence on the injector type,
- Carrier gas temperature.

4.5.7.3 Application to L2PSA

The main uncertainties related to pool scrubbing are connected with the capability to adequately describe the accident scenarios, as well as the inherent uncertainty in predicting fission product release and transport/retention through the release path up to the water pool itself. These uncertainties are mainly related to: magnitude of the gas flow entering the pool and bubble swarm behaviour, chemical properties of aerosol particles and their size distribution and shape. In particular, the rising velocity of the bubbles, which strongly influences the DF, is sensitive to the gas mass flow, the pool geometry and water convective motions.

4.5.7.4 Overview of decontamination factors

In WASH-1400 [216], the recommended conservative DFs were:

- Primary system pool scrubbing : DF=10 for all fission products, except noble gases, and only in BWR,
- Containment system pool scrubbing : DF=100 for I_2, DF=50-100 for submicron solid particles and DF=2-1 for CH_3I and noble gases.

In NUREG-1150 [218] the assumed values for the decontamination factor were:

- At Grand Gulf NPP: for in-vessel release a range from 1.1 to 4000, with a median of 60, and for the ex-vessel release, a range from 1 to 90, with a median of 7,
- At Peach Bottom NPP: for in-vessel release, a range from 1.2 to 4000, with a median of 80, and for the ex-vessel release, a range from 1 to 90, with a median of 10.

Powers and Sprung [217] performed an analysis of the uncertainty in aerosol decontamination using the Monte Carlo method. These calculations have shown that the DF increases sharply, in comparison with values obtained for saturation conditions, as water sub-cooling increases from 2 K to about 10 K. In the case of sub-cooling greater than 10 K, the effect on the DF is less evident, see Table 35.

Table 35 Aerosol decontamination by pool scrubbing using the Monte Carlo Method (Powers and Sprung [217])

Pool depth (cm)	Sub-cooling (K)	50% quantile of DF at 50% confidence level
30	0 5 10	2.2 - 2.3 21.7 - 22.6 44 - 49
50	0 5 10	2.8 - 2.9 34 - 39 80 - 90
100	0 5 10	4.9 - 5.1 89 - 96 209 - 257
300	0 5 10	19 - 23 1830 - 2448 8358 - 10263

4.5.7.5 Experimental and modelling research programmes

Study of the aerosol and pool scrubbing phenomena has been an object of research in many centres throughout the world:

- Advanced Containment Experiments (ACE), Phase A,
- EPRI/BCL experimental programme,
- GE experiments,
- Suppression pool aerosol retention test apparatus (SPARTA) experiments,
- JAERI programme using the EPSI facility,
- Light water reactor advanced containment experiments (LACE) in the CIEMAT facility (Spain),
- Pool scrubbing effect on iodine decontamination (POSEIDON) programme of PSI,
- UKAEA experiments for the steam generating heavy water reactor (SGHWR).

In addition to the experiments, computational methods have been developed which model the physical phenomena essential for aerosol and vapour retention in water pools:

- SPARC (Suppression Pool Aerosol Removal Code) developed by the Pacific Northwest Laboratory,
- BUSCA (Bubble Scrubbing Algorithm) of UK origin,
- SUPRA (Suppression Pool Retention Analysis) property of EPRI,
- ECART (Code for analysis of radionuclide transport) property of CESI Ricerca.

The state-of-the-art report EUR 16241 [215] summarises the experimental programmes and codes.

4.5.7.6 References

[215] Escudero M. et al. (1995). *REVIEW ON FISSION PRODUCTS AEROSOL POOL SCRUBBING UNDER SEVERE ACCIDENT CONDITIONS.* A state of the art report. EC Nuclear Science and Technology, Contract No. FI3S-CT93-0009, Final Report. Directorate General XII, Science, Research and Development, EUR 16241 EN, 1995

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[217] A.Powers, J.I.Sprung, "A simplified Model of Aerosol Scrubbing by Water Pool Overlying Core Debris Interacting with Concrete". NUREG/CR-5901, SNL Albuquerque NM, October 1992

[218] Severe Accident Risk: An Assessment for Five US NPP". NUREG-1150, vol.1 December-1990

4.5.8 Melt propagation into ducts and channels

4.5.8.1 Description of accident phenomena

In the ex-vessel phase the core melt will propagate into the rooms below or surrounding the RPV, the distribution depending on parameters including the containment design and the RCS pressure at the time of vessel failure. Then the PSA has to address whether and how the melt might progress further. The key question is whether the containment will remain functional.

In many plant designs there is a concrete cavity below the RPV at the containment bottom. It is a standard issue in PSA to assess the erosion process of the concrete (see section 4.5 of this document). In general, considerable time will pass before the containment fails due to melt penetrating the concrete containment boundary and it is possible that containment venting or failure will already have occurred earlier, due to overpressurisation.

The present section addresses a potentially more direct and faster containment failure mode: If there are ducts or channels at the containment bottom, the core melt could proceed into and through such ducts, and under unfavourable circumstances this could result in containment failure by penetrating them. This could potentially open direct and fast release path to the environment.

4.5.8.2 Tasks within a PSA

First of all it is essential to identify potential routes for the core melt. From the examples given below, it becomes evident that this can be largely plant specific. Therefore, previous experience from other plants may not be sufficient, and it is necessary to make use of all available information, e.g. flow diagrams for ventilation or drainage systems, inspection of the potentially affected areas, photographs, technical drawings.

If potential flow paths have been identified, it has to be assessed quantitatively how far the melt could penetrate and how much melt may be transferred. The following issues should be addressed in this context:

- Potential distribution of melt in the containment and available melt mass in the respective compartment,

- Potential melt level (melt below duct openings is not available for flow),
- Properties of the melt (e.g. temperature below or near liquidus or considerably above liquidus, containing concrete constituents or not),
- Pipe geometry (hydrostatic situation to be considered for melt flow, surface/volume ratio and length of pipe required for melt freezing considerations).

In general, an oxidic melt can flow considerable distances before bulk freezing and plugging of a channel occurs. This is due to the fact that the oxidic melt tends to form crusts on the channel wall which minimise the heat loss from the melt ("conduction freezing model"). If the flow is highly turbulent and/or frozen crusts are instable, the heat transfer to the wall is increased, and channel plugging may occur earlier ("bulk freezing model"). Models have been formulated for liquid metal cooled reactors, but their fundamentals can be generally applied [219].

4.5.8.3 Examples

Protection of penetrations at the bottom of BWR containment

There is a strategy in some BWR plants to flood the lower drywell of the containment to cool down the core melt. However, penetrations exist in the affected area which may be attacked by core debris. Several proposals have been made to protect these penetrations accordingly [220].

Engineered coolant flow path promotes core melt progress

If a loss of coolant into the cavity occurs, it may be necessary to provide a flow path leading from the cavity to the containment sump to make the coolant available for emergency core cooling. After RPV melt-through this same flow path would enable the melt to quickly reach the containment sump. Depending on the sump design, this may lead to severe consequences (see following issue).

Sump suction lines threatened by core melt

Generally there are sump suction lines installed at the bottom of the containment sump which will recirculate coolant to the core cooling system. If core melt reaches the sump, it may also reach the sump suction lines. If the core melt enters these lines, it may proceed inside them until it reaches a position outside of the containment. If stored heat in the core melt and decay heat are sufficient to destroy the sump suction line at such a position outside of the containment, the containment function is lost.

Ventilation ducts in concrete basemat reached by core melt

There are containments which have ventilation ducts within the concrete basemat below the cavity. When core melt erodes the concrete from above, it will eventually reach these ventilation ducts. Depending on the design of these ducts and of the ventilation system, this may open undesirable flow paths for the melt.

Drainage at cavity bottom is weak point for melt retention

There is almost always some kind of drainage at the cavity bottom. Generally, the related piping is narrow, valves are closed and the flow path is long. In such cases there is only little potential for core melt relocation through such piping. However, reactors exist with short drainages leading to rooms outside of the containment, and with valves which are not protected against core melt. In such cases the drainage is a preferred route for failing the containment (see following examples for VVER reactors).

Example for drain lines in VVER-440 [221]: The drain pipe (connected to the waste water cooling system) is built into the reactor cavity bottom, the most part of its length is embedded in the basemat concrete. When melted core debris enters the cavity, it can be presumed that the flap valve at the front of the channel made of thin (not thicker than 10 mm) material is very likely to become damaged, and then the melt flows into the pipe. The melt behaviour in the drain pipe was studied with simple code calculations. The results of the sensitivity calculations show that the solidification time under such conditions is more affected by the decay heat of the melt and the pipe diameter, than by the heat transfer coefficient. Cooling of the melt's front section to the solidus temperature (not to total solidifying) takes a relatively short time (1-35 s). Quite considerable penetration length can be expected for the case of high inlet temperature and high decay heat of melt. The pipe can be filled with corium if initial melt superheats are high. The melt may damage the outlet section of the pipe and pour into the collector tank of liquid waste.

Example of instrumentation lines in VVER-1000 [222] : There are 27 holes (108 mm diameter) situated in the concrete wall of the reactor cavity at a radial distance of 150 mm from the inner side (RPV side). Neutron flux measurement devices are located in these holes. These holes are connected to the room situated below and this room is outside of the containment, so radial ablation of 150 mm of concrete by molten corium leads to a loss of containment function. Ablation of this small layer of concrete is estimated to last less than an hour. Therefore, failure of the reactor pressure vessel will result in large releases into the environment about one hour later through this pathway. There is another set of 27 holes (62 mm diameter) situated at a radial distance of about 300 mm from the first set.

4.5.8.4 References

[219] [European applied research report – Nuclear Science and Technology, Vol 6, No 5(1985), pp1323-1332]

[220] [K.M. Becker et al., "Enhancement of core debris coolability", KTH-NEL-51, Stockholm, May 1990].

[221] [AGNES Project: Containment Phenomena During Severe accidents Part II VEIKI report March 1994]

[222] [Phare project BG 011001 BG01.10.01: Phenomena investigation and development of severe accident management guideline]

4.6 CORIUM RECRITICALITY AND REACTIVITY ACCIDENTS

One of the most important safety issues in NPPs is to control criticality of the core in all operational states and during operational transients and accidents. Additionally, in severe accidents this issue has to be taken into consideration and core or corium recriticality issues should be studied in L2PSA. In a severe accident, core recriticality might be possible during core melting if conditions in the core are changed for some reason to favour core recriticality. Recriticality of the corium should also be studied particularly after RPV failure if corium might contact non-borated water. In this section the most important recriticality issues have been described. Some information considering reactivity accidents caused by heterogeneous boron dilution has also been included in the section, since this might become an important issue for L2PSA particularly in shutdown states.

4.6.1 In-vessel corium recriticality

4.6.1.1 Introduction

During a severe accident occurring in a BWR, while the core is in the process of heating, the steel blades containing B_4C control material are likely to reach the melting point before of the fuel.

The neutron absorber material is encased in stainless steel blades, which have a melting point lower than fuel and cladding ones. As the core heats-up, the control material may melt and leave the core. After the relocation of control blades material, but before the core melting, the core could be flooded with unborated water, with a possible re-criticality event.

It should be remembered that this kind of event is prevented in PWR cores by flooding the core with borated water.

4.6.1.2 Accident scenarios

The risk assessment of BWRs demonstrates that accident sequences as station blackout and Anticipated Transients Without Scram (ATWS) events, can be selected to characterise the core melt scenarios where re-criticality may be possible.

Station blackout

Station blackouts are defined as the loss of all the AC power (both normal AC power source from the off-site electrical grid and the emergency AC power source from the on-site diesel generators) and can be divided into two groups based on the timing to reach the core damage: Short-term Station Blackout (SSBOs) accident sequences (core damage within 1 hour of the initiating transient or event) and the Long-term Station Blackouts (LSBOs) events (core damage within 9 to 12 hours after the initiating transient or event). These two groups of station blackouts are essentially similar as consequences, with the exception that core power is lower in the long-term sequences because of the lower decay power.

If the whole electrical power is lost, the capability to cool the core is lost. The water level decreases, the core temperature increases, then causing core damage. In both short-term and long-term station blackout conditions, if AC power is restored in the plant and un-borated injection starts just within the time window between the beginning of the control blades melting and the beginning of the fuel rod melting, the potential for re-criticality condition can occur. It is estimated that between 1 % and 12 % of events, depending on the specific sequence, AC power will be restored and coolant injection will be initiated within the re-criticality window. Due to the fact the core damage proceeds from the centre to the peripheral zones, the potential of re-criticality will occur first in the central zone of the core and then in the peripheral ones.

Anticipated Transients Without Scram (ATWS)

The ATWS events are characterised by the failure of control rod insertion into the core (mostly mechanical failure of the control rod system) upon receipt of a scram signal following an unspecified transient. In some sequences, various systems used to recover from an ATWS event fail, so that the water level in the vessel decreases and the core damage begins. If the water injection is subsequently initiated, a re-criticality becomes possible.

4.6.1.3 Consequences of recriticality

The consequences of a re-criticality have to be characterised to qualitatively select the most appropriate and effective accident management strategies. If the available water supply is insufficiently borated, it might be preferred not to re-flood the core, to avoid very severe consequences. On the other hand, if the consequences are estimated to be minor, the current procedures are to immediately reflood the core with the maximum available water flow rate (borated or unborated). The possible core conditions arising from the accident scenarios are summarised as follows.

Super-prompt critical excursion

The primary core condition from the accident scenarios is the super-prompt critical excursion which results in rapid disintegration of fuel, rapid molten fuel-coolant interaction, and production of a large pressure pulse capable of direct failing the reactor vessel integrity. Analyses conducted on this subject indicate that the rapid disintegration of fuel is unlikely under conditions of reflooding a hot core. The Doppler feedback is estimated to be the main mechanism terminating rapid transients in low-enriched uranium-water systems thus limiting the energetics of reflood re-criticality. Concerning the energy deposition in the fuel during power excursion, the results of the recent SARA project, supported by numerical simulations [224], differ from those provided by early studies. As a matter of fact, preliminary studies performed at the beginning of 1990s [223] concluded that the energy deposition in the fuel due to super-prompt power excursion would be below the threshold for fuel fragmentation and dispersion (about 8.40×10^5 J/kg – 1.2×10^6 J/kg for low burn-up fuel and to about 2.93×10^5 J/kg for high burn-up fuel). On the contrary, the SARA results suggest that, for high reflooding rates (several hundreds of kg/s), these threshold are approached or exceeded, in some cases with large margin, so that the risk of fuel fragmentation and dispersal during a re-flooding transient cannot be excluded.

Core remaining critical after initial power excursion

If the re-flooding is managed without boration, and the reactor remains in critical conditions following an initial excursion at the time of re-flooding, it can either enter an oscillatory mode as consequence of the fact that water periodically enters and is expelled from the core or approach a quasi-steady-state power level. In both cases, the average power level achieved will be determined by the balance between the reactivity added and the feedback mechanisms. The coolant void fraction and the fuel temperature are other factors affecting criticality.

Analyses performed within the SARA project showed that a recriticality event is likely to produce a core power level lower than about 20% of the nominal power, but significantly above the decay heat level.

4.6.1.4 References

[223] W.B. Scott et al. Recriticality in a BWR following a core damage event. NUREG/CR-5653 (1990)

[224] W. Frid et al. Severe Accident Recriticality Analysis (SARA) 1999

4.6.2 Ex-vessel corium recriticality

4.6.2.1 Introduction

In case of a severe accident with spreading of corium melt in the reactor cavity, it is a difficult task to evaluate the exact distribution and composition of fuel and its mixing with other structural materials used in the fuel assemblies (claddings, spacers, tube guides, etc.), as well as the stainless steel of the vessel. The presence of fuel and coolant within the reactor cavity suggests a warning because, in some geometrical configurations, this can potentially lead to recriticality; the likelihood of a chain reaction could be increased by the presence of non-borated water, or produced by phase separation phenomena promoted by steam condensers working in the containment atmosphere (water coming from condensed steam may have a low boron concentration).

4.6.2.2 Condition for criticality

The main parameters affecting the risk for re-criticality in the corium are as follows:

- Physical properties of fissile materials loaded in the core;
- Concentration of the fissile materials in the corium;
- Shape and dimension of corium fragments;
- Ratio between number of nuclei of fissile and any moderator material i.e. degree of moderation ($H/^{235}U$, $C/^{239}Pu$, etc.);
- Presence of materials with properties of reflectors or poisons for neutrons;
- Temperature of the corium melt.

Type, density and enrichment of fuel directly affect the neutronic properties of system by the values of fission cross sections, and the parameters ν and η (average number of neutrons emitted for fission and average number of neutrons emitted for neutron absorbed in the fuel). As the critical dimension of a system is inversely proportional to its density ρ, for bare finite geometries such as spheres, cubes and finite cylinder, the relation between the critical mass m_c and ρ can be expressed approximately as $m_c \sim 1/\rho^2$.

Shape and dimension (i.e. the ratio surface/volume) have influence on the neutron leakages of the system and, therefore, on the minimum dimension for criticality. The dilution of the fissile concentration in the corium melt by mixing with structural materials (whose effect is to negatively affect the neutronic balance), invariably reduces the potential of re-criticality.

While un-moderated systems containing highly enriched uranium show a critical geometry progressively increasing with the content of ^{238}U, up to become unbounded for systems whose enrichment in ^{235}U lie between 5 – 6 % (see [225]), uranium at low enrichment can be critical with the addition of a third material as moderator (usually hydrogen, deuterium, beryllium, or carbon). The apparent effect of a neutron moderator mixed with fissile material is to increase the critical mass. However, due to the reducing effect on the neutron energy and subsequent increasing in the spectrum-averaged fission cross sections, as the volume fraction of moderating diluent is increased, the critical mass is reduced to a very low value. [225] reports the calculated values of the critical radii of a bare and water-reflected sphere composed by a mixture water - ^{235}U at 5% of enrichment (both in the form of metal and oxide) as function of the ^{235}U concentration in the mixture (the lower bounds are referred to the water-reflected spheres). As depicted in the figure, these enriched systems show a minimum as function of the ^{235}U contents in the mixture and two asymptotes at low and high degree of moderation (the minimum critical volume for the system metal-water system occurs at the contents in ^{235}U of about 0.1 g/cm^3 (corresponding to 2 g/cm^3 of U in the mixture), while the effect of the oxide on the minimum is to cause a shift to the value of 0.085 g/cm3) [226].

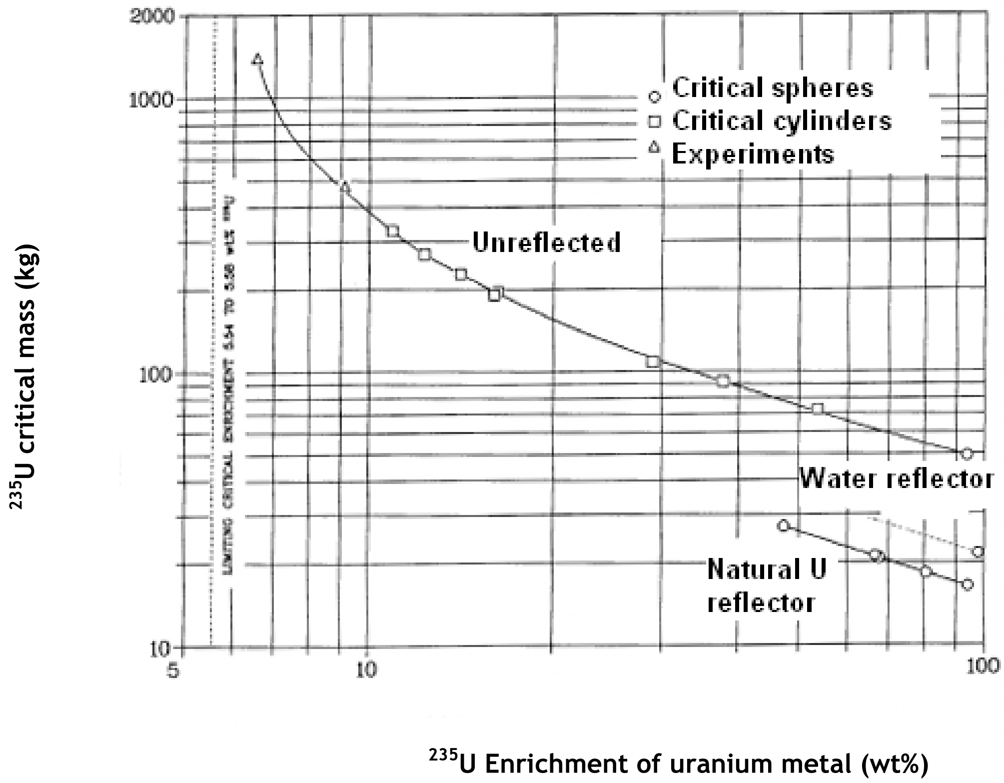

Fig. 49 Critical mass of un-moderated system vs. ^{235}U enrichment of uranium metal (for un-reflected, water and uranium reflected systems). The dashed line represents the enrichment below which a piece of uranium metal cannot become critical [225].

Particular attention should be focussed on those systems containing a significant amount of Pu (core loaded with MOX or pure PuO_2) due the lower amount of fissile material needed to reach the criticality conditions (critical mass m_c) and that Pu mass can reach the supercritical conditions in the rapid spectrum.

The presence of large amounts of reflector materials, mainly cooling water and steel from vessel but also concrete, can affect the criticality of the system. It is well known that the reflection effect is essentially the maximum attainable with water; a water layer up to 3 - 4 cm is a very efficient reflector; an increase of thickness beyond several centimetres (i.e. 10 cm or more) adds little to its influence (at very low energy and after the passage through several centimetres of water, neutrons are more likely to be captured by the hydrogen than reflected back). Concrete is common as structural material and has somewhat better reflector properties than water, specially if fitted closely to the fissile material (the presence of gaps can reduce the reflection effect).

Hydrogen and ^{238}U capture neutrons readily enough to behave like mild neutron poisons (^{240}Pu is similar). Carbon and oxygen have a negligible poisoning effect while nitrogen is a mild poison. Among structural materials, aluminium has a small neutron-capture effect, copper and component of steel are mild poisons.

The fuel temperature also has an important effect on the reactivity of the system; an initial subcritical system at high temperature, if adequately cooled, could become critical or supercritical due to the lack of negative reactivity feedback effect of the neutron resonance captures of ^{238}U and ^{239}Pu in the epithermical region (Doppler effect).

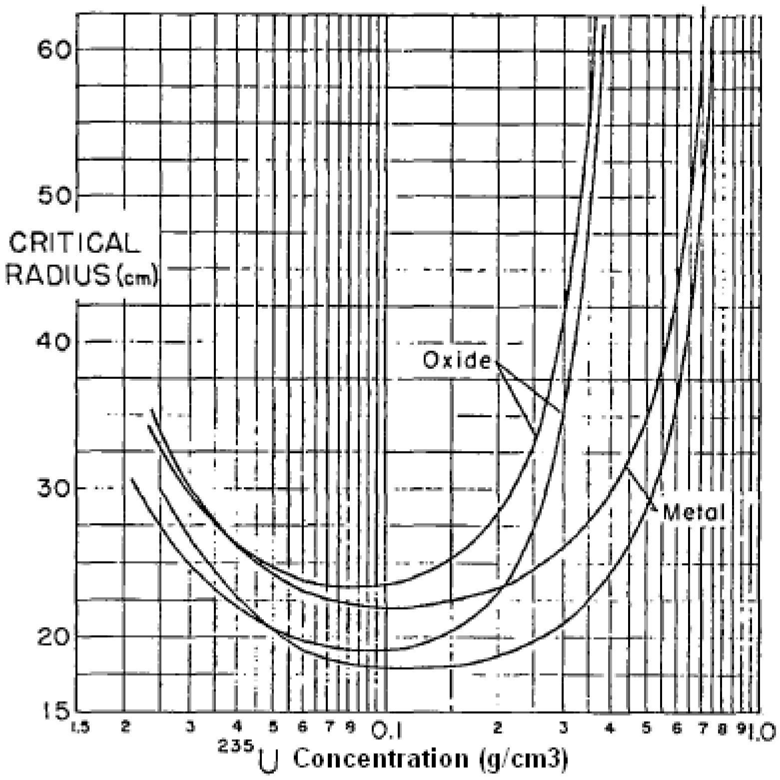

Fig. 50 Critical radii of bare and water reflected spheres of homogeneous mixtures water - 235U (5% enrich.) vs. 235U density [225].

4.6.2.3 Conditions favouring the development of re-criticality events in LWRs

In case of a severe accident with ex-vessel spreading of corium melt in the reactor cavity, the conditions that could affect the potential risk of re-criticality are mainly related to the presence of moderators (as un-borated water in case of flooding of the reactor cavity) and materials acting as neutron absorber (poisons).

In absence of a significant corium-concrete interaction (for example, if any early cavity flooding and corium freezing is promoted), no neutron absorbers could be present. In fact, at the temperatures expected for molten core interaction both with cavity materials and coolant, the typical PWR control rod materials are likely to be completely vaporised (silver, indium and cadmium boil at 2485 K, 2353 K and 1038 K, respectively) or segregated from the fuel material by early relocation, as in TMI-2 scenario [227].

In BWRs, the B_4C of control blades has a boiling point at approximately 3800 K. Although the reactor kinetics need to be investigated, this boron is likely to react with steam, forming boron oxide. Boron oxide has its boiling point at approximately 2100 K. As shown in some tests, B_4C can also react with steel from the control blades and relocate into the vessel lower head. Thus, separation of fuel and neutron absorbers could also be maintained in ex-vessel conditions, increasing the likelihood of the development of possible re-criticality events.

4.6.2.4 Design measures preventing ex-vessel recriticality

The main design measure able to prevent the recriticality of corium melt spread in the reactor cavity is the adoption of a crucible core catcher containing absorber materials for neutrons (the poisons most suitable for this purpose are boron, cadmium and the rare earths as samarium, europium and gadolinium). The main aim of core catcher is firstly to favour the sub-criticality of the system by splitting the potential critical mass (geometrical division) and secondly to increase the surface area in contact with absorbent materials to increase the neutron capture. Several solutions have been proposed on this matter, for instance the implementation of a core catcher containing hafnium di-boride (HfB_2), a refractory ceramic and absorbent component with an efficiency similar to that of B_4C, for those PWR cores loaded with MOX and PuO_2 [228].

The ablation of the sacrificial material forming the core catcher, and the subsequent mixing with fissile materials, should further prevent from the risk of re-criticality.

4.6.2.5 Application to L2PSA

Generally ex-vessel corium criticality is screened out from L2PSA. Detailed calculations with separate codes are needed to assess the conditions and justify screening out. Calculations of the recriticality potential of corium when catched in concrete structures and cooled by water, have been performed in the past [226]. In the hypothesis that a uniform layer of corium is released on the concrete and cooled by water from above, the potential for corium recriticality strongly depends on the porosity of the debris bed and on thickness of the water layer covering the corium. However, even if significant values of porosity for debris bed are considered (20-25%), the risk of corium recriticality seems to be precluded. As example of such calculations, trend of corium reactivity (k_{eff}) vs. water thickness, for a debris bed porosity of 15%, is reported in the figure 46.

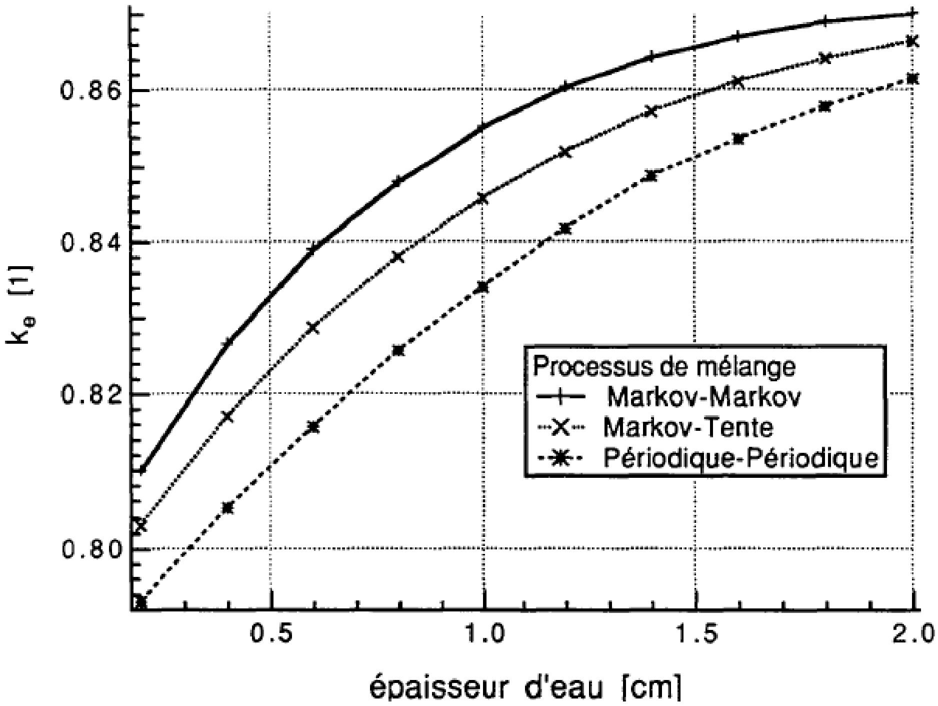

Fig. 46 – Corium reactivity (k_{eff}) vs. water thickness for 15% porosity of debris bed (multi-energy transport calculations by using three different homogenization techniques of the macroscopic cross sections – from ref. [226]).

If the corium - concrete interaction occurs, the presence of species with low mass number in the concrete, increases neutron thermalization and then the risk of recriticality at high degree of fragmentation.

In cases where design measures to prevent corium criticality are introduced, the phenomena can be screened out.

4.6.2.6 References

[225] H.C. Paxton et all. Critical dimensions of systems containing U235, Pu239 and U233. LA-10860-MS, Los Alamos, 1987

[226] W.R. Stratton. Criticality data and factors affecting criticality of single homogeneous units. LA-3612, Los Alamos, 1967

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4.6.3 Core recriticality initiated by rapid heterogeneous boron dilution

4.6.3.1 Introduction

Boric acid is used as a soluble neutron absorber for long-term reactivity control in pressurised water reactor. The main functions of boric acid are to compensate for fuel burn up and xenon poisoning with the required reactivity margin during normal operation, and to provide the necessary subcriticality of the core during refuelling and maintenance[19].

During normal reactor operation, the specified boron concentration is maintained by the Chemical and Volume Control System (CVCS). In the case of a loss-of-coolant accident, borated water is injected by the Emergency Core Cooling System (ECCS) by high-pressure safety injection pumps, accumulators and low-pressure pumps.

In certain scenarios it is possible that non-borated water or water with reduced boron concentration could flow to the core causing an increase in reactivity level. This chapter describes the phenomena and introduces different possible Boron Dilution Accident (BDA) scenarios.

4.6.3.2 Description of possible scenarios

If an accidental reduction of the boron concentration in the core region (referred as Boron Dilution Accident or BDA), occurs early in the fuel cycle, there may be sufficient excess reactivity in the core for the de-borated coolant to bring the core critical even though all the control rods have been inserted. The possible power excursion may be sufficient to cause severe damage of the core, even though the emergency core cooling system has successfully kept the core covered with coolant[20]. The reactivity accidents caused by boron dilution of the coolant that result of more interest are those occurring when the plant is in cold shutdown state, the pressure is very low and there is no coolant circulation (cold zero load conditions).

In the slow boron dilution accident considered usually in the Safety Analysis Report (SAR) , the addition of diluted water is gradual enough that the core flow can anyway assure adequate mixing in the reactor coolant system (homogeneous dilution), resulting into slow and low reactivity rise. These slow changes in the core boron concentration leave sufficient time for the identification of the problem and operator intervention.

Alternatively, if the addition is large compared to the mixing capability (heterogeneous dilution) there is the possibility that plugs of diluted water rapidly enter the core, resulting into a severe reactivity excursion. Such accident scenarios can be initiated for instance by:

- Discharge of diluted water from ECCS during plant refuelling. This BDA is due to a blowdown of a completely diluted accumulator during plant refuelling, assuming the water at the core inlet to be uniformly diluted. For this BDA accident, the correspondent check valve is supposed to be open, all

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[19] The smaller the fuel burn up is, the higher the concentration must be.

[20] Unacceptable fuel damage is determined by the local fuel (pellet-average) enthalpy. For instance the NRC currently uses 280 cal/gr as value which would lead to unacceptable fuel damage, although recent experimental researches have indicated that this failure limit may be significantly lower, particularly in high burn-up fuel conditions (100 cal/gr).

pumps are in standby conditions and the coolant circulation is initiated by the pressure difference between the accumulator gas and the coolant pressure in the piping and components, until an equilibrium value is reached.

- Start up of reactor coolant pump(s) after improper boron dilution. This BDA is due to the injection of un-borated water pushed by the start-up of one Reactor Coolant Pump (single RCP start-up). The clean coolant is pumped through the core causing a reactivity rising. The following accident scenarios involving RCPs can be mentioned [1]:
    1. *Scenario A (dilution during RCS filling)*. The reactor is shutdown and being cooled by the Residual Heat Removal (RHR). The steam generator tubes are drained, but the Reactor Cooling System (RCS) is in the process of being filled and pressurised. As a result of a Chemical and Volume Control System malfunction or operator error, for some period of time diluted water is supplied to the charging pumps, which in turn supply seal injection water to the RCPs. This supply water will be colder than the RCS coolant, and will run into the RCP suction piping and displace the water that is already there. When the RCP is started in that loop, the clean water pocket is swept into the core.
    2. *Scenario B (steam generator inleakage - The "Swedish Scenario")*. The reactor is shutdown and being cooled by the RHR system. As a result of a steam generator tube leak (most likely caused by improperly completed steam generator maintenance or inspection), secondary water enters the reactor coolant system, and collects in the RCP suction piping, steam generator outlet plenum, and perhaps in the steam generator tubes. Subsequent start of the RCP sweeps the clean water into the core. Apart from the mechanism of introducing unborated water into the RCP suction, it is identical to Scenario A.
    3. *Scenario C (Loss of AC Power during Dilution - The "French Scenario")*. The reactor has just been refuelled and is the process of being started up. A boron dilution (toward the critical boron concentration) is in progress when a loss of offsite power occurs, resulting in the trip of all RCPs. Decay heat is low and natural circulation does not occur in the reactor coolant loop(s) receiving the diluted changing flow from the Volume Control Tank (VCT). Emergency power comes on and automatically restores the charging flow. In the absence of alarms drawing the attention to the dilution progress, the operators fail to secure the dilution and the entire volume of the VCT is discharged into the RCS. When the offsite power is restored, the operators restart an RCP, and the clean water is swept into the core. This boron dilution accident is sometimes referred to as the "French Scenario" since it was first considered in France by EDF.
    4. *Scenario D (Boration after shutting off RCPs)*. During RCS cooldown at the beginning of a refuelling shutdown, the RCPs are postulated to be turned off at a relatively high temperature (such as 60 °C) and before borating to refuelling boron concentration. With hot water left in the steam generator, RHR flow would be unlikely to force circulation through the steam generator tubes, with formation of a stagnant pocket of low boron concentration.

During subsequent draining and refilling, this pocket would remain in the crossover leg and/or steam generator. Common PWR practice is to continue RCP operation until refuelling boron concentration is reached.

An other scenario is described hereafter (provided by IRSN from French PWR PSA) [233]:

On PWRs, some components of the reactor coolant pump (RCP) are not designed for hot temperatures. A thermal barrier ensures the thermal insulation between these components and the volute casing which is at the same temperature as the primary circuit.

The thermal barrier consists of a heat exchanger cooled by the component cooling system (CCS). Water in the CCS system is cold and unborated. At power states, pressure in the CCS is below primary coolant pressure.

In case of a leak on the thermal barrier at a power state, a leak rate will flow from the primary circuit to the CCS. It will be detected due to the changes in the temperature of the CCS or due to its flow or its level in the buffer tank of the CCS and an alarm in the control room will inform the operators.

The corresponding procedures ask operators to isolate the thermal barrier from the rest of the CCS by closing valves located upstream and downstream the thermal barrier and to depressurise the primary circuit.

According to the procedure, the affected pump must be stopped by the operators.

Assuming that due to an error of the operators the thermal barrier is not isolated from the rest of the CCS, the leak rate will reverse from the CCS to the primary circuit during the depressurisation and lead to the dilution of the primary coolant in the cross-over leg of the affected loop, between the steam generator and the RCP. As a result, a significant volume of unborated water is accumulated in the primary circuit.

If operators do not drain this volume and start up the pump of the affected loop before starting up the pump of another loop, the unborated volume of water will be injected in the core.

Additionally, other mechanisms have been identified that lead to heterogeneity in boron concentration without any external source of diluted water. One of these considers the dilution mechanism via accumulation of boron free condensate in the cold leg loop seals, due to reflux/boiler-condenser mode operation (decay heat removed by phase separating natural circulation), during certain accidents, such as small break loss of coolant accidents (LOCAs)[21]. The subsequent change in flow conditions, such as re-establishment of natural circulation flow, may provide an effective mechanism to drive the slug of diluted water into the core.

[21] The steam that is generated in the core is largely devoid of boric acid. Due to subsequent condensation in steam generator, a portion of boron-free condensate, can run down the downflow side of steam generator tubes and accumulate in the loop seals between the steam generator outlet plenum and the reactor coolant pumps.

4.6.3.3 Example of consequences of re-criticality

As example of core neutronic transient in a typical (2-loops and four pumps) PWR 1900 MWth power rated, during a boron dilution accident initiated by spurious start up of reactor coolant pump, figure (Fig. 51) reports a typical peak power trend during such postulated accidents, as calculated by numerical simulation. The initial plant conditions are of all rod inserted (ARI), cold zero power (CZP), no Xenon and Samarium, core coolant with initial boron content of 2000 ppm; the core coolant is progressively de-borated by the injection of 25 % of the nominal rated coolant flow. Significant increasing in the fuel temperature T_f and enthalpy H_f is expected for more stressed bundles (see Table 36) as the power level as well (Doppler and moderator feedback limit the peak). In this example the power excursion in the core occurs in correspondence of a reduction of the boron content of 400 ppm with a reactivity insertion in the core of approximately 6000 pcm.

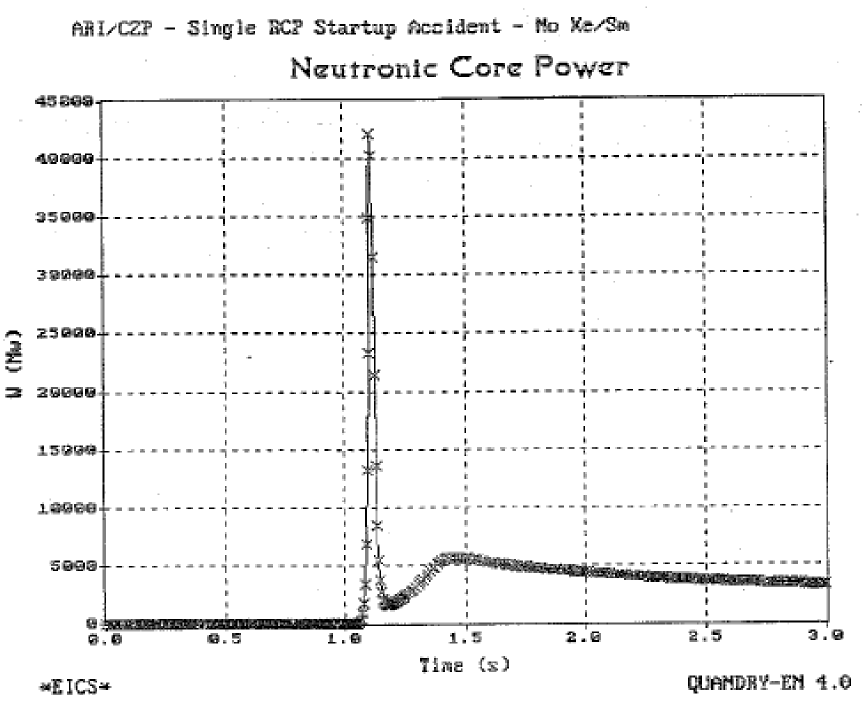

Fig. 51 Example of peak power trend for a BDA caused by spurious RCP start up [2]

Table 36 Example of core parameter trends for BDA caused by spurious RCP start up [2]

Time (s)	T_f ave (K)	T_f max (K)	H_f max (cal/gr)	P^{max} (MWth)	b (ppm)
0.25	333.0	333.0	2.7	257×10^{-6}	1911
0.50	333.0	333.0	2.7	327×10^{-6}	1822
0.75	333.0	333.0	2.7	749×10^{-6}	1733
1.00	333.0	333.0	2.7	228×10^{-3}	1644
1.05	330.0	333.2	2.7	32.9	1626

1.10	342.7	432.5	8.6	23.3×10^3	1608
1.15	390.6	901.9	43.6	2702.3	1590
1.20	393.2	925.9	45.4	1775.1	1572
1.25	396.4	955.9	47.8	2418.6	1555
1.50	438.7	1307.3	76.0	5582.9	1465
1.75	482.9	1604.9	100.6	4827.4	1374
2.00	517.9	1779.4	115.6	4352.5	1281
2.50	570.0	1943.5	130.2	3628.7	1091
3.00	608.0	2002.5	135.7	3251.2	895

Legend

T_f^{ave} = average fuel temperature

H_f^{max} = maximum fuel node enthalpy

b = boron concentration in the core

T_f^{max} = maximum fuel temperature

P^{max} = max. peak power (MWth)

From the previous example, it is clear that scenarios with large volume of unborated water anywhere in the RCS appear capable to induce significant power excursion. If the accident occurs early in the fuel cycle (e.g. in plant shutdown after refuelling), there may be sufficient excess reactivity in the core for the unborated coolant to bring the core to criticality even though all the control rods have been inserted. Due to the rapid progression of the transient, Doppler feedback is the only mechanism which is able to quickly limit the power increase. However, the consequences of the event must be put into perspective by considering the probability of having a pump restart and how that probability changes the core damage frequency for the event. With reference to the SBLOCA scenarios with RCP start-up and unborated plug entering in the core, analyses [232] performed on typical PWR B&W (Babcock & Wilcox) designed plant, led to the evaluation of a very low core damage frequency (around 10^{-8}/reactor-year). Scenarios with a large volume of unborated water upstream of a RCP appear capable of causing severe core damage as a result of a rapid boron dilution._However, the necessary volume of unborated water appears larger than the crossover leg piping itself, and should therefore be both detectable and preventable[22].

4.6.3.4 Application to L2PSA

The reactivity issues are often screened out from the L2PSA. This is possible if core damage frequency arising from L1PSA is sufficiently low. However, in case of reactivity accidents it should at least be checked in L2PSA that the definition used for core damage is suitable for L2PSA purposes. Partial core damage might be screened out from L1PSA, but in L2PSA these sequence might cause unacceptable releases. For example, if reactivity

[22] Possible precautions able to unsure a very low probability of occurrence for the above scenarios could be for instance: a) do not dilute unless at least one RCP is running, b) restrict dilution to conditions under which natural circulation would be expected if the RCPs were tripped, c) when borating for a shutdown, keep at least one RCP running until the desired boron concentration is reached.

transients during plant shutdown state are found to be possible, the reactivity accident might lead to unacceptable releases even though the core would not be totally damaged if the RPV is open and only water pool separates fuel from containment atmosphere. Fission products that are released from the fuel due to reactivity accident are partly retained in the water pool, but gaseous forms of fission products will be released to the containment. It should also be remembered that during shutdown the containment might not be leaktight.

If the containment is leaktight, the possible consequences to the containment should also be assessed. It can include analysis of fuel cladding and vessel failure following a reactivity accident.

4.6.3.5 References

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5 CONTAINMENT PERFORMANCE

5.1 INITIAL CONTAINMENT PERFORMANCE: PRE-EXISTING LEAKAGE AND FAILURE OF ISOLATION SYSTEMS

5.1.1 Introduction

Fission product confinement and control is one of the main safety functions that the containment shall address and initial containment leaktightness and isolation are important issues in L2PSA.

There are several important containment boundary penetrations in all containment designs. When containment isolation succeeds, and the containment integrity is maintained, the containment will still have a certain pre-existing leak rate, typically called the design leak rate, which depends on the containment design. Depending on the NPP design, there may be additional rooms and volumes around the containment boundary (e.g. the reactor building) which constitute another obstacle for radionuclides, and which may decrease and/or delay radionuclide releases to the environment. Nevertheless, these pre-existing leak paths mean that there will be some flow to the environment in all accident sequences, even when the containment integrity is not threatened by any phenomena. In accident sequences where radionuclides are released into the containment, these pre-existing leak paths will result in gas flows to the environment and, depending on the potential for retention mechanisms in the leak path, releases of radioactivity into the environment. In most recent containment designs this leak is very small and pre-existing leakage will not result in substantial environmental releases. However, the leak from the containment to surrounding plant rooms might have an effect on the plant accessibility and the doses for the operational personnel responding to the accident.

At the time of an accident, containment penetrations can be initially open or closed. During the accident, front line mitigating systems may require some penetrations to be open, while others must be closed by the automatic containment isolating signals. Also, severe accident management may require total closure of the containment after a mitigating phase. There is obvious potential for radionuclide releases to the environment to occur due to failure of some penetrations to be isolated or properly sealed due to the equipment failure, loss of operating power (pressurised air, electricity...) or unavailability due to maintenance work.

In certain situations, such as during a maintenance outage, there might be penetrations that are not normally open during power operation. In particular, some large leakage paths may be present. For example, equipment hatches that are open during some reactor outage operations may result in a large leakage path to the environment.

The following chapters will introduce how these issues should be addressed and modelled in a L2PSA. Background material considering containment design and initial containment performance can be found from European utility Requirements (EUR) section 2.9 [234] and IAEA Safety Series document considering containment design [235].

5.1.2 Initial containment performance

Initial containment performance and the magnitude of the pre-existing leak rate from an intact containment depends on many issues including the containment design, materials used and the number and size of containment penetrations. The design leak rate typically consists of many very small leakages through sealed penetrations i.e. instrumentation penetrations, leakage through structural materials (i.e. concrete) or leakage through structural joints.

For a new plant design, the pre-existing containment leak is specified as the design leak.

The design leak rate is usually defined as the leak rate from the contained mass of free gas and steam per day at the design pressure (and temperature) conditions of the containment. Typically, this leak rate is defined in the NPP technical specifications. Periodic integrated leak rate tests are the usual way to verify that the design leak rate of the containment is maintained. The control of containment leaktightness is mainly carried out by the containment leak rate testing programme consisting of local leak rate tests (LLRT) and integrated leak rate tests (ILRT). LLRT verify the leaktightness of single equipments and periodic ILRT verify that the overall design leakage of the containment is below the technical specification limit.

Typical design leak rates for a concrete pre-stressed primary containment vary in a range from 0.5 % to 1 % of the contained mass of free gas and steam per day. For other containment types the design leak rate might be smaller, which is the case for containments with a steel liner where the design leak rate is typically between 0.1 % and 0.5 % per day. However, in some older containments the design leak rate might be significantly higher, up to 15 % per day. For L2PSA purposes, the design leak rate of the containment under study should be used.

It is recommended to use actual measured leakage rate data when available. Expert judgement can be applied to define the method (average over many years, weighted average or last measured data) of processing of the used data.

Normally periodic integrated leak rate tests are preceded by efforts to reduce each potential leak path. Only after this procedure is the leak test performed and, normally, the design leak rate can then be verified. As long as no event occurs affecting the leaktightness after the test, the PSA could assume the measured leak rate, or the design leak rate. However, if interventions with potential influence on the leaktightness have occurred after the test, there is a non-negligible probability that the actual leak rate of the containment is not limited to the design value. This may be the case in particular if the containment leak test is not performed after each major shutdown period. Document [234] provides the results of measured leak rates for a significant number of containments. A considerable fraction of the containments had larger leak rates than the design leak rate.

In severe reactor accidents, the containment pressure will be higher than the design pressure in many cases, but the containment will not fail until the ultimate failure pressure of the containment is reached. This could be the case if the systems used for containment heat removal fail i.e. slow containment over-pressurisation sequences. However, when the containment pressure is higher than the design pressure, the containment leak rate will be higher than the design leak rate. To take this into account, the leak rate should be defined as a function of the containment pressure. The assumed leak rate for pressures higher than the design pressure shall be adequately supported by analytical considerations, by experimental data, and by design and testing provisions. However, if the sequences analysed in the L2PSA, where the containment pressure is higher than the design pressure, quickly result in containment failure it might not be necessary to put a lot of effort into defining the containment leak rate as a function of pressure. This depends on the studied containment design and should be assessed on a case by case basis.

When assessing the leak rate through small leak paths, it is usual to set the leak rate to the defined value at a pre-defined pressure level. However, there are differences in leak rates at lower pressure levels depending on the geometry of leak path. A circular flow path results in turbulent flow, whereas a narrow flow path has a much smaller hydraulic diameter and thus the flow is in the laminar regime. Flow rate in the laminar regime is more or less directly proportional to the pressure drop through the flow path, but this is not necessarily the case with turbulent flow. Fig. 52 below tries to illustrate the difference. The circular path may result in flow rates 2 to 3 times higher than those in a narrow gap, at certain pressure levels below the pre-defined pressure level. Whether this effect has to be taken into account or not depends on the results from source term calculations that may show that, even in the worst case, the releases are of very low magnitude. However, it is an issue to be considered when dealing with uncertainty analyses.

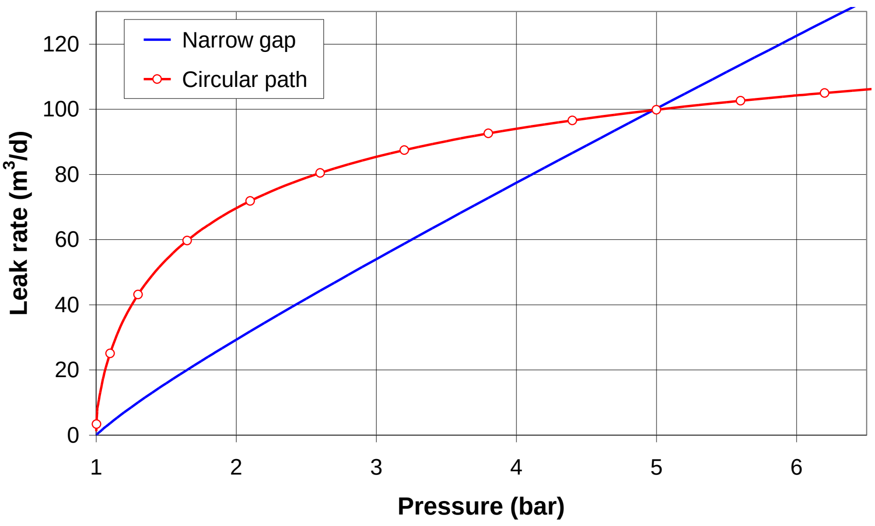

Fig. 52 Comparison of leak rates in two types of leak paths

The leak rate is set equal at an absolute pressure in the containment of 5 bar, assuming the pressure level of 1 bar outside the containment. The blue line shows the pressure dependency of the leak rate in a narrow gap, and the red one with a circular path. The path length in both cases is set to 100 cm. Steam partial pressure and temperature effects on the gas properties are modelled in some detail.

The intact containment design leak rate is part of the overall containment performance and it is related to containment structural issues, which are considered in chapter 5.3 of this guideline. Whichever approach is taken, the design leak rate (especially for new plants), or its modification by a function to account for any periodic containment leak test results, or directly the historical leak test results should be taken into account to define the containment leak rate before the accident.

In some reactor designs, the containment leaktightness can be monitored online. For example for French PWRs, a system is able to detect on-line a large containment leak e.g. failure to close the personnel access lock. This type of system may provide some upper bound of the initial containment leakage in a L2PSA.

As discussed, the significance of the radionuclide release via pre-existing leakage depends on the design leak rate. This release can only be mitigated by ventilating the leak flow through systems with effective filters, i.e. a secondary containment. However, it is not always practicable to capture all design leak paths within a secondary containment envelope. It should be understood that the effective containment design leak rate is composed of many potential release routes and, in the case where ventilation systems fail or the capacity of the filters is insufficient, this will have an effect on both the environmental release and the accessibility of rooms outside the containment along the leak path.

5.1.3 Failure of isolation systems

There is no single system for containment isolation, as many separate components of different systems will be needed to isolate the containment successfully. All the containment penetrations should be accounted for in the evaluation of containment leakage paths. Penetrations shall be analysed for both the following types of release pathways:

- Failure to isolate normally-open or normally-closed lines that might be open at the time of an accident, e.g. due to mal-positioning of valves, due to the initiating event, due to human error or due to additional failures;
- Leakage through a penetration where no pathway should be present, e.g. leakage past the seat of a closed valve or through a normally closed and sealed penetration which is open for maintenance purposes, etc.

Identification of containment penetrations is an important part of the plant specific containment performance assessment. Containment penetrations can be:

- Equipment hatch(es);
- Personnel hatch(es);
- Piping penetrations;
- Electrical (cable) penetrations;
- Instrument (cable and piping) penetrations;
- Purge line(s);
- Vent line(s).

Normally there are two barriers in each containment penetration, one inside the primary containment and one just outside the primary containment. These barriers can be: two automatic isolation valves, an isolation valve and a normally closed isolation valve or automatic isolation valve, and a normally closed or sealed system or tank either inside or outside the containment. Containment hatches and access doors are normally closed (or closed and sealed). Some penetrations that are normally closed and sealed during power operation will be in use, and not closed, during maintenance outages. This should also be taken into account when the likelihood of containment isolation is assessed.

For L2PSA purposes, all the containment penetrations shall be carefully analysed to decide if they should be included or not. Not all containment penetrations have the potential to be important pathways for radionuclide releases. To focus the L2PSA effort on the penetrations that are most likely to be important, screening criteria may be applied. This should be done carefully, case by case. Containment penetrations can be screened from the analysis if they can meet one of the following criteria:

- Closed loop inside or outside the containment. Any system that starts and terminates in the containment, without any release path to the environment, can be excluded from the containment penetration model provided that its design against external event hazards and internal hazards, i.e. missiles from initiating event, is adequate. However, this means that all the systems should be separately evaluated and possible threats to the integrity of closed loops have to be assessed e.g. pumps, heat exchangers etc. If it is evaluated that a pressure peak, high temperature, missile or any other cause might jeopardise the integrity of closed loop, it should be included in the reliability models of containment isolation. It may be relevant in that case to separate clearly the events “initial containment isolation failure” and “containment isolation failure during accident progression”;
- Conditional probability of leakage, or of failure to be isolated, is small taking into account the dependencies with support systems. Examples of penetrations that could be expected to have a low failure rate include the following:
  - Lines containing a blind flange;
  - Lines in which there are at least two automatic isolation valves plus an additional, normally closed valve.
- The magnitude of the radionuclide release is low. Each process system should be evaluated separately. Examples include:
  - Release through a line that will remain filled with enough water throughout the accident. If the reason for the low consequence is a phenomena such as pool scrubbing, a screening justification of the degree of filtration provided by the water should be performed before this route can be screened out;
  - Release through tortuous cracks or very small paths, e.g. instrumentation lines (inner diameter with less than 10 mm). Small lines can become plugged quickly and are generally not important potential release pathways. Again some screening justification should be provided. It can be noted that a leak path with a diameter 10 mm, direct to environment, would be considered as very significant if the design leak rate is around 0.3 % /day at design pressure.

Other containment isolation failure modes may be screened out if carefully justified. For example, if the position of isolation valves will be manually locked after successful containment isolation, the unintended cancellation of isolation by operator or by automation failures is not very probable.

The dependencies involved in containment isolation should be taken into account, i.e. power feed to motor operated valves, DC power and possible battery back-ups to actuators, automatic isolation signals, etc. The manual actions should also be precisely identified and quantified with HRA methods. The following should also be assessed: if the protection of systems against possible internal hazards is adequate and if the possible conditions during a severe accident will jeopardise the containment isolation, e.g. after hydrogen deflagration.

5.1.4 Initial containment performance and failure of isolation systems in APET/CET

After identification of penetrations that should be taken into account, the L2PSA probabilistic model can be constructed. The methods are similar to the system failure modelling used in L1PSA and models can be either included in the APET/CET or in the extended L1PSA event trees. Fault tree models of containment isolation should include possible failure modes of valves (fails to close, spurious opening and leak through) and dependencies (power feed, etc). In addition, manual recovery operator actions can be included if justified. Common cause failures should be included. The reliability data for the components should be gathered from real plant data when possible.

The important inputs from the reliability models to the L2PSA source term calculations are the size and location of the isolation failure (leak size) and information about possible issues affecting radionuclide retention in the leak route, e.g. if the leak path is through a water pool. Containment isolation will typically occur right after the initiating event or before the onset of core damage, so the timing of containment isolation, or failure of containment isolation, is not usually considered very important. Where containment isolation has not been successful this is normally assumed to occur from the beginning of an accident sequence. However, in some cases, containment isolation failure is also possible following initial successful containment isolation and, in this case, timing might be an important input for the source term calculation.

It should also be noted that the containment water inventory might be lost from the containment through an isolation failure path, e.g. through an open hatch or line. This would be the case if a reactor cavity door fails to close or isolation of penetrations in the lower compartment fails. Normally this is not possible during power operation but in shutdown states additional evaluations might be needed.

Normally, at least 2 sub-groups of isolation failure are determined, according to leak size: containment isolation failure and containment leak. Isolation failure means that containment isolation has failed and the penetration or hatch is open whereas containment leak means that valves or hatches have been successfully closed but maybe not totally, i.e. a leak through closed valves or seals may persist. If a simplified grouping is used, the leak sizes from the containment isolation failure should be conservatively chosen. When minimum cut sets are used and the grouping of different sizes of isolation failures is not made in detail, there is the possibility to lose information in cases where more than one of the penetrations failed to close. After cut set minimisation, it might seem to the user that only one small penetration is open, even though there might have been more than one open penetration before minimisation. This should be taken into account either in the modelling or when the leak size of the containment is chosen.

Below is an example of more detailed grouping:

Group 1: Very large isolation failures (containment practically open): doors and hatches, ventilation lines

→ open containment, containment will not pressurise, potential impact on the cooling water inventory.

Group 2: Large containment failure: penetrations which could cause a direct connection between the containment atmosphere and the environment

→ isolation failure, leak rate has an impact on containment pressurisation, potential impact on the cooling water inventory.

Group 3: Small containment failure: small lines or lines which will not cause a direct connection between the containment atmosphere and the environment, e.g. leak through water pool

→ isolation failure, leak rate has no effect on containment pressurisation.

Group 4: Containment leak through closed valves or seals

→ containment leak, leak rate has no effect on containment pressurisation.

There are several ways to do the grouping and modelling of containment isolation in an adequate manner. The level of detail in the modelling should be consistent with the other parts of the L2PSA study and it also depends on the containment design.

5.1.5 Examples and comments on specific plant designs

5.1.5.1 Initial leak and failure of isolation systems in Loviisa L2PSA

Containment isolation has been studied in Loviisa L2PSA. All the containment penetrations and different failure modes have been considered and the important ones have been included in the L2PSA modelling. The reliability models of containment isolation have been included in CET and leak sizes have been used as an input for source term calculations. This example will shortly describe the containment isolation failure modelling of Loviisa NPP L2PSA. All the details of the modelling have not been included here.

Isolation systems and failure mechanisms precluded from the modelling

Penetrations that are very small in size or/and through which the leak is very unlikely are precluded. This means that for example impulse piping and penetrations that are closed with flange (sealed and leak-tested after previous opening) are not included in reliability modelling. In addition, penetrations that are not to be isolated and penetrations that are already isolated at the time of initiating event are precluded.

After successful isolation operator will secure the isolation according the SAM guidelines and the possibility that operator would unintended cancel the isolation which has already succeeded is precluded. In case of total loss of electricity the automatic isolation would fail, but isolation can be manually achieved by using electricity feed from diesel generators dedicated for severe accident management.

Isolation systems and failure mechanisms included in the modelling

There are four different branches considering the containment isolation:

- Hatches or ventilation lines isolation failure (containment practically open);
- Isolation failure (large in size / atmospheric connection);
- Isolation failure (small in size / connection through water pool);
- Containment leakage.

If hatches (entrance hatches, material hatch, cavity door) or ventilation lines of the controlled zone (both lines used either during power operation or during shutdown) are left open, the containment is practically open and does not provide any mitigation of releases. Containment is open from the beginning of an accident. This branch in the CET is important mainly in shutdown study, since during shutdown the containment might be open, i.e. material hatch or cavity door is open, at the time of initiating event and leaktightness of containment has to be recovered.

Large isolation failures are failures of process systems which might lead to atmosphere connection between the containment and a room or non leak-tight tank or system outside of the containment. Some isolation failures might need additional failures, or opening of the safety valve in the systems. In some cases failure of the isolation system is only considered possible in sequences where missiles inside the containment are considered possible. For example, isolation failure of the valves of the normally open fuel pool overflow line leads to large isolation failure.

Small isolation failure is created in cases where the isolation of the system containing fluid fails. In this case, isolation failure is in many cases only possible due to other failures or missiles. For example, leak through pumps of the special drainage system is considered possible in cases of missiles.

Containment leakage is created if both isolation valves in the line are not properly closed.

Reliability data used for components is based on actual plant data. Failure modes considered for the valves are 'fails to close', 'repair of incipient failure' and 'leak through'. Electricity feeds and battery back-ups of electricity are included as well as manual operator actions from local control centres, using dedicated severe accident electricity feed in case of total loss of electricity. In addition, faults of automation signals have been included as well as manual operator actions during shutdown states, for recovery of containment leaktightness. Manual repair actions on the valves during power operations are not considered possible.

Leak sizes, used in source term calculations for the different classes, have been chosen according to the actual size of the process line dominating the results. The size of containment leakage has been chosen according to periodic tests of containment isolation valves and leak sizes found in the test.

The possibility to lose water outside of the containment from the steam generator space has been taken into account. This is critical in the case of Loviisa, because in-vessel retention of corium is a cornerstone of the

Loviisa SAM strategy. Water can be lost, because of isolation failures, only during shutdown states when penetrations that are normally closed and sealed are used for cabling.

Pre-existing containment leakage and leakage rate testing program

In all the L2PSA cases the pre-existing leakage is taken into account, which means that the pre-existing leakage is also assumed in success scenarios. The pre-existing leak rate of the Loviisa ice condenser containment (steel liner) is 0.2 % in a day (24 h) at the design pressure (1,7 bar^{abs}). In source term calculations the pre-existing leakage has been divided into 5 different release routes.

The integrated leakage rate test for Loviisa NPP is done every 4 years. Local leakage rate tests are done on a regular basis for each component, according to their testing program. The leakage rate testing for hatches is done every 6 months. The leakage rate testing for the most important valves is done on a yearly basis, either during power operation or during shutdown.

5.1.5.2 Efficiency of condensation pools

Several plants – in particular BWRs, but also some VVERs – contain water pools in order to condensate steam from a reactor coolant leak and thus to reduce the containment pressure. In addition, the scrubbing in the water pool catches a large fraction of aerosols, reducing radioactive effluent from the containment. The following failure modes can be imagined for those pools:

Loss of condensation pool water: This may occur if a leak exists in a system (e. g. in an ECCS) which is connected to the pool and which cannot be isolated. Then the function of the pool is lost, together with a containment leak through the failed system. The consequences of such failure are significant because it will initiate a core damage and a containment failure at the same time. This type of failure typically is an initiating event for the whole sequence, so that its quantification has to be done in L1PSA. It is in principle also conceivable that the leak develops during the accident sequence after core damage onset.

Loss of condensation efficiency due to high temperature: normally, the condensation pool temperature will be kept at a certain design limit below boiling. However depending on the accident sequence (e.g. recirculation of condensation pool water with loss of heat sink) the temperature could increase and even reach boiling. In this case steam condensation as well as scrubbing aerosols will be less efficient. A typical sequence would be a transient in a BWR with heat removal by circulation through the condensation pool. If the heat sink fails, the circulation continues with rising temperature until the pumps fail due to cavitation. The core damage occurs, and the effluent from the reactor coolant system is conducted into the inefficient condensation pool.

Bypass of the condensation pool: If steam from the containment could penetrate into the gas space of the condensation pool bypassing the pool, the pool looses its function. Such a leak could e.g. occur in a Mark I BWR if condensation tubes develop a leak at their top. Depending on the size of the bypass, the containment pressure could exceed the failure threshold, leading to early large containment failure.

pH Control: If during the course of the accident the pH drops into the neutral or acidic range a hight fraction of iodine will be converted into volatile species (elemental iodine and organic iodide) in the sump or suppression pool water, increasing the volatile species contribution into the containment atmosphere after the scrubbing phase or by late evaporization of the suppression pool. In some NPP designs, pH control is practised either by storing chemicals in the containment at a place where they are inherently flooded and dissolve in the sump water or by pumping them into the containment during an accident using the containment spray system. In others NPP designs, there is not engineered system for controlling the pH. In this case, the pH will need to be calculated in order to evaluate the iodine source term [235]. This item should be taking into account into the computer codes.

5.1.6 References

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5.2 QUASI-STATIC LOADING / DYNAMIC LOADING – STRUCTURAL RESPONSE, STRUCTURAL ANALYSIS, FRAGILITY CURVE

5.2.1 Introduction

The timing and the way in which the containment fails is a crucial factor influencing the magnitude of the off site consequences. Containment design criteria are usually based on a set of deterministically selected load scenarios. For example, pressure and temperature challenges are usually based on the design basis LOCA. External events such as earthquakes, floods, high winds or aircraft crash are considered as well in some cases. Assessment of beyond design accident sequences show that, in some cases, significant containment loading can occur, reaching or exceeding the design loads. None of the design basis accident scenarios involve rapidly increasing containment loads. Therefore, transient loads like fluid jet impingement, direct containment heating, rapid deflagration or detonation of hydrogen pockets which may occur during severe core degradation accidents, may pose significant threats to containment integrity.

Structural analysis of the containment leads to a probabilistic structural response function for gradual pressure rises (quasi-static loads) and also for rapid pressure rises (dynamic loads). This structural response plays an important role in the evaluation of physical phenomena in the context of L2PSA. The goal of the structural analysis is to create a probability of containment failure as a function of pressure under different conditions, this is also known as a fragility curve. In a L2PSA, often more than one fragility curve is relevant.

According to IAEA Safety Series [237] two basic approaches have been used in L2PSA studies to characterise the loss of containment integrity due to overpressurisation: the 'threshold model' and the 'leak before break model'. The threshold model defines a threshold pressure, with some associated uncertainties, at which the containment is expected to fail resulting in a large rupture with the potential for rapid blowdown of the containment atmosphere. In the leak before break model, pertinent to liner tear and penetration failure, enhanced containment leakage is expected to precede a large rupture. In general, enhanced leakage begins at pressures below the threshold pressure and progressively increases up to that pressure, at which point a large rupture is expected to occur. This is not necessarily the case as it is also possible that enhanced leakage will become so large that the containment pressure rise will stop before the rupture threshold is reached. Pressure in the containment will either remain high or may even start to decrease, depending on the enhanced leak rate. Whatever the approach, all such accident scenarios will result in environmental releases if there is radioactivity in the containment atmosphere.

5.2.2 Dynamic loads versus quasi-static loads

Quasi-static pressure loads would results from the protracted generation of steam and non-condensable gases through MCCI, the interaction of molten core material with the concrete floor beneath the RPV. 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 ESFs. 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.

Some of the phenomena associated with severe accidents, modelled in L2PSA, are characterised by very short time scales, typically a few seconds, e.g. hydrogen combustion and direct containment heating. The loads from such energetic events are characterised by short term high pressure spikes. As dynamic loads are difficult to translate into experiments for a specific containment, the dynamic loads should be transferred to an equivalent quasi-static load on the containment.

The equivalence value between dynamic and quasi-static loads changes with the natural frequencies of vibration of the containment. Non-linear 3D containment analysis techniques can be used, where the load is decomposed using the Fourier series (harmonics) to obtain the first modes (axial symmetry, lateral displacement and ovalising) and the eigen-frequency of the containment structure. See Fig. 53.

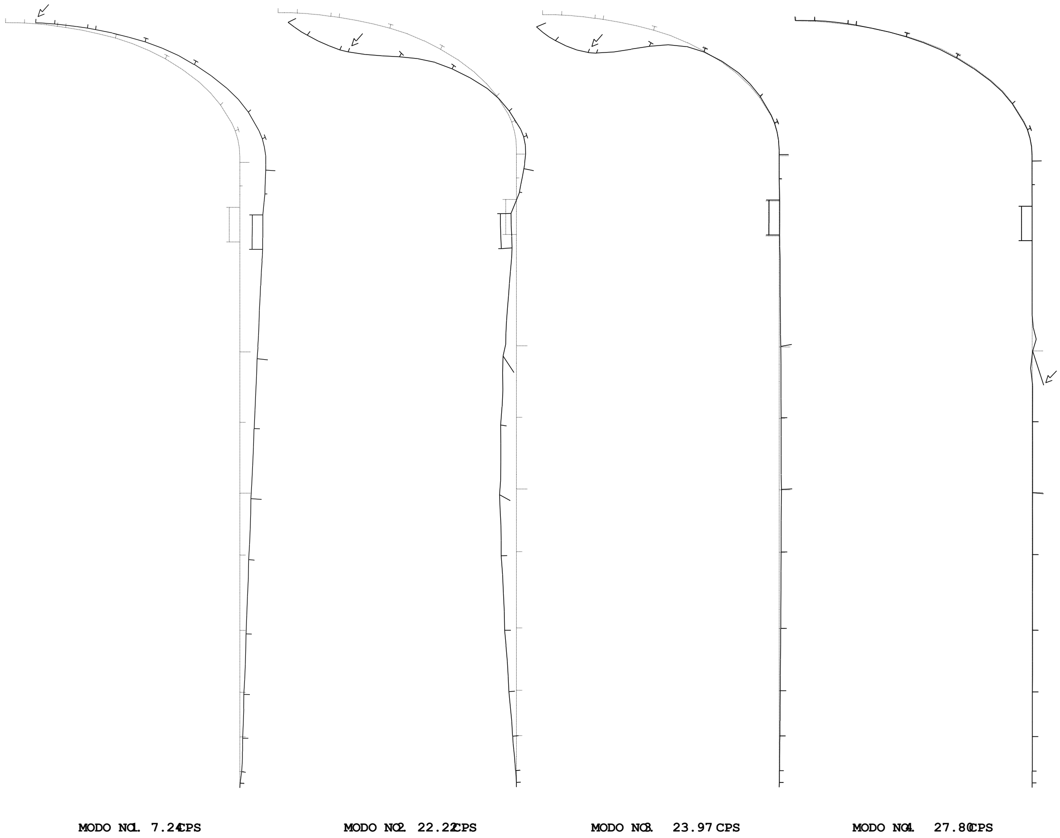

Fig. 53 Graphics of the first four modes of vibration of the steel metallic for the second Fourier term

As an example, in section 2.7.2 in OECD SOAR [239] it is stated that for a concrete containment with a low eigen-frequency (in the range of 5 to 25 Hz) the equivalent pressure imposed by a hydrogen combustion on the containment lies in the order of 2 times (about 1.8) the AICC pressure, irrespective of whether the combustion mode is a fast deflagration, DDT or a detonation. Other studies, for example Boyack et al. [240], show that in specific situations, the equivalent loads can be up to an order of magnitude greater than the AICC pressure.
It is recommended that dynamic loads are taken into account by assuming a load distribution based on the quasi-static loads and then apply a dynamic load factor to this result.

5.2.3 *Determination of the fragility curve*

To obtain a fragility curve, different failure modes of the containment need to be evaluated. This includes but may not be limited to:

- Penetrations;
- Expansion joints;
- Air locks;
- Seals;
- Changes in stiffness;
- Containment expansion hindrances;

- Global failure due to inner or outer pressure.

Each failure mode is then classified as either a break or leak condition of the containment. A break in the containment may be treated with a sufficiently large break area that deposition of fission products in the containment can be neglected, but a leak condition has a sufficiently small area that significant deposition of fission products in the containment is possible. The value of the border between a break and a leak of the containment may vary depending on the size of the containment but, as a working value, 0.1 m² is suggested.

An important factor in determining the containment failure pressure is the temperature of the structures. In early phenomena during the in-vessel phase, e.g. hydrogen combustion, the assumption that the containment structures do not have a high temperature may be valid, whereas for late phenomena this is generally to be checked. Hot gases emanating from MCCI or hot plumes from hydrogen recombiners may locally increase temperatures.

At higher structural temperatures, the elastic limit of the material will decrease and the failure pressure needs to be multiplied with a reduction factor to account for higher temperatures (see Fig. 54).

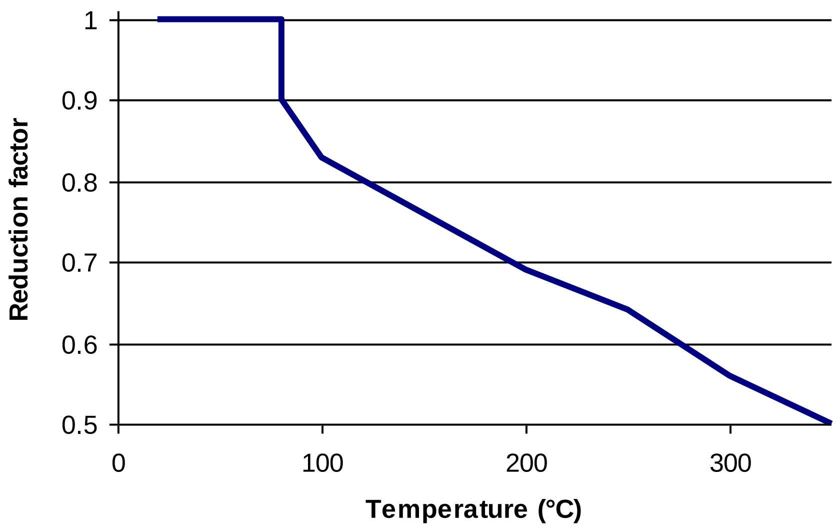

Fig. 54 Temperature dependent reduction factor to apply to the elastic limit (KTA 3205.1 [241])

Several well established code systems are available for quantification of containment load capacity, for example NASTRAN, ABAQUS, DYNA3D, NEPTUNE, CAST3M. Usually, containment failure is described by a cumulative probability function.

Typical values for PWR containments, with an inner steel liner, begin to rise from practically zero at about 5 bar until the probability of failure approaches 1 in the range 13-14 bar. Typical values for BWR containments begin to rise from practically zero at about 4-5 bar (7-8 bar for the Mark-I containment) and the probability of failure approaches 1 in the range 7-8 bar (18-20 for Mark-I containment) [242]. There is an example showing failure modes and the estimated failure pressure in Table 37 and Fig. 55.

Table 37 Identification of the containment local failure modes for BWR Mark-III (values from Cofrentes NPP).

Failure mode	Estimated failure pressure Mean value (MPa/psig)	Supporting reference
Air lock inflatable seal leakage	0.60 / 87.02	NUREG/CR-5394
Expansion bellows failure of mechanical penetration	0.60 / 87.02 (worst case)	NUREG/CR-6154
Blade failure of mechanical penetration	0.60 / 87.02 (worst case)	Similar to Perry NPP
Blade failure by contact with thickening in building shielding	➢ 0.70 / 101.53	Global failure
Cylindrical anchorage blade	➢ 0.80 / 116.03	Similar to River Bend NPP
“O-rings” leakages	➢ 0.90 /130.53	Global failure
Air locks structural failure	➢ 1.00 / 145.04	NUREG/CR-5118
Equipments door structural failure	➢ 1.00 / 145.04	Internal analysis
Leakages in electrical penetrations	➢ 1.00 / 145.04	NUREG/CR-5334

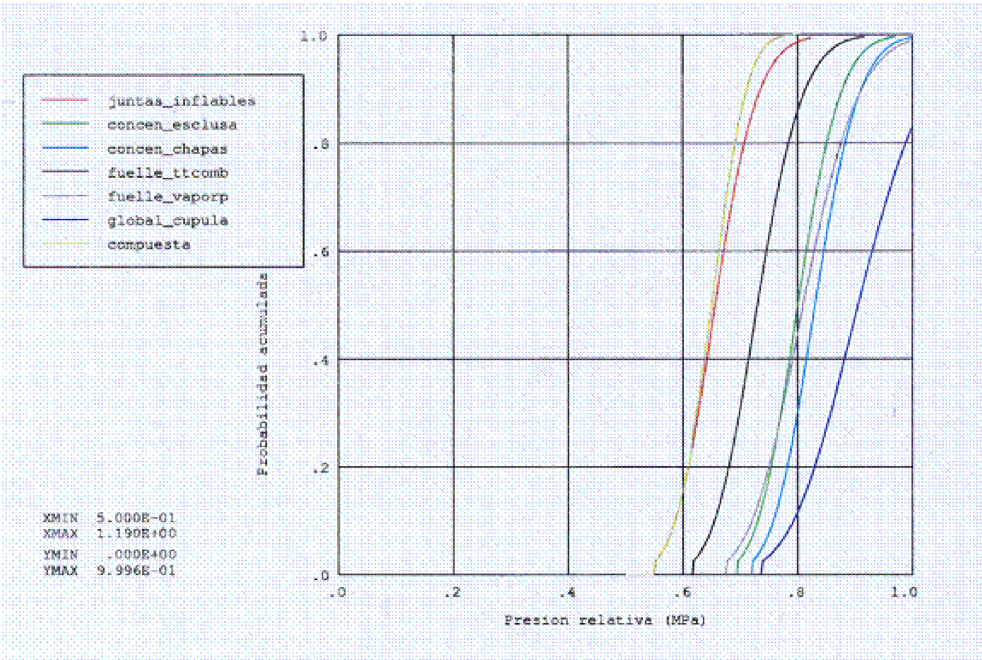

Fig. 55 Fragility curve of the local containment failure mode for Cofrentes NPP

5.2.4 Determining branch probabilities based on the fragility curve

For each phenomenon, the loads to the containment shall be calculated and this load distribution shall be placed on the fragility curve to obtain a containment failure probability for the phenomenon, as is illustrated by Fig. 55. For each pressure, a fragility curve provides a conditional containment failure probability for the corresponding containment failure mode.

For phenomena that happen over a very short time and successively, e.g. hydrogen combustion then DCH, the probability of a leak and a break can be calculated successively in the APET. The impact of a second peak may not be equivalent to a first one that may have already damaged the containment. Sensitivity studies on APET quantification may help to develop an understanding if this is an important issue or not. However, special care needs to be taken for a slow pressure increase, as in the case for containment overpressurisation e.g. the containment pressurises due to failure of heat removal. In this case, the fragility curves for break and leak of the containment will generally overlap, due to the uncertainties in the determination of the failure pressure. In this case, the probability of leak before break needs to be determined. For this it is necessary to distinguish between two kinds of uncertainties in the containment failure pressure:

- Variability of the material properties (aleatoric uncertainty);
- Uncertainty in the determination of the failure pressure (epistemic uncertainty).

Generally, the variability of the material is uncorrelated between different failure modes, whereas the epistemic error is correlated between different failure modes. This partial correlation determines the probability that a containment leak condition occurs which halts the pressure increase, thus avoiding a containment break condition.

5.2.5 Integration of the L2PSA APET

For an event that induces a pressure peak, the more appropriate way of calculating the conditional probability of containment failure would be to calculate the convolution of the density probability distribution of pressure for the event with the density distribution calculated for each containment failure mode. A distribution law of containment failure can easily be obtained by Monte-Carlo runs. The Fig. 56 illustrates this method. In the case where the “risk zone” corresponds to rare events (tails of distribution), it is more appropriate to use the integrated density probability function for the containment failure. That explains why the use of fragility curves is generally used in L2PSA, especially when the convolution is directly performed by quantification of the APET.

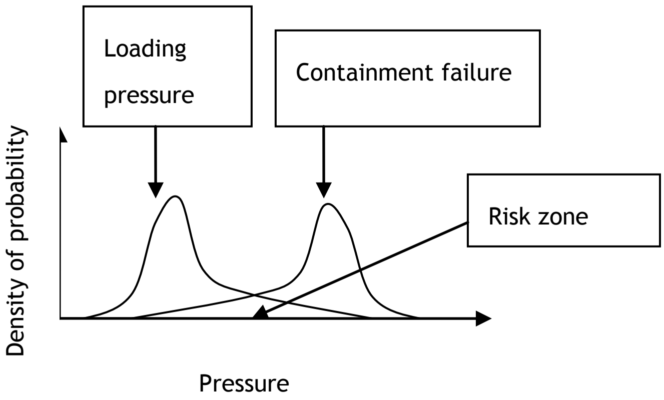

Fig. 56 Convolution of pressure loading / containment failure density probability functions

The APET must be able to calculate precisely the containment failure probability in case of a pressure peak, either slow or fast. Two methods, with variants, are available:

- If the APET mainly includes probabilistic nodes, i.e. APET built with a L1PSA probabilistic, then the containment failure probability has to be calculated outside the APET;
- If the APET includes users-functions, i.e. APET with probabilistic tools like EVNTRE or KANT able to calculate explicitly the pressure loading for each event in each accident sequence, then the containment failure probability can be calculated for each event. The final result may be more precise because each conditional probability is related to one sequence and takes into account dependencies with previous events.

5.2.6 References

[239] Breitung W. et al.:Flame Acceleration and Deflagration to Detonation Transition in Nuclear Safety, SOAR, NEA/CSNI/R(2000)7, (2000)

[240] Boyack K.W. et al.: Loads from the Detonation of Hydrogen-Air-Steam Mixtures, Sandia Report SAND92-0541 UC-741 (1992)

[241] Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen für Primärkreiskomponenten in Leichtwasserreaktoren, KTA 3205.1 (2002)

[242] CNRA (WGIP) “L2PSA methodology and severe accident management”, OECD/GD(97)198

5.3 CONTAINMENT BYPASS

5.3.1 Introduction

In case of severe accidents, the containment is a significant barrier for preventing radionuclides reaching the environment. Several potential failure modes for the containment have been addressed within this chapter, resulting in leak or break failure events. Containment bypass events have to be distinguished from containment failure events as follows:

- <u>Containment failure</u> occurs if the containment is not, or is no longer, able to retain radionuclides according to its specification. Typical containment failure modes are the failure to close the containment ventilation ducts or a structural failure of the containment due to overpressure;
- <u>Containment bypass</u> occurs if there is a release path for radionuclides from the core into the environment without passing through, i.e. bypassing, the containment. Typical containment bypass sequences are steam generator tube leaks with releases through secondary relief valves (in PWRs) or leaks in the reactor coolant loop outside the containment with failure to isolate the reactor coolant loop (in BWRs).

There are plant designs which distinguish between “primary” and “secondary” containments. A typical example is a steel containment which is housed inside a concrete building. An intermediate controlled volume between these two structures exists. Where appropriate, a distinction may be made between primary and secondary containment.

Emergency core cooling systems may be connected to the containment sump as well as to a reactor coolant loop, partially passing through rooms outside of the containment. If such a system develops a non-isolated leak outside of the containment boundary it is considered a containment bypass, i.e. from the reactor coolant loop to the rooms outside of the containment, and a containment failure, i.e. from the containment sump to the rooms outside of the containment.

There may be accident sequences, e.g. RPV melt-through late in a containment bypass scenario, where a containment bypass and a containment failure are at stake. However, these two issues can be addressed separately. The present section exclusively deals with the containment bypass issue.

5.3.2 Issues to be addressed in L2PSA

The following issues have to be addressed with regard to containment bypass:

a) Which systems are potential candidates for generating a containment bypass event?

b) What are the consequences of containment bypass?

c) What is the probability that containment bypass occurs?

d) What is the quantity of radionuclide release (source term) through a containment bypass? For this issue it is important to distinguish whether the bypass flow is directed immediately into the environment or whether it is directed into an intermediate room inside the plant.

5.3.3 Potential containment bypass routes

The identification of potential containment bypass paths is a standard task for PSA. These paths should be identified by a rigorous search for systems which lead from the reactor coolant loop to locations outside of the primary and / or secondary containment. Classical bypass scenarios are as follows:

a) A leak in a steam generator tube or collector cover for WWER type develops either as an initiating event or as an induced event due to accidental conditions. Radioactivity from the degrading core passes through the leak path into the secondary loop. There is a path to the environment via one of the relief valves of the secondary loops (for PWR only). In this case the bypass flow is directed immediately into the environment, bypassing both primary and any secondary containment;

b) A leak in the reactor coolant loop outside the containment develops either as an initiating event or as an induced event due to accident conditions. The isolation valves in the affected reactor coolant loop fail in an open position (for BWR only). In this case the bypass flow is directed into the reactor building or the turbine hall. Depending on their design, these structures may provide another significant barrier to radionuclide release to the environment;

c) A leak in the volume control system develops outside of the containment. Isolation valves in the affected loop fail in an open position. This type of accident is sometimes called a “V-sequence” or “interfacing LOCA”. Depending on the plant design, the bypass flow is directed into rooms which may provide another significant barrier to radionuclide release to the environment;

d) A leak in an emergency core cooling line develops outside of the containment. Isolation valves between the reactor coolant loop and the leak location fail in an open position. Since emergency core cooling systems are generally located in well protected rooms inside the plant, the radionuclide release into the environment will probably be mitigated by some remaining barriersl;

e) A leak in a heat exchanger of an emergency core cooling system develops and there is a flow path from the secondary side of the heat exchanger to the environment. Depending on the plant design, the bypass flow is directed into systems or rooms which may provide another significant barrier to radionuclide release to the environment.

In addition to these potentially large routes there may be a number of smaller containment bypass routes, e.g. through instrumentation lines.

5.3.4 Consequences of containment bypass on accident progression

Containment bypass leads to transfer of primary coolant and emergency core coolant outside the containment boundary. The influences of these coolant inventory losses should be taken into account in the L2PSA. Particularly, availability of coolant in the containment might be critical for plant designs relying on in-vessel retention, i.e. the initial primary circuit inventory or emergency core coolant is needed for the cavity flooding.

Bypass might also have other influences on the L2PSA. For example, if recovery actions are modelled in the L2PSA, the effects of radioactivity migration along the bypass path should be assessed, e.g. contamination in the room where leakage occurs and any external radiation fields, from pipes on the bypass path, on accident recovery actions. Also, the effects on the systems used for SAM which are credited in the L2PSA should be assessed. For example, even a very small bypass leakage through piping to a room outside of the containment boundary might cause the room temperature to rise above the design criteria of equipment located in the room.

5.3.5 Probability for opening containment bypass routes

An early bypass, before core damage, may occur due to a combination of an initiating event and related system failures, e.g. steam generator tube leak and the failure to prevent the automatic opening of the secondary relief valves to the environment. Since potential containment bypass routes are protected by isolation valves, there is generally a chance for the operators to recover such failures. The probability of an early occurrence of a containment bypass will be determined by the L1PSA and will not be addressed here. However, it must be stressed that the L2PSA team is responsible for thoroughly reviewing the L1PSA work in this area. Within a L2PSA it has to be assessed whether containment bypass paths will be created due to accident phenomena. In particular the induced steam generator tube leak has to be considered in PWRs as follows:

During the core melt process high temperature gas will flow through the hot leg of the reactor coolant loop. Within a L2PSA these temperature histories will be determined by application of integral codes. However care should be applied when interpreting the results because the models may not adequately represent the complicated counter-current flow which may develop in the hot leg of the RCS under such circumstances. When the steam generator tube temperature increases, there is an increasing probability for a tube leak. The analysis of the associated probability has to consider:

- What is the pressure inside the primary system and the secondary system? To be taken into account: depressurisation of the primary system due to accident management or due to failure of a safety valve in an open position, pressure surges due to relocation of core melt into the lower plenum, pressure history in the secondary loop;
- Will the hot leg or the surge line fail due to high temperature before a steam generator tube fails? To be determined by appropriate thermodynamic and structural mechanics approach;
- What is the condition of the steam generator tubes before the accident?
  Statistics of pre-existing tube deficiencies may be applied.

In the German guidelines for L2PSA [245] (section 7.2.3.2), there are data for temperature and pressure induced failure of steam generator tubes. They take into account the time of exposure to the high temperature and pre-existing reduced wall thickness.

5.3.6 *Quantity of radionuclide release through containment bypass*

Traditionally L2PSA assume very high releases to the environment for bypass sequences because the mitigating features of the containment are bypassed. Most L2PSA for PWRs identify steam generator tube leaks as high ranking risk contributors [243]. This document references several L2PSA studies where up to 90% of the highest release categories are due to bypass sequences.

However, there are some mitigative phenomena which should be taken into account if the L2PSA aims at producing realistic source terms. Therefore, it is recommended to carefully assess the source terms. In general, source terms for a L2PSA will be determined by applying integral codes or fast-running codes. The analyst should be aware of the uncertainty of such results, in particular for radioactivity transport through consecutive rooms or complicated flow paths. The models should analyse these paths as accurately as is feasible.

One of the most obvious mitigative phenomena is pool scrubbing. If the leak in a heat exchanger is covered with water to a sufficient depth, the quantity of radionuclides released is significantly reduced [244]. This reference suggests a decontamination factor of 2-5 for “shallow” water pools and a factor of 10-20 for “deep” water pools. Estimation of the flow path retention factor in such scenarios is typically dependent on the depth of water covering the break which may be an uncertain parameter, i.e. is the break at the bottom or the top of the tube bundle, the length of time for which this water pool can be maintained and the state of any pressure relief valves in the steam side. These conditions are especially significant if SAM measures consider the possibility of flooding the secondary side.

Radionuclide releases which are not directed immediately to the environment but into any secondary containment, or even rooms outside the containment boundary, may also benefit from additional natural and engineered mitigation mechanisms. Depending on the flow rates, building volumes, ventilation systems and atmospheric conditions, a significant quantity of radionuclides may be retained in the buildings due to aerosol deposition and / or filtration, reducing the releases into the environment.

The PDS definitions, for early bypass pathways and/or the L2PSA accident progression analyses for induced bypass paths, usually contain some information describing the bypass path and potential flow conditions. If this path is well defined, it may be possible to reasonably estimate the parameters, e.g. pipe geometry and thermohydraulic flow conditions, which are needed as input information for estimation of the flow path retention factor. Hence, it may be theoretically possible to give credit to retention mechanisms when modelling a bypass sequence in the L2PSA even if this is not routinely done. It should be noted that if the conditions are favourable to retention of fission products, even quite a short release path may be sufficient to ensure very efficient deposition.

However, when the release path is less well defined there is the potential to have the release almost anywhere in a building or suite of rooms e.g. from any location in a long pipe run that passes through a number of rooms, it is quite difficult to estimate the global flow path retention factor with any confidence and the treatment of uncertainties is quite difficult. Giving credit to such mitigative mechanisms in a real L2PSA application for bypass sequences may, therefore, not be straightforward. The ability of an auxiliary building to retain radioactivity discharged from an interfacing system is likely to be dependent on:

- The scenario under consideration;
- The location of the discharge point in relation to any water pools, spray systems and HVAC filtration, and the availability of diverse flow paths to the environment;
- External conditions, especially the wind, which influence the leakage rate of a building, especially if the ventilation systems are stopped.

The amount of effort invested in the calculation of flow path retention factors should take into consideration:

- The likelihood that the necessary parameters can be calculated with confidence;
- The uncertainty in / number of potential flow paths with different conditions;
- The importance of the sequence to the global risk profile.

5.3.7 References

[243] OECD/GD(97)198 “L2PSA methodology and severe accident management”

[244] Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Specific Safety Guide N° SSG-4.

[245] Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Stand August 2005, BfS-SCHR-38/05, ISSN 0937-4469, ISBN 3-86509-415-5

6 SYSTEM BEHAVIOUR IN SEVERE ACCIDENT CONDITIONS

6.1 INTRODUCTION

This section discusses the use of systems in severe accidents. There are two different approaches used for severe accident management for operating power plants: use of non-dedicated systems and use of dedicated systems. Those approaches are separately described in sections 6.2.2 and 6.2.3. Partner examples for both approaches are provided in sections 6.3.1 and 6.3.2.

Finally, specific systems such as the core catcher, the filtered venting system and some passive systems are described in section 6.2.4.

6.2 SYSTEMS USED IN SEVERE ACCIDENTS

6.2.1 Severe accident safety functions

Severe accident management should be structured around well defined safety functions and systems which are used to fulfil these safety functions. Systems used in severe accidents and considered in Severe Accident Management Guidelines (SAMGs) should be taken into account in L2PSA if the PSA is to claim credit for accident management.

For example in WOG SAMG for PWRs, there are systems that can be used for well-defined main functions:

- Mitigation / reduction of Fission Product (FP) releases;
- Injection into the Steam Generators (SGs);
- Reactor Coolant System (RCS) depressurisation;
- Injection into the RCS;
- Injection into containment and reactor cavity flooding if there is a path between containment sumps and reactor cavity;
- Control of containment conditions / containment depressurisation;
- Reduction of containment hydrogen / control of hydrogen flammability.

Alternative safety functions or main functions could also be chosen, if justified, and other approaches (than WOG) might use different phrasing.

6.2.2 Approach with non-dedicated systems

Generation II NPPs were not designed to cope with severe accidents. Systems were initially qualified for use in design basis accidents and particularly for use in conditions that can prevail after a Large Break Loss Of Coolant Accident (LBLOCA) (temperature, humidity and pressure conditions for short term and integrated dose over one year).

The systems used in severe accidents are the systems that are referred to in the SAMGs. In general, the philosophy followed in LWR SAMG is to maximise the use of existing equipment. Plant-specific, non-standard, use of equipment is considered in addition to standard use of equipment. The possibility to use temporary connections is usually reviewed. For each item of equipment, the best-estimate limitations for its use are identified and introduced in the guidance.

This approach is said to be workable [253] because:

- The instrumentation is environmentally qualified for design basis accidents conservatively and, therefore, exhibits necessary capabilities to remain operational in severe accident conditions, especially given the reduced requirement for accuracy;
- The identification of redundancies and alternative means to obtain information on key parameters (see Table 38) can increase the confidence in the capabilities of existing instrumentation in severe accident conditions.

Therefore in the APET, if the PSA is to claim credit for accident management, the failure probability of each system has to be evaluated and no specific assumptions are made on the possibility that a system could not be qualified in SA conditions.

6.2.2.1 Use of systems

The information concerning the availability of the systems is provided by the outcomes of specific attributes, defined during the interface between Level 1 and L2PSA, for example the status of the Safety Injection (SI) systems. The outcomes allow differentiation of the following status for the systems:

- Available for operation, with a distinction for the systems that can be used in direct injection and recirculation modes: they can be available for both modes or only for one of them (the second mode is failed);
- Failed (not recoverable) in all circumstances;
- Available as soon as related AC power supply (first or second level) is recovered i.e. the only reason for non-availability of the system is the loss of AC power;
- Available following active recovery, if this is possible - see below.

The availability for operation of a system assumes the availability of its support systems.

If active recovery operations are considered, i.e. manual repair of components, the conditions in the plant should be taken into account. For example, some rooms may not be accessible because of radiation conditions. If these evaluations are not made, or if the manual actions are considered not to be successful, the only possibility that can be considered for recovery of the system in the APET is the recovery of the related AC power.

At the beginning of the SAMG, the control switches of the non-operating equipment are locked to avoid the automatic start of any equipment. If a system is available for operation, and the conditions for its operation are fulfilled (for example, RCS pressure for LHSI pumps and RWST content for RCS injection in direct mode), it can be started based on the application of the SAMG. The probability for its start to be successful depends on the HRA performed for the implementation of the SAMG strategy. If several systems for the same function are available, and the conditions for their operation are fulfilled, the most usual and efficient system for the function is first considered. For the case when HRA says the start of the most usual and efficient system is not performed, due to human error at a given moment, it is assumed that the start of some alternate and atypical systems is also unsuccessful at that moment.

In the APET, one point that has to be taken into account is the efficiency of the system operation. For example, the success of RCS depressurisation has to be assessed:

- For the opening of the pressurizer PORVs to achieve a low RCS pressure;
- For the SG depressurisation to effectively depressurise the RCS leading to efficient heat transfer from primary to secondary.

Due to a specific plant condition or thermal hydraulic phenomena, it is possible that the objective expected to be achieved by using a system is not attained. This has to be dealt with in combination with HRA quantification. One of the issues of the HRA quantification is the grace period i.e. the time available to perform the action successfully. The efficiency of the system operation has to be taken into account in the same issue.

If equipment and systems used to cope with design basis conditions are supplemented by additional equipment to mitigate severe accidents, the latter equipment should preferably be independent. For dedicated or upgraded equipment, there should be sufficient confidence in the equipment and, where possible, demonstration of its capability to perform the required actions in beyond design basis and severe accident conditions should be provided. Demonstration of the capability of equipment should be provided where other assessment methods cannot provide sufficient confidence. However, the level of qualification applied to such equipment need not necessarily be the same as that typically required for components and systems that cope with design basis conditions. Similarly, requirements on the redundancy of such systems may also be relaxed compared to the requirements applied in the design basis domain.

The assumptions defined in the APET, i.e. the conditional probability of failure in severe accident conditions, must take into account the depth of the demonstration of the ability of systems in such conditions.

The following is a list of systems and strategies which may be used during SAM, grouped by safety objectives [254]:

Maintain Coolant Inventory:

- Refill RWST with borated water or CST with condensate;
- Reduce containment spray 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 utilize 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 mobile battery charger to recharge batteries;
- Enable emergency cross-tie of AC power between two units or to onsite gas turbine generator.

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 MCCI;
- 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;
- Use recombiners or igniters to control combustible gases;
- Vent containment to control combustible gases (pre-vessel failure and/or post-vessel failure).

Prevent Basemat Melt through:

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

Mitigate Radionuclide Release:

- Control Transport Out of RCS:
  - Use auxiliary pressurizer 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 side (SGTR);
  - Flood break location/interfacing system (LOCA).

6.2.2.2 <u>Qualification of systems</u>

When systems that are already implemented at the plant, i.e. non-dedicated systems, are used for severe accident management, the systems are typically not qualified against severe accident environmental conditions. The formal qualification of systems is not necessarily done but the usability of systems in the prevailing conditions during a severe accident is separately evaluated and some actions to support the usability of systems might be taken. Systems might be missile protected or radiation shielding might be installed. Mechanical components of systems (pumps, valves, etc.) can normally withstand conditions in severe accident reasonably well (attention can be paid to elastomeric behaviour in radiation conditions). Particular care must be put into the evaluation of usability of instrumentation and control systems. Below are some requirements for instrumentation and control that should be taken into account [255]:

- Since the SAMGs depend on the ability to estimate the magnitude of several key plant parameters, the plant parameters needed for both preventive accident management measures and mitigative accident management measures should be identified. It should be checked that all these parameters are available from the instrumentation in the plant. Where instruments can give information on the accident progression in a non-dedicated way, such possibilities should be investigated and included in the guidance. For example, ex-core neutron detector readings are influenced by the location of core debris in the vessel and the amount of remaining water, so these readings could be used to acquire information about the evolution of the accident;
- The existing qualification for relevant instruments should be taken into account, and it should be recognised that such equipment may continue to operate well beyond its qualified range. Alternative instrumentation should be identified where the primary instrumentation is not available or not reliable. Where such instrumentation is not available, alternative means should be developed, e.g. computational aids;
- Use of instrumentation that is qualified for the expected environmental conditions is the preferred method to obtain the necessary information;
- The effect of environmental conditions on the instrument reading should be estimated and included in the guidance. It should be taken into consideration that a local environmental condition can deviate from global conditions and, hence, instrumentation that is qualified under global conditions may not function properly under local conditions. HPME, for example, will spread debris all around the containment and while global conditions may remain within qualification envelopes, the local environment may be quite challenging e.g. radiation due to locally deposited fission products, excessive heating due to decay heat from deposited fission products. The expected failure mode and resultant instrument indication (off-scale high, off-scale low, floating) in severe accident conditions should be identified;
- Severe accidents may present challenges to instrumentation where such instruments operate outside their design range. As the indication from instruments may be unavailable, due to harsh conditions, all indications used to diagnose plant conditions for severe accident management should be benchmarked against other direct or derived indications to reduce the risks associated with faulty readings. In practice, every key instrumentation reading from a non-qualified dedicated instrument that is used for diagnosis or verification should have an alternate method to verify that the primary reading is reasonable. When an alternative means of obtaining a key parameter value cannot be identified,

consideration should be given to upgrading or replacing the instruments to provide that alternative indication;

- In the development of the SAMGs, the potential failure of important non qualified instrumentation during the evolution of the accident should be included and, where possible, alternative strategies that do not use this instrumentation should be developed. The ability to infer important plant parameters from local instrumentation or from unconventional means should also be considered. For example, the steam generator level can be inferred from local pressure measurements on the steam line and steam generator blow down lines;
- The need for development of computational aids to obtain information, where parameters are missing or their measurements are unreliable, should be identified and appropriate computational aids should be developed accordingly.

In PWR, severe accidents can be successfully managed by monitoring the status of only a few key plant parameters (Table 38, [254]). The probability that these parameters are correctly interpreted by the operators is linked to the quantification of the human reliability (see section 3). In addition to the main method of measurement for those key parameters, some alternate methods of measurement and/or computational aids should be provided. All possible methods of measurement have to be identified during plant-specific implementation. They should be assessed to determine their availability and capabilities during severe accident conditions. Some methods are provided to verify instrumentation accuracy. Information concerning instrumentation is documented.

Table 38 Definition of Plant parameters

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

* = required for AM-measures

6.2.3 Approach with dedicated systems

Severe accident management can also be totally structured around dedicated systems, which are designed to be used in severe accidents for mitigation of the consequences of the accident. These systems are designed to operate in severe accident conditions and normally they are completely independent from the systems that are used during normal operation or design basis accidents. Also, auxiliary systems such as power feed and instrumentation are dedicated for severe accidents. This kind of approach for SAM has been taken for GEN III+ plants, but some operating plants have gone through extensive modifications for SAM purposes and dedicated systems have been installed.

6.2.3.1 Use of systems

The approach with the use of dedicated systems is based on the idea that to end up in such a rare situation as a severe accident, the systems used to prevent the accident must have failed in some way (otherwise the accident would be prevented) and independent systems are needed to manage the consequences of the severe accident. The dedicated systems, whose purpose is to ensure continuing containment leaktightness, are independent from the systems used for accident prevention and design basis accident management. These systems create an additional defensive line against severe accident environmental releases.

A typical approach, and systems used, can be:

- Primary circuit depressurisation with dedicated depressurisation valves;
- Corium stabilisation and cooling in core catcher (systems enabling the flooding of the core catcher);

- Cavity flooding with use of the dedicated systems (in-vessel retention of molten corium);
- Systems enabling / favouring hydrogen management (measures to mix the containment atmosphere);
- Hydrogen management (PARs, passive hydrogen igniters, glow plugs);
- Dedicated system for containment heat removal / management of long-term overpressurisation (dedicated spray, passive containment heat removal systems, filtered venting / purge systems);
- Dedicated instrumentation & control system;
- Independent/additional system for electricity feed for dedicated SAM systems and I&C systems.

All the dedicated systems included in the SAM strategy should be included in L2PSA if the PSA is to claim credit for accident management. Typically these systems are qualified against severe accident environmental conditions, as explained in the next chapter, or survivability of them in severe accident environmental conditions is demonstrated. When this is the case, the system modelling in L2PSA is very close to reliability modelling used in L1PSA. However, if the qualification of the systems is not done, the usability of the system in severe accident environmental conditions should be separately evaluated and taken into account in reliability modelling, just as it should be done in the approach without dedicated systems.

6.2.3.2 Qualification of systems

System qualification actually means the qualification to prevailing environmental conditions of both mechanical components and automation equipment (including cabling) which, in a severe accident, is more demanding than for design basis accident conditions.

For all the SAM equipment which is located inside the containment, the environmental conditions in the containment have to be taken into account. The equipment located outside containment is not exposed to extreme environmental conditions and special qualification is not needed; however evaluation case by case is recommended. In the containment, the thermal hydraulic and radiation conditions have to be taken into account.

It is suggested to group functions required during severe accident into main functions and supporting functions. Main functions are the functions that are definitely required in severe accident situation. For example, controls and measurements that are part of SAMG and required for successful containment integrity retention are main functions. Supporting functions might be used to gain additional information on the progression of a severe accident, but supporting functions are not necessarily required for successful severe accident consequence mitigation. For main functions qualification against severe accident environmental conditions is necessary, for supporting functions qualification rules can be less strict.

The qualification time, which will be applied for individual equipment, depends on the function of the equipment. For example, requirements are different for the long-term monitoring equipment, that has to stand the conditions for years, and the equipment that is used once in the beginning of a severe accident.

For some of the SAM equipment, the qualification against LOCA conditions might be sufficient if it will be operated at the early stage of an accident where prevailing containment conditions are like in large LOCA, i.e. before core damage has occurred. For example, closing of the inner containment isolation valves, when valves are the controlled type of valves, and opening of the primary circuit dedicated depressurisation valves will be made at the early stage of an accident.

Qualification time can be for example:

- 72 hours for components that are needed at the early stage of an accident;
- 1 year for long-term monitoring system.

Pressure and temperature requirements for each item of equipment can be obtained by combining long lasting, rather steady conditions and transient pressure and temperature peaks. Long term temperature and pressure conditions can be assessed based on thermal hydraulic severe accident analysis and high peaks are mainly related with hydrogen burns. The most demanding conditions should be chosen or probabilistic criteria can be used to choose representative conditions.

Conditions after hydrogen burn can be assessed by analysing different hydrogen release scenarios with code calculations and choosing the representative environmental conditions. It should be ensured that models for hydrogen burn phenomena that are used are reasonably well validated when applied at a local scale, in terms of loading on specific systems. Conditions after hydrogen deflagration are different, depending on the hydrogen release rate and assumptions used in hydrogen ignition. Also, the presence of hydrogen management systems affects the conditions and this should be taken into account when analyses are made.

When plant specific radiation loads are assessed, both the radiation load during normal operation (from the entire plant lifetime) and additional radiation load during a severe accident should be taken into account. Radiation loads differ in different containment compartments. If important equipment is located below expected water levels, the radiation doses to these equipments should be separately evaluated. In the long-run the activity in the containment will be deposited into the water pools, and doses for components located in the water pool might be significantly higher.

The effects arising from separate issues such as hydrogen diffusion flames, cable burns, component submergence to water and bursting of hot gases to steam generation space have to be taken into account. This can be done also by using redundant systems, physical separation and avoiding placing of components near possible primary system leak locations. For example hydrogen recombiners and glow plug igniters could be located in a way that, in case of very high local temperatures i.e. hydrogen diffusion flames, only a minor portion of the components could get damaged.

6.2.4 Specific system descriptions

6.2.4.1 <u>Core catchers</u>

Core catchers are specific equipment designed for arresting and cooling the corium which is spread from pressure vessel after the pressure vessel failure. Core catchers are part of the design for some of the GEN III+ plants and detailed design of a core catcher varies depending on the plant design.

A core catcher is located in a way that it allows corium that is spread out from pressure vessel, in case of pressure vessel melt through, to flow to the core catcher. In some reactors like EPR, the core catcher design consists of layers of sacrificial concrete below which there are cooling channels, so that corium can be cooled from below. The structure might be separated from the basemat with refractory layers. After corium has spread to the core catcher, the core catcher is flooded so that eventually corium will be cooled from bottom and below. There are also some core catcher designs in which the corium is only cooled from above.

Core catcher designs are tested experimentally. Some important issues for operability of a core catcher are material issues e.g. the suitability of sacrificial concrete material and its effects on corium viscosity and non-condensable gas production. Heat transfer from the corium to sacrificial material and to cooling channels below, and heat transfer from corium to water pool above are also important issues.

The core catcher can operate either fully passively or the operation can be partly dependent on operator actions. For example, flooding of the core catcher after corium has spread out can be made either passively, by actuating the valve opening from an appropriate signal, e.g. high temperature in the core catcher, or flooding can be actuated manually by operator. In EUR it is required that a Corium Collecting and Cooling Device (CCCD), usually called a core-catcher, shall not have any active components inside the containment.

The assessment of core catcher reliability should be included in L2PSA, taking into consideration the issues important for specific core catcher design. The detailed example from AREVA EPR^{TM} has been given in section 6.3.4.

6.2.4.2 Filtered venting systems

Containment filtered venting systems are dealt with in section [125] (In-vessel) and 4.5.6 (Ex-vessel).

6.2.4.3 Passive systems

Passive systems are widely used for severe accident management in GEN III+ plants, but some severe accident management features of currently operating power plants are also based on passive systems. Passive systems are normally designed to operate in certain conditions and the reliability of systems is very high, as long as the actual operating conditions are as assumed in design. However, as for active systems, the reliability of passive systems performing a safety function in severe accident conditions must be considered, and any passive systems installed specifically for SAM should be qualified for severe accident conditions. Currently there are no common methods for reliability assessment of passive systems, since the use of systems is still rather new.

In this chapter the most common passive safety features and specific issues to take into consideration in level 2 are described.

Passive Autocatalytic Recombiners (PARs)

PARs are installed in many plants for hydrogen risk management. PARs are passive and self-starting; they have extended working conditions and are resistant to poisoning agents. They are active at low temperature and withstand high temperature with high humidity. They are periodically tested to check their performances and to keep a watch on potential ageing effect.

Consequently, PARs can be considered as highly reliable systems in severe accident conditions. Nevertheless, a sensitivity study could be performed for a penalising case with a catalytic surface partially unavailable, to assess the impact on hydrogen risk.

Passive containment heat removal systems

Passive containment heat removal systems are part of the design of some of the Gen III+ plants. Containment heat removal is considered to be passive in cases where heat exchangers (and/or possibly water pools) are part of the containment and heat removal will be passively actuated in case of pressure rise in the containment. Typically in passive containment heat removal systems, the primary side of the heat exchanger is connected with the containment. In case of pressure rise inside the containment, the steam flows to the heat exchanger where steam is condensed. The primary side of the heat exchanger is part of the containment boundary. The condensate is led back to the containment and a natural convection loop flow is formed. The secondary side of the heat exchanger (for example a water pool) is not part of the containment. The heat exchanger might be located either inside the containment or, as in most cases, outside of the containment. In the latter case there are penetrations that lead steam from the containment to the heat exchanger and condensate back to the containment.

The operability of passive heat exchangers in severe accident conditions should be verified experimentally to show that system works as expected in all possible scenarios and conditions. During verification it is possible to identify issues that might have an effect on operability of passive heat exchangers. For example, the amount of non-condensible gases that would be generated during a severe accident might be different in different scenarios and the effect of non-condensible gases should be taken into account when passive heat exchanger operability is assessed. In many cases some parts of passive heat exchanger are also part of containment boundary. The heat exchanger should not jeopardise the containment integrity and the possible failure of the heat exchanger should be assessed in L2PSA.

Passive features favouring the containment atmosphere mixing

Mixing of the containment atmosphere favours the hydrogen management. The idea is to make sure that the hydrogen concentration in the containment is low enough to assure that detonable mixtures are not formed. Hydrogen formation, containment atmosphere composition and hydrogen combustion in different phases of a severe accident are handled in sections 4.3.3, 4.3.6, 4.3.8 and 4.5.5 of this document.

The containment atmosphere mixing can be promoted by using either active systems or passive structures that force open flow routes between different containment compartments. Forcing open of hatches, or fans, are typical active systems. Passive systems can be flanges, rupture disks or foils that open passively in severe accidents due to containment conditions. Opening can be initiated by temperature or pressure differences between containment compartments.

Verification is important for features favouring containment atmosphere mixing. It should be reasonably well shown that systems will operate as expected in all possible conditions during severe accident. When verification has been done properly, systems can be considered as very reliable. In L2PSA the effect of partial unavailability or mis-operation can be studied, for example with sensitivity studies.

6.3 PARTNER EXAMPLES

6.3.1 Tractebel Engineering example from Belgian PWR (WOG SAMG) [256]

This paragraph presents the Belgian example for the use of systems in PWRs for each severe accident safety function (see section 6.2.1) based on WOG SAMG.

The systems considered in L2PSA that can be used for mitigation / reduction of radionuclide releases depends on the origin of radionuclide releases.

If FP are detected in the containment, the systems are:

- Containment spray pumps;
- Containment fan coolers;
- Alternate means of injection into containment.

Containment spray pumps can be used either in direct injection mode or in recirculation mode. Alternate means of injection into containment does not consider one specific system: they can include any non-standard plant-specific equipment that can be used for injection into containment. They also include the possibility to refill the RWST.

Containment spray pumps and fan coolers are considered separately in the APET. Containment spray pumps are first considered as they are the most efficient. When the start of containment spray pumps is not performed due to a human error, the start of containment fan coolers or of an alternate means is assumed not to be successful. The alternate means of injection into containment are not considered as attributes: their availability has to be assessed in combination with the probability from human reliability analysis.

If FP are released to the environment via the steam generators (in case of SGTR, there is direct pathway to the atmosphere via steam generators relief valves), is the following are considered in L2PSA:

- The possibility to isolate the ruptured steam generator;
- The possibility to scrub leaking fission products by injecting into the ruptured steam generator (details for injection into the steam generators are given in the dedicated paragraph).

If FP are released to the environment via the auxiliary building, is the following are considered in L2PSA:

- The possibility to isolate the containment if FP releases are detected in the containment: in case of failure of the containment isolation signal or in case the signal was not emitted (possible in some shutdown states);
- The different ventilation/filtration systems outside containment.

The systems considered in L2PSA that can be used for injection into the SG are:

- Auxiliary feedwater;
- 2nd level feedwater: such a system may not exist.

For the steam generators, their availability depends on the availability of one of the two (or any other) feedwater system. The steam generators are used in the APET to provide a heat sink for the RCS allowing evacuating the decay heat and depressurising the RCS. Their availability and the presence of water in the tubes protect the steam generator tubes from the occurrence of an induced SGTR by creep failure. In case of SGTR, if the ruptured steam generator can not be isolated, the possibility to feed the ruptured steam generator is assessed: it allows decreasing releases by scrubbing fission products.

The systems considered in L2PSA that can be used for RCS depressurisation are:

- Pressurizer Power Operated Relief Valves (PORV);
- Steam generators depressurisation followed by feed and bleed operation.

Concerning the pressurizer PORVs, the possibility to have at least one stuck-open relief valve while cycling is considered in addition to their manual opening. The possibility to have a hot leg or surge line creep failure is accounted for in the APET and in such a case RCS depressurisation is induced. Steam generators feed and bleed request the availability of one feedwater system with the availability of the steam generators safety or relief valves.

The systems considered in L2PSA that can be used for RCS injection are:

- High Head Safety Injection(HHSI);
- Low Head Safety Injection (LHSI);
- 2nd level safety injection: such a system may not exist;
- Alternate means of RCS injection.

High Head and Low Head Safety Injection systems can be used either in direct injection mode or in recirculation mode.

The first three systems (High Head, Low Head and 2nd level Safety injection) are considered as main (most usual and efficient) RCS injection means and each of them is considered separately in the APET. Alternate means of RCS injection does not consider one specific system: they can include Chemical and Volume Control system or any non-standard plant-specific equipment that can be used for RCS injection. They also include the possibility to refill the RWST.

The possibility to use the three main systems is not only related to their availability (transmitted by level 1 through attributes): it depends also on the RCS pressure (not for 2nd level safety injection) and the presence of a water source. The definition of the different RCS pressure levels has to be at least partially based on the shutoff head for the pumps of the system.

The alternate means of RCS injection are not considered as attributes: their availability has to be assessed in combination with the probability from HRA.

The systems considered in L2PSA that can be used for injection into containment (to flood containment and reactor cavity if there is a path between containment sumps and reactor cavity) are:

- Containment spray pumps;
- RWST gravity drain.

Containment spray pumps in this case are only considered in direct injection mode. RWST gravity drain may not be possible (according to plant-specific design).

The systems considered in L2PSA that can be used for control of containment conditions and containment depressurisation are:

- Containment spray pumps;
- Containment fan coolers;
- Alternate means of injection into containment;
- Containment venting.

Containment spray pumps can be used either in direct injection mode or in recirculation mode. Containment spray pumps, fan coolers and venting are considered separately in the APET. Containment spray pumps are first considered as they are the most efficient. Alternate means of injection into containment does not consider one specific system: they can include any non-standard plant-specific equipment that can be used for injection into containment. They also include the possibility to refill the water storage tank.

When the start of containment spray pumps is not performed due to a human error, the start of containment fan coolers or containment venting is assumed not to be successful. The alternate means of injection into containment and containment venting are not considered as attributes: their availability has to be assessed in combination with the probability from HRA.

Other systems can be used for the functions depending on plant-specific capabilities. For example, in Belgium, a connection between the LHSI system and the containment spray system allows using the pumps of those systems for injection into the RCS and injection into containment. The possibility to make that connection and to use the pumps of those systems for the two functions is considered in Belgian L2PSA.

The systems used for the reduction of hydrogen content in the containment are the Passive Autocatalytic Recombiners (PARs). PARs were designed for passive hydrogen risk management. As they are periodically tested to check their performances, the complete catalytic surface is taken into account in the supporting calculations to follow the hydrogen content evolution in the containment. For Belgian units, specific equipments for Severe Accident management are neither required nor qualified, except the PARs which have been qualified by Siemens. Other systems, namely the containment spray and the fan coolers, influence the hydrogen flammability. Both have the capacity to condense the steam which may provoke the flammability of the containment atmosphere which was initially inert. However, they have the advantage to decrease the base pressure so that the loads amplitude in case of combustion will be lower. Moreover, they allow the mixing of the atmosphere between the different compartments of the containment and avoid stratification effects. The inhomogeneities that could have led to local flammable mixtures are thus limited.

Regarding the hydrogen ignition risk, hot particles separated from the catalyst with elevated surface temperature above 700°C (973K) have been shown to lead to ignition for some recombiners [257]. To take this risk into account, higher ignition probabilities are assigned in compartments where PARs are installed.

Regarding the qualification of Severe Accidents equipments as a whole, the Belgian approach is based on that of the WOG. The WOG approach consists of using (mitigation) means existing within the plants, even if those means are used beyond their design basis (as it is the case for Severe Accidents). In addition, during the development of Severe Accident Management Guidelines, the survivability/adequacy of instrumentation has been addressed and the availability of necessary information to monitor Severe Accidents key parameters as recommended by the WOG has been assessed.

6.3.2 Fortum example of SAM with the use of dedicated systems

In Finland, severe accident management with dedicated systems is required. Systems credited in severe accident management have to be independent from the systems used to cope with design basis accidents. The single failure criterion is also required. All the instrumentation and control systems have to be qualified against severe accident environmental conditions.

The Loviisa NPP is a two-unit VVER-440 plant with ice condenser containment. Units 1 and 2 were commissioned in 1977 and 1981 respectively. The reference plant concept did not include the containment.

Since a containment was definitely required in Finland, ice condenser containments were built. Other significant modifications to the original plant design were also carried out, most notably modifications of the ECCS, the reactor coolant pumps, and the inclusion of Siemens I&C systems.

Studies considering severe accident management for the Loviisa NPP have been structured around the identified containment-threatening mechanisms. The aim has been to find solutions that would reliably protect the containment. It has to be recognised that even though the Loviisa NPP has certain well-known vulnerabilities to severe accident phenomena, it also presents some unique opportunities for selection of mitigation strategies. For example, water from melting the ice would quickly (and passively) flood the small-sized cavity in an accident. This feature, in combination with the fact that the decay power level is low and the reactor pressure vessel lower head has no penetrations, makes in-vessel retention of molten corium feasible through external cooling of the RPV. A well-known vulnerability is that the ice-condenser containment has a rather low estimated failure pressure in relation to loads that could take place during severe accident (e.g. from global hydrogen deflagrations). On the other hand, it was found that the ice condenser configuration would ensure efficient mixing of the containment atmosphere, in the case where the ice condenser doors were forced open. The containment steel shell makes it possible to control long-term pressurisation through external cooling.

Implementation of the SAM approach at the Loviisa NPP includes several different lines of action. The most notable tasks are the following:

- Hardware modifications have been carried out at the plant to ensure that core damage can be reliably prevented and severe accident phenomena can be mitigated;
- Substantial new I&C (instrumentation and control) qualified for severe accident conditions has been installed;
- New SAM guidelines, procedures, and a SAM Handbook have been written;
- The emergency preparedness organisation has been revised;
- Versatile training approaches, including the development of a severe accident simulator, APROS SA [246] are being developed.

The Integrated ROAAM (Risk Oriented Accident Analysis Methodology) approach was applied for the development of an overall SAM strategy for Loviisa [247], [248] . The strategy that ensures a sound balance between prevention of core damage and mitigation of containment-threatening phenomena consists of four steps:

- The reliability of prevention of core damage should be demonstrated by a L1PSA to meet the prevention requirements;
- Prevention of core melt sequences with imminent threat of a large release (usually sequences with an impaired containment function) that cannot be mitigated has to be demonstrated to be sufficiently reliable according to PSA;
- Reliable mitigation of severe accident phenomena that could pose a threat to containment integrity should be demonstrated for all relevant accident scenarios;

- To show a compliance with Finnish safety requirements with regards to SAM, we have to demonstrate that radioactive release limit[23] is not exceeded due to normal leakages out of an intact containment in a severe accident.

Sequences with an imminent threat of a large radioactive release are e.g. primary-to-secondary leakages, interfacing system LOCAs, high-pressure sequences, and reactivity-induced accidents. The goal is to ensure that such sequences can be rendered into the residual risk category, which in some cases has warranted plant modifications to reduce their frequencies. In other cases we have taken the approach to update the L1PSA so that certain overly conservative modelling assumptions could be modified into more best-estimate ones. Dealing with the containment bypass sequences identified in the L1PSA has forced us to develop our source term calculation capabilities, and to study aerosol behaviour in containment and pipes. Retention of aerosols in narrow pipes may act as a significant mitigating mechanism in some of the interfacing system LOCA sequences.

The mitigation part of the SAM approach is built around the following SAM safety functions:

- Successful containment isolation: New approaches for actuating isolation signals, ensuring isolation status, and monitoring containment leak-tightness have been developed;
- Primary system depressurisation: Installation of high-capacity depressurisation valves (manually operated relief valves), which are separate from the primary system safety relief valves;
- Absence of energetic events (mitigation of hydrogen combustion, since successful in-vessel retention of molten corium excludes other energetic events.) A new hydrogen mitigation scheme based on containment mixing through forcing open ice condenser doors, and controlled removal of hydrogen through passive autocatalytic recombiners (PARs) and deliberate ignition has been developed [249];
- Cooling of the reactor core or core debris (reactor pressure vessel lower head coolability and melt retention). Features which enhance the possibility at Loviisa for RPV lower head coolability and melt retention are a flooded cavity, a lower decay power level, and a RPV lower head without penetrations are fulfilled in case of Loviisa [250]. Certain plant modifications were necessary to ensure e.g. access of water to the vessel wall and sufficient flow paths for steam at the boiling channel;
- Mitigation of slow containment overpressurisation (long-term containment cooling): the approach was taken to install a containment external spray system instead of filtered venting due to certain Loviisa-specific features such as sensitivity to sub-atmospheric pressures and low steaming rates 337. No other non-condensible gases than hydrogen are generated and containment steel shell makes it possible to cool containment from the outside;

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[23] In Finland the Government Decree on the safety of nuclear power plants (27.11.2008/733) says: "The limit for the release of radioactive materials arising from a severe accident is a release which causes neither acute harmful health effects to the population in the vicinity of the nuclear power plant, nor any long-term restrictions on the use of extensive areas of land and water.

The requirement applied to long-term effects will be satisfied if there is only an extremely small possibility that, as the result of a severe accident, atmospheric release of cesium-137 will exceed the limit of 100 terabecquerel (TBq)."

- Monitoring of sub-criticality of the core;
- Ensuring of coolability of spent fuel pools.

The goal of SAM approach is to achieve severe accident safe state (SASS).

All aspects of the strategy, like hardware and I&C modifications, have been targeted towards ensuring the safety functions in a highly reliable manner. The SAM guidelines and procedures and the SAM Handbook have also been structured around the SAM safety functions. SAM safety functions and success or failure of mitigation systems affect L2PSA and are taken into account in source term calculations.

During recent years, SAM in sequence classes arising from plant shutdown states have been studied and plenty of operational and procedural changes have been made to ensure the ability to manage severe accidents during shutdown states. This work is still ongoing.

6.3.3 Example from Iberdrola Eng. on an integrated containment flooding strategy for BWR (BWROG SAMG) [258]

The objectives of primary containment flooding are: re-establish core cooling, remove heat from the RPV, retain core debris in the RPV, quench debris outside the RPV, preserve containment integrity, scrub fission products, minimise MCCI and facilitate long-term recovery. On the contrary, the consequences of the containment flooding may require venting, loss of SRVs, loss of vent capability, loss of pressure suppression capability or loss of electrical equipment.

BWROG accident management principles related to integrated containment flooding strategy and grouped by severe accident phenomenology are:

- Steam explosion:
  - A large ex-vessel steam explosion is unlikely.
- Molten Core Concrete Interaction (MCCI):
  - MCCI will continue until ex-vessel core debris is flooded and quenched.
- Recriticality:
  - A debris bed in-vessel or ex-vessel will not become critical if submerged with water.
- External vessel cooling:
  - This phenomenon is only possible with a design of the cavity that permits the natural circulation of the hot flows. If it is not possible, only a short delay of RPV failure will occur~~,~~
  - Flooding the containment to above the RPV lower head before lower plenum dryout will preclude subsequent in-core instrument thimble failures.
- Ex-vessel debris cooling:
  - Submerging all core debris in the Mark I drywell will preclude drywell failure due to creep rupture or melt-through of the liner at the core debris/liner interface,
  - Ex-vessel debris must be cooled to preclude containment failure,
  - A severe accident will not be controlled and terminated until all fuel and core debris is quenched and submerged.

The main priority of the containment flooding strategy is to mitigate the breach whilst maintaining the pressure suppression capability, establishing a pool of water on the drywell and submerging ex-vessel debris. For a LOCA initiator there are additional priorities related to the RPV flooding through the break: restore RPV water level up to active fuel and prevent RPV breach by core debris, submerging the debris in-vessel and maintaining a sufficient RPV injection for molten debris heat removal.

To cover this strategy some functions can be used:

1. RPV and primary containment injection:
   - While core debris remains in the RPV, injection systems should be preferentially aligned to retain the debris, even if primary containment integrity is challenged.,
   - If RPV breach by core debris is anticipated, primary containment flooding should be restricted to preserve pressure suppression capability, even if RPV injection must be reduced,
   - If it is determined that core debris has breached the RPV, injection systems should be preferentially aligned to submerge the debris,
   - The primary containment should be flooded only if a large primary system break may exist, pressure suppression capability is not required, and SRV actuation is unlikely,
   - Injection directed to the location of core debris should be reduced only if the core melt progression will not be accelerated, even if continued injection may challenge primary containment pressure and level limits.
2. Primary containment venting:
   - Primary containment venting is appropriate if it will prevent RPV breach by core debris or a containment failure resulting in an uncontrolled radionuclide release,
   - Early venting, before significant fuel damage or RPV breach has occurred, may be appropriate if it will avoid the need to vent later, when the potential for radionuclide release is greater,
   - If RPV breach by core debris is anticipated, primary containment venting is appropriate to establish or preserve pressure suppression capability.
3. RPV venting:
   - Venting the RPV will remove trapped noncondensibles, thereby permitting water to fill the RPV during primary containment flooding, but will likely release significant radioactivity. RPV venting should therefore be delayed as long as possible and performed only when an immediate and significant benefit is likely.
4. Containment sprays:
   - Spray operation may be required to:
     a. Control primary containment temperature and pressure,
     b. Scrub the containment atmosphere,
     c. Reduce the containment peak pressure in case of gas combustion,
     d. Cool unsubmerged debris in the drywell,
     e. Add water to the drywell in anticipation of RPV breach or to maintain pressure suppression capability.

- At most plants, containment spray operation requires diversion of RPV injection flow. Containment spray operation must therefore be coordinated with RPV injection,
- At some plants, a water source outside the primary containment can be aligned to supply containment sprays. If primary containment flooding is required, sprays should be aligned to sources external to the primary containment, if possible, to increase the flooding rate. However, the additional water may also challenge primary containment limits. Use of external spray sources must therefore be coordinated with other actions taken to preserve primary containment integrity.

The availability of systems used for the containment flooding strategy will be analysed in the different stages of the severe accident progression. Processes of dynamic pressurisation with impact on the structures like steam explosion, DCH or hydrogen detonation, may also cause the failure of systems. This failure probability changes with the design and location of each system. Some generic failure values can be found in the NUREG/CR-4551 (50% for internal systems and 10% for external system).

6.3.4 Example on core catcher

To ensure the integrity of the containment in a core melt accident some nuclear power plants are equipped with a core catcher. The aim is to collect and store the melt, possibly in a special device where it can be cooled and stabilised so that MCCI is stopped, a penetration of the basemat can be prevented and the integrity of the containment is maintained as the melt cannot attack parts of the containment shell like liner or concrete walls.

In the L2PSA, events that might jeopardise the functioning of the core catcher have to be identified. The probabilities of such events and their consequences have to be evaluated.

In the section some general aspects are discussed before giving ideas on how a core catcher can be modelled in the L2PSA are given based on the example of the EPR™.

6.3.4.1 General discussion

Depending on whether the core catcher relies on presence of water at time of RPV failure or on a dry reactor cavity one has to analyse the following items:

- The availability of water and possibly the risk of a steam explosion, the unavailability of water, the consequences to the core catcher.

The effect of different corium release modes from the RPV have to be investigated.

If the core catcher involves quenching of the melt by water, steam and hydrogen production and the resulting pressure increase in the containment have to be analysed, taking into account quenching, production and combustion of hydrogen at the same time.

If the core catcher relies on fragmentation of the melt the effect of steam explosion on the one hand and the effect of non-fragmentation on the other hand in relation to the RPV breach size and location and the RCS pressure have to be investigated.

If the core catcher relies on active measures (triggered by e.g. electrical signals, operators actions etc.), the success of the activation signal, human action (if applicable) and the non-functioning of devices have to be quantified.
If the core catcher relies on passive operation of devices (triggered by physical processes alone), such as valves, the failure of the activation of the devices and their non-functioning have to be analysed.
Finally a careful investigation of all possible long term threats is essential.

6.3.4.2 Identification of events that might fail the core catcher (example of EPR™)

The main components of the EPR™ core melt stabilisation system are:

- The sacrificial and protective layer in the reactor pit,
- The melt plug,
- The melt discharge channel,
- The core catcher in the spreading compartment.

The purpose of the EPR™ core melt stabilisation system is to stabilise the core melt in the containment without exceeding loads critical to the integrity of the containment liner. Melt stabilisation is achieved by removing the residual heat from the spread melt in the core catcher. Heat removal is performed by cooling the bottom and sidewall structures of the core catcher as well as the melt's free surface with water from the in-containment residual water storage tank (IRWST).
The phases of melt stabilisation are as follows:

- Temporary melt retention in the reactor pit,

The core melt stabilisation includes an initial phase of temporary melt retention in the reactor pit immediately following melt discharge from the RPV. Temporary retention is achieved by sacrificial concrete that has to be dissolved by the melt. During this phase, the residual heat generated in the melt is partially consumed by the ablation of the sacrificial concrete and partially transferred to the residual RPV and the surrounding concrete structures.

- Melt outflow through the melt discharge channel,

During transfer from the reactor pit to the core catcher through the melt discharge channel, a fraction of the stored and residual heat is transferred to the surrounding structure by direct contact and thermal radiation from the melt free surface.

- Melt spreading into the core catcher,

During spreading of the corium into the core catcher, the upper surface of the melt is not yet covered with water. In this phase heat is removed from the melt's free surface and transferred to the surrounding structures, i.e. the ceiling of the spreading compartment, predominately by thermal radiation. At the same time heat is removed from the melt through the heat-up of the sidewall and bottom structures of the core catcher.

- Melt flooding,
- By design, all the melt is collected temporarily in the pit before the gate melts through. After penetration of the plug, the melt is guided through a melt discharge channel into the core catcher.

The flooding of the melt is initiated passively: upon arrival, the melt triggers the opening of the redundant valves (the spreading melt will thermally destroy receptors that relieve pre-stressed cables which have kept the valves closed, then the valves will open automatically) that start the gravity-driven overflow of water from the IRWST. The water is distributed by means of a central supply duct into a system of flow channels at the bottom and side of the core catcher. After submerging these channels the water spills over onto the melt's surface from the edge. Long term melt cooling,

Passive water overflow continues until the hydrostatic pressure levels within spreading room and in-containment residual water storage tank (IRWST) are balanced. Then the spreading compartment, melt discharge channel, and the lower pit are also flooded. In this passive mode of core catcher operation, the water in the spreading room is saturated, so practically all decay heat is converted into steam which enters the containment. The evaporated water is constantly re-supplied by overflow from the IRWST.

As an alternative to this passive mode of operation, the containment heat removal system (CHRS) can be used to actively feed cold water into the core catcher. As a result, the water in the cooling channels and in the water pool atop the melt will become sub-cooled. Decay heat will then be removed from the spread melt by single-phase flow, instead by evaporation.

Heat removal from the melt can be achieved in the mid- and long-term by both by passive and active mode of operation.

The core melt stabilisation system ensures the containment of radioactive substances:

- liquid and solid radioactive substances contained in the core melt will become immobilised and stored within the core catcher and will not penetrate the basemat containment barrier;
- it prevents the spreading of contaminated water in the ground ;
- it also avoids an uncontrolled release of the radioactive gas outside the containment through the basemat.

The following events are considered to endanger the functionality of the EPR^{TM} core melt retention system:

- Premature opening of the melt plug before all melt has been collected in the reactor pit

There are two mechanisms for a premature opening of the gate: energetic events which lead to a pressure build-up in the reactor pit exceeding the failure pressure of the melt plug and penetration of the gate due to MCCI faster than foreseen by design.

Pressure build-up in the pit may, in principle, result from high pressure RPV failure where steam is blown down into the reactor pit. Furthermore there is a possibility that the melt plug is failed as a consequence of a violent melt water interaction in the reactor pit (ex-vessel steam explosion).

Premature opening of the gate from MCCI before all melt has been collected in the pit is unlikely but is imaginable if there is an incomplete release of melt from the RPV into the pit and gate penetration while the rest of the core material is still remaining in the core. One possible consequence of a premature opening of the melt plug is that the melt in the core catcher is flooded and the pit fills with water before all melt has left the RPV. Then a late pour of melt may lead to a violent fuel coolant interaction. Furthermore, there is a potential for unlimited MCCI in the pit and basemat melt-through.

- Failure of melt flooding,

Failure of melt flooding can result from failure of the valve to open or from flood valve initiation failure.

- Containment failure from over-pressurisation following melt quenching,

In case of successful flooding of the melt in the spreading area the containment is pressurised from quenching the melt.

- Failure of the long-term retention function,

Heat removal from the melt can be achieved in the mid- and long-term by both passive and active mode of operation.

However, the long-term retention function is in danger if the supply of water from the IRWST is not guaranteed.

- High pressure vessel failure.

It is likely that a core catcher cannot be credited in case of high pressure breach of the RPV. In this case melt is dispersed throughout the containment. Depending on the geometry of the reactor pit, the following issues have to be analysed:

- What is the minimum failure pressure for significant dispersal?
- What is the minimum amount of melt that can accumulate outside of the core catcher to fail the basemat?

6.3.4.3 Evaluation of probabilities

In this section the issues used to derive failure probabilities for the above mentioned events is described. No actual probabilities are given but the events are broken down into sub-events the probability of which can be derived easier.

- **Premature opening of the melt plug before all melt has been collected in the reactor pit due to MCCI faster than foreseen by design**

The following series of events is investigated:

- Incomplete release of melt from the vessel into the pit,
- Premature opening of the gate due to MCCI before all melt has been collected in the pit,
- Late pour of melt which forms a non-coolable configuration and leads to basemat penetration.

- **Premature opening of the melt plug following violent fuel coolant interaction**

The following series of events is investigated:

- Presence of water in the reactor pit. By design the EPR^{TM} pit is dry. Hence exceptional conditions have to be identified and quantified which might lead to water in the pit, for example an induced RCS rupture at the reactor pressure vessel nozzle for high pressure transients,
- Triggering of a violent fuel coolant interaction,
- Melt plug failure due to violent fuel coolant interaction,
- Basemat penetration in case of melt plug failure.

- **Failure of melt flooding**

In the L2PSA the probability of failure of melt flooding is broken down into two aspects: unavailability of flood valves and failure of flooding initiation.

- Unavailability of flood valves: There are two redundant butterfly flood valves. The concurrent unavailability of both valves can be evaluated in a fault tree including common cause failure.
- Probability for flooding valve initiation failure.

- **Containment failure from over-pressurisation following melt quenching**

The flooding of the melt in the core catcher leads to release of steam into the containment with the potential of containment failure due to the pressure spike. To evaluate the failure probability the pressure loads have to be compared to the load bearing capacity of the containment:

- Pressure loads from steam spiking must be evaluated taking into account the degree of fragmentation.
- The load bearing capacity of the containment with respect to pressure loads is based on the structural analysis of the containment where the pressure limit for a failure of the containment concrete shell or disturbances such as hatches, penetrations etc. are evaluated.

- **Failure of the long term retention function**

The long-term retention function is in danger if the supply of water from the IRWST is not guaranteed. One mechanism would be clogging of the cooling structure by debris which is, however eliminated in practise by the sump screens of the sump suction. Another mechanism is, in case of long-term operation of a containment venting system, the loss of IRWST water through the vent line if IRWST make-up is not performed.

6.3.4.4 Evaluation of consequences

Apart from the derivation of probabilities the consequences of a failure of the core melt stabilisation system have to be evaluated.

For containment overpressure failure due to melt quenching, the consequence is either a containment leak or a containment rupture based on the results of the structural analysis.

For all other events described above a non-functioning of the core melt stabilisation system leads to unlimited MCCI in the spreading area. Apart from the eventual basemat penetration and possible consequences on water contamination etc. (ground path) the question is whether an equivalent air path is connected with this failure mode, such as leakages of the liner, cracks in the concrete or in the worst case a loss of containment integrity. Hence the source term may lie between an enhanced design leakage and that of a containment leak.

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7 SOURCE TERM ASSESSMENT

7.1 INTRODUCTION

The source term assessment provides information about the characteristics of the release categories in terms of composition of the release and the time of release. The source term is combined with release category frequencies in presentation of results. Depending on the scope of the PSA, source terms can be simple e.g. above or below a certain threshold of released quantity, or sophisticated e.g. time dependent release rates of different isotopes for further processing in a PSA level 3.

The source term calculations are usually based on the choice of one or more sequences to be representative for each release category. If there is more than one source term calculation for a single release category, this information can be used to both support the definition of release categories and the assignment of release categories to different release sequences in the APET.

The source term assessment process includes the following steps:

- Choice of representative severe accident sequences within each release category,
- Identify the needs of the source term characterisation due to L2PSA objectives and criteria.
- Calculation of source terms for the representative severe accident sequences.

This chapter concentrates on the issues that influence the magnitude of environmental releases divided into:

- Phenomenological issues influencing radionuclide transport,
- Modelling issues in both integral codes and dedicated (fast running) source term models.

The chapter also introduces the concept of allocating a hazard index to source terms using a simplified off site dose assessment methodology. This issue is sometimes considered within a Level 2+ PSA.

7.2 IMPORTANT STRATEGIES FOR DIFFERENT PURPOSES / END-USERS NEEDS

The end user survey identified 6 areas of L2PSA applications to be prioritised in the development of this guidance document:

1. To gain insights into the progression of severe accidents and containment performance.
2. To identify plant specific challenges and vulnerabilities of the containment to severe accidents.
3. To provide an input to determining whether quantitative safety criteria which typically relate to large release frequencies (LRF) and large early release frequencies (LERF) are met.
4. To identify major containment failure modes and their frequencies, including bypass sequences; and to estimate the corresponding frequency and magnitude of radionuclide releases.

5. To provide an input to the development of plant specific accident management guidance and strategies.

6. To provide an input to plant specific risk reduction options, especially in view of issues such as ageing, plant upgrades, lifetime extension, decision making in improvements, maintenance, and cost benefit analyses.

All the objectives are supported by some kind of source term assessment, but all do not require that a detailed calculation is performed. A more detailed assessment is needed for objective number 4. For objective number 3, it is necessary only to estimate whether the release is above or below the stated LERF/LRF threshold.

End user objectives 5 and 6 will also need some source term assessment if the mitigation of releases to the environment is seen as the final goal of accident management and risk reduction.

Whenever environmental effects are to be considered, it has to be defined which are the most relevant radionuclides. If, for example, acute health effects to the public are in the focus, Iodine will be the key element. If long term restrictions of land use are of interest, caesium will be more important. If hazard to the public or to the environment has to be determined quantitatively, e.g. in a PSA level 3, all relevant nuclides have to be taken into account. From a technical point of view this does not require a large additional effort since most PSA apply integral computer codes for analysis, and these routinely provide source terms for all radionuclides of interest anyway.

7.3 CALCULATION OF SOURCE TERMS FOR REPRESENTATIVE SEVERE ACCIDENT SEQUENCES

There is a broad base of published information regarding “source terms” for severe reactor accidents from the USA, e.g. [262], [263], [264], [265]. However, this definition of “source term” is usually restricted in these publications to the fraction of the fission product inventory that is released from the fuel (or molten corium) during the various phases of the accident progression. The next step, to account for subsequent deposition and mitigation of fission product releases before their ultimate release off-site, is sequence and design specific and not often documented in the open literature. However, this step can potentially have a much greater impact on the absolute magnitude of the offsite releases than the initial release fractions from the fuel (or molten corium).

The combined effect of these processes is typically calculated using integrated accident analysis codes e.g. MELCOR, MAAP, ASTEC which model the release and transport of various fission product groups. The grouping schemes are summarised in section 7.5. Use of such integral codes may be considered as the minimum requirement for estimating environmental releases in a modern PSA. However, there is a spectrum of approaches even within the integral codes, with some adopting simple “lumped parameter” models and others a more complex modelling approach. Even within a single integral code, both approaches may be used in different sub-models.

For specific issues, most commonly related to chemistry effects that are in general not explicitly modelled in the integral codes (except in the ASTEC code), additional analyses can be used to supplement the source term analysis. Recently, dedicated source term computer codes have been developed which model the source term phenomena more simply but have CPU time low enough to consider a much wider range of accident sequences (section 7.7).

The source term calculations carried out for the representative sequences are used to represent the entire set of APET end states allocated to the respective Release Category.

7.4 IDENTIFICATION OF KEY ISSUES FOR SOURCE TERM ASSESSMENT

In addition to the uncertainties in modelling severe accident phenomena which impact on the accident evolution, many of these physical and chemical processes influence fission product release, transport and retention. Furthermore, there are additional sources of uncertainty specific to the evaluation of environmental releases.

The following tables summarise the key parameters for source term assessment during the in-vessel and ex-vessel phases [266]:

Table 39 Key parameters for source term assessment during the in-vessel phases

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	Core/debris temperature Chemical reactions Time after scram Burn-up of fuel, and type of fuel (UO_2, MOX) Oxygen potential surrounding the fuel Refill/quench	Maximum burn-up of fuel Enrichment of fuel Amount of fuel Reactor power Zirconium mass Amount and composition of absorber material in control rod Amount of structural materials
RCS fission product transport	Recirculation flows Break flow or flow via primary relief valves Potential for containment bypass Suspension, resuspension, agglomeration, plate-out, revolatisation, etc. of fission products Fuel-coolant interaction Refill/quench Debris bed dryout/rewet RCS failure prior RPV failure via induced SGTR or RCS line failure Amount of water/steam	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

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 Kinetic chemistry in RCS	RCS design RPV design reactor cavity design Potential for ex-vessel scrubbing (e.g., suppression pool)

Table 40 Key parameters for source term assessment during the ex-vessel phases

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 composition • Potential for adding water on melt in cavity • Cavity geometry • Containment sump chemistry
Transport of fission products in containment	• Aerosol behaviour • Effects of hydrogen combustion on resuspension, deposition, revaporisation, agglomeration, de-agglomeration, etc., of aerosols • Chemical reactions in the gaseous phase and the liquid phase (including the effect of dose rate in gaseous phase, or the sump pH) • Interaction between gaseous species and surfaces	• Containment geometry and design • Active and passive ESFs • Composition of containment atmosphere (e.g. inertisation)
Pool scrubbing (BWRs only)	• Effects like: inertial impacting and deposition, diffusion 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	• Design of ESFs • Availability of EOPs and SAMGs

Many of these issues have been discussed previously in Chapters 4 to 6. However, several are of particular significance in calculating environmental releases and are summarised here:

Chemistry effects

During the heat up, degradation and meltdown (relocation and slump) phases of an accident, a substantial fraction of the volatile fission products will be released from the fuel. As volatile species, these are most likely to undergo chemical reactions which can influence release and transport, compared to less reactive species. The important fission products are iodine, caesium, tellurium, molybdenum, boron, silver. In very oxidising conditions or for high burn-up fuels, ruthenium may also be considered.

The integral codes, typically, have limited chemistry models (except for ASTEC where models are detailed as far as possible). For these elements, for some accident scenarios, additional codes can be used to supplement the source term analysis. This is discussed further in section 7.7.

Retention and deposition of fission products inside RCS

Following release from fuel, fission products are carried along with steam and hydrogen, both as vapour 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. Fission product aerosol can agglomerate to form large particles which can in turn settle on structural surfaces. Chemical interactions can occur between these vapours themselves and between vapour species and aerosols and diverse substrates. Decay heat could lead to heat up of structural surfaces beyond the re-vaporisation temperature of unbound fission products previously deposited. Hence, transport of unbound deposited particulate, in the form of “creep flow” can occur. Re-entrainment mechanisms may also cause re-volatilisation or resuspension.

The integral codes, typically (except for the ASTEC case), have limited revaporisation / resuspension models, due to the simplified chemistry assumptions described above. For some accident scenarios, additional codes can be used to supplement the source term analysis; however, this is not typically done unless a specific issue is being investigated.

Aerosol behaviour inside the containment

The phenomena affecting aerosol behaviour in the containment are complex and strongly sequence and plant (e.g number of compartments) related. The most important sequences when considering aerosol deposition in the containment are those in which the containment fails on a timescale of some hours after the release of fission products from the RCS. In sequences involving either very early or very late containment failure (or no failure at all), changes in the deposition rates of aerosols are not likely to significantly change the source term into the environment.

Deposition by gravitation is the most important retention mechanism of aerosols in the containment (in the absence of sprays), and particle growth is an important uncertainty affecting the environmental release estimate.

From the view of modelling aerosol deposition in the containment, the most important parameters are atmospheric relative humidity and temperature gradients at structure surfaces. For particles of inert materials the problem is fairly simple; however, a substantial part of the airborne radioactive material in a severe reactor accident is hygroscopic, or soluble, even in superheated conditions. Hygroscopicity leads to faster particle growth, and thus also faster deposition inside the containment, reducing the environmental release. Because of this, an accurate thermal-hydraulic and convective pattern model is essential for detailed analysis of aerosol behaviour. In a NPP containment, there are numerous compartments and rooms which maintain different thermal-hydraulic conditions. The conditions may vary even locally inside a large volume. These local differences between inter-connected volumes are very difficult to model with present integral codes.

Thus, the uncertainties in containment aerosol modelling can be traced not to the aerosol model package itself, but to its coupling to the thermal hydraulic calculation as well as user parameter choices, e.g. the nodalisation can have a significant influence on the final results. For some accident scenarios, specialist aerosol codes can be used to supplement the source term analysis; however, this is not typically done unless a specific issue is being investigated.

Energetic containment phenomena

In addition to the above, any energetic processes in the RCS or containment can impact the environmental release. Phenomena like hydrogen combustion or steam explosion influence the transport phenomena and the convective flow and, hence, the environmental release. For some accident scenarios, specialist codes can be used to supplement the source term analysis; however, this is not typically done unless a specific issue is being investigated. Simplified fast-running source term codes may help understanding the importance of this issue by sensitivity analyses.

Pool scrubbing and filtration in the release path

The role of engineered systems to mitigate radionuclide releases has been considered in Chapters 4, 5 and 6. However, the potential for radionuclide retention in all release paths should be considered. This may be particularly important for some bypass sequences where there are no engineered systems to mitigate releases but, by virtue of the accident conditions, the potential for scrubbing exists. This is discussed further in section 7.6.3.

Release path and failure mode effects

The timing and the way in which a containment fails is important to the environmental release calculation.

Early failure or bypass of the containment could result in a large release of fission products to the environment. Late containment failure, occurring more than a few hours after the start of core damage, allow removal mechanisms to greatly reduce the concentration of fission products in the containment atmosphere and, hence, the magnitude of the release. In addition, it reduces the activity of short half-life fission products, which have a high radiological impact.

For smaller release paths, or for leakage through an ‘intact’ containment boundary, the magnitude of the fission product release is determined by the size and location of the release paths and the pressure in the containment. Often the slow leakage through cracks in the containment will be efficient in removing aerosol components from the atmosphere. Most modern reactor designs have an additional structure around the primary containment boundary; this structure may be qualified as part of the containment boundary, as a secondary containment boundary or it may have a less defined role that is not credited in the safety case. Many PSAs, pessimistically, do not consider transport and retention of fission products in such structures. Additional discussion on issues to be considered to account for such effects is given in section 7.6.3.

7.5 GROUPING OF FISSION PRODUCTS IN SOURCE TERM CALCULATIONS

In terms of fission product release and transport behaviour, the integral severe accident analysis computer codes, discussed in section 7.6, perform calculations based on groups of fission products elements or chemical compounds rather than individual radioisotopes. This simplification is necessary to reduce the hundreds of potential radioactive isotopes to a reasonable number of groups that can be tracked in an integrated code, i.e. to achieve reduction in memory requirements and run time.

Different group structures may be used in different computer codes. Different group structures may also be used internally in the same code for different calculations. However, most grouping structures used for presentation of source terms results are based on similarities in the assumed physical and chemical properties of fission product elements. The group structure also accounts for similarities in the chemical affinity of the elements to reactions with other radio-elements and non-radioactive materials that they might encounter during transport within the reactor coolant circuit and containment – for example, steam, hydrogen and structural materials. Typical group structures used in the analysis of fission product release in integral codes are shown in Table 41 and Table 42.

Table 41 Default material classes in the MELCOR RadioNuclide (RN) package [274]

Group	Elements in Group	Representative Element
Noble Gases	He, Ne, Ar, Kr, Xe, Rn, H, N	Xe
Alkali Metals	Li, Na, K, Rb, Cs, Fr, Cu	Cs
Alkaline Earths	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm	Ba
Halogens	F, Cl, Br, I, At	I
Chalcogens	O, S, Se, Te, Po	Te
Platinoids	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni	Ru
Early Transition Elements	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W	Mo
Tetravalents	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C	Ce
Trivalents	Al, Sc, Y, La, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf	La
Uranium	U	U
More Volatile Main Group	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi	Cd
Less Volatile Main Group	Ga, Ge, In, Sn, Ag	Sn
Boron	B, Si, P	B
Water	H2O	H2O
Concrete	-	-

Cesium Iodide	Alkali Metals and Halogens	CsI

Table 42 Typical Group Structure for fission products in integrated codes (MAAP)

Group	Characteristic Species	Elements in Group
1	Noble Gases	Xe, Kr
2	CsI	I, Br
3	TeO_2	Te
4	SrO	Sr
5	MoO_2	Mo, Ru, Rh, Tc,
6	CsOH	Cs, Rb
7	BaO	Ba
8	$La2O_3$	La, Zr, Nd, Nb, Pr, Sm, Y, Cm, Am
9	CeO_2	Ce, Pu, Np
10	Sb	Sb
11	Te_2	Te
12	UO_2	U
13	Ag	Ag

The estimation of releases of radioactivity into the environment is typically obtained from the user defined containment leakage paths and models of the group behaviour within the containment. This aspect may be inherent in the integrated plant calculations or based on simple correlations driven by containment parameters taken from the integrated plant calculations. For most radionuclide groups this process is relatively straightforward, e.g. noble gases released from the fuel remain in the gas phase throughout and less volatile fission products remain as particulate aerosols; and do not undergo complex chemical interactions. However, the volatile / semi-volatile species (including the radiologically significant iodine, caesium, tellurium and ruthenium) can undergo significant physical or chemical changes within the containment. The modelling of these changes in the integrated codes is generally simplistic and can introduce a significant degree of uncertainty.

Numerous chemical interactions can occur, which cause elemental forms of these species to react and form compounds with a wide spectrum of physical properties. The different forms may then be acted on in different ways by the available safety features leading to uncertainties in interpreting the results of integrated plant calculations. For this reason, for a number of species it is not uncommon to use additional single effect codes to enhance the source term calculation for specific elements.

7.5.1 Summary for iodine

One of the most chemically complex species is iodine and the knowledge base of processes governing longer term iodine behaviour in the containment remains incomplete. Based on experimental and modelling studies, the weight of evidence indicates that iodine will predominantly be released into the containment as aerosol iodide ions (I- or as combined ionic species such as CsI). However, a fraction of the iodine released from the primary circuit may be in the form of molecular iodine (gaseous inorganic) as this is seen in the containment atmosphere at a very early stage in some experiments (e.g. Phebus experiments). In steam rich environments (especially out of the RCS at LOCA break where steam condenses), the prevalent CsI will readily dissolve forming aqueous iodide solutions and as such the iodine will be transported predominantly with the water, mostly into the containment sumps although some will be transported to surfaces and some may remain suspended in small water droplets in the form of a persistent aerosol.

In the presence of intense radiation fields, as would be expected in severe accident conditions, complex iodine chemistry can develop over time resulting in the formation of additional gaseous molecular iodine (I_2) via a number of routes, as well as stimulating other reactions with containment surfaces and aerosols that consume the gaseous iodine. Mass transfer processes will determine the rate at which the volatile iodine fraction, predominantly formed in the aqueous phases, will transfer into the gaseous phase. This, in turn, will partially determine the long term standing concentrations of iodine within the containment atmosphere.

A number of iodine retention mechanisms and near permanent 'sinks' have been identified.

- A fraction of the iodine transported to painted surfaces will be trapped, although this is not completely irreversible.
- In the aqueous phase radiolytic oxidation shows a strong pH dependency: in the pH range 5 to 8 an increase of one pH unit results in an increase of the potential to dissolve molecular iodine of about one order of magnitude. Thus, the maintenance of alkaline pH values in the water pools constitutes an important element of managing the iodine issue for many plants by holding up the iodine in a water pool.
- The scavenging of iodine in the containment sump by silver particulates, derived from control rod materials, is also widely accepted as essentially an irreversible iodine sink if sufficient silver surface area is present.

The reaction of deposited and dissolved iodine species with organic compounds present in the containment, such as paints and oils, is expected to eventually lead to the production of organic iodine compounds. This is an active research area and it is difficult to generalise about the magnitude of the organic component or the rate at which it is formed; however, its production is generally thought to be dominated by surface effects on wetted but non-submerged painted surfaces.

Another possible source for gaseous iodine is the conversion of CsI aerosol into gaseous iodine upon passage through an operating PAR. Experiments indicate that conversion rates in the order of several % [271]are possible when the recombiner is at high temperature. It is suggested in a PSA to determine the period when

PARs are operating at high temperature. During this time the CsI concentration in the atmosphere should be estimated, and the conversion according to experimental results should be applied.

The understanding and modelling of these complex chemical reactions improved considerably in the 1990s. However, the processes are still not fully understood and iodine chemistry experts are wary of making explicit recommendations for use in plant analyses. The PSA community generally acknowledges the iodine behaviour described above but the next step, to incorporate iodine chemistry into PSA methods, is not obvious. A review of current approaches was conducted within the SARNET project [272], starting from the assumption that severe accident source terms, both in-containment source terms and releases into the environment, presented in PSAs are based on the group release fractions predicted by the integrated accident analysis codes. The current different approaches adopted by the L2PSA community [272] can be summarised as follows:

- Do not address chemistry effects. This approach is justified for large releases where small additional contributions due to chemistry effects are insignificant. However, for intact containment scenarios and particularly for small filtered releases, the relative contribution of gaseous species can be decisive. Not addressing chemistry effects in these cases sometimes is because there is no requirement for source term information. However, this may also be attributed to the complexity of the issue and the proprietary nature of the approaches that have been developed, i.e. there is no obvious “off-the-shelf” methodology to adopt if one doesn’t already have one.
- Use global modifying factors based on interpretations of available chemistry data. These factors may be defined as an attempt at describing best-estimate behaviour or they may be defined with a bounding or conservative bias. In some cases qualitative rather than quantitative arguments have been used to justify that volatile iodine forms (inorganic and organic) have a relatively small impact on the overall PSA.
- The use of accepted “chemistry community” iodine codes (be it ASTEC/IODE, IMPAIR or INSPECT based) as an integral part of the PSA is not generally seen as a practical way ahead and several simplified models are being developed. It is not clear whether there is a consensus that these models are intended to be fully integrated into PSA methods or used as supporting justification / sensitivity analysis. Where simple or explicit iodine chemistry models are adopted, it is difficult to draw any general conclusions about their completeness and verification without additional information on the analyses, which may be proprietary. IRSN is currently seeking to address all the phenomenological issues.

In summary, chemistry models have not been widely adopted for PSA applications but some simplified models have been developed and are in use. It can also be argued that there remains considerable uncertainty in the thermal-hydraulic parameters needed to drive these chemistry models – particularly in the longer timeframe where the uncertainties in the accident progression of the integrated plant models also increase.

7.5.2 Summary for caesium

Since the number of moles of caesium in the core is typically 10 times more than that of iodine, usually most of the iodine combines with caesium and is released from the RCS as CsI. The rest of the caesium is released from the RCS as CsOH. CsI is thermodynamically stable up to at least 2000°C in typical severe accident conditions, but above 2000°C it will react with steam to form CsOH + HI. There is also strong experimental evidence that CsI reacts with boric acid (and with B_4C when present in the absorber rods) and, if significant reduction occurs in the coolant system, gaseous iodine and caesium borates will be produced. The caesium source term might be attenuated in the RCS by any reaction of both CsI and CsOH with boric acid, because these reactions produce less volatile caesium borates. In a similar way, Cs may interact with Mo, leading to less volatile caesium molybdate, but potentially increasing the gaseous iodine fraction. Additionally, CsOH also interacts with steel, diffusing into the inner chromium oxide layer, providing further potential attenuation.

7.5.3 Summary for ruthenium

In “The international nuclear event scale” [290] the radiological relevance of different radioactive isotopes is listed. The release of 1 Bq of Ruthenium-106 into the environment has approximately seven times more radiological effects than the release of 1 Bq of Iodine-131. In a typical LWR, the total activity of Ruthenium-106 is not very different from the total activity of Iodine-131. According to analyses with present-day integral codes not considering volatile Ruthenium species, the fraction of Ruthenium inventory which is released into the environment in a core melt accident is much smaller than the Iodine release. Therefore, Ruthenium releases traditionally have not been in the focus of analyses. However, more recent information about generation of volatile species, in particular due to reaction with oxygen from air, casts doubt on this judgement. Therefore, in a present-day L2PSA the issue of ruthenium volatility should be addressed. Attempts to model the Ruthenium release have been made in integral codes [291]. But it must be admitted that the state-of-the art for this issue is not settled, so that the PSA practitioner will probably have to apply appropriate uncertainty bands to the release fractions, based on scoping studies and/or on engineering judgement. A few considerations are given below in order to support such evaluations. It is recommended that PSAs refer to existing (e.g [292]) and probably upcoming documents in this field. For example, modelling improvement are in progress within ASTEC in relation with the OECD ISTP and STEM programmes.

Ruthenium may be considered separately from other refractory materials, due to its distinct oxidation processes, which may occur in the course of an accident and create species which are considerably more volatile than those currently modelled in the integrated codes.

The vapour species released from the fuel will generally condense in the cooler primary circuit conditions, either to form new aerosol particles (homogeneous nucleation), or on already existing aerosol particles (heterogeneous nucleation) or on surfaces such as pipes walls. At temperature close to those expected in the containment, all fission products are condensed with the noticeable exceptions of iodine and ruthenium. At low temperature the kinetics of chemical reactions are slow and, also due to limitations in mass transfer for heterogeneous reactions, the behaviour of transported material is then governed mainly by aerosol physics. During the condensation process, there are in most cases no chemical changes in speciation (congruent condensation) but this is not always the case as illustrated by the example of ruthenium oxides, with congruent condensation for the dioxide and non congruent condensation for the tetra-oxide:

$$RuO_2(g) \leftrightarrow RuO_2(c)$$

$$RuO_4\ (g) \leftrightarrow RuO_2\ (c) + O_2$$

The condensed species, once deposited on surfaces, maybe be re-vaporised in case of changes in temperature, in carrier gas composition (H_2O/H_2 ratio) or if their partial pressure in the gas phase decreases. Before being re-vaporised, the deposited species may have reacted with the substrate forming new and different compounds.

In certain accidents, air may be in contact with the fuel, submitting it and the transported fission products to a very oxidising environment. This is the case, for instance for a reactor severe accident after melt-through of the reactor pressure vessel, for a reactor accident in shutdown conditions with open RPV, for a spent fuel storage pool draining accident, or for fuel handling accidents. In such conditions, ruthenium is released from the fuel in the same manner as other volatile fission products such as cesium and iodine. Experiments on its transport under air performed by AEKI in Hungary and VTT in Finland show the presence of gaseous ruthenium tetroxide RuO_4 at temperatures typical of a cold leg break (about 450 K). RuO_4 is not thermodynamically stable at these temperatures and should normally be condensed as ruthenium dioxide RuO_2 (see above). The measured vapour pressures at low temperature in the AEKI first series of tests correspond to the equilibrium vapour pressure of RuO_4 over condensed RuO_2 at about 600-700°C. This means clearly that chemical reactions kinetics play a role in ruthenium transport under very oxidising conditions, i.e. in the presence of air, explaining the presence of a metastable species at low temperature.

7.5.4 Summary for tellurium

Tellurium is released from the fuel at about the same rate as noble gases, iodine and caesium. Empirical evidence from radionuclide release experiments suggests that tellurium has a chemical affinity for metallic zircaloy and its release occurs during the oxidation of the cladding, particularly near complete oxidation as occurs with the tin content of the cladding. Tellurium release into the containment will therefore be delayed, compared to iodine and caesium.

Within the current generation of integral codes, the sensitivity of the tellurium group environmental release fraction to the tellurium chemistry depends on the extent to which tellurium is assumed to react with the Zircaloy cladding and remain chemically bound in vessel. Experimental data indicate that there may be a significant in-vessel tellurium release, mainly after reaching the melting temperature of cladding. The chemical release form of tellurium also has an impact on the environmental release, since it affects the deposition and the removal of tellurium from the containment atmosphere.

Within the integral codes tellurium behaviour remains an issue of uncertainty. The magnitude of the in-vessel tellurium release depends on the release-correlation model chosen and on the specified sensitivity coefficients for the bounding to the zirconium. The CORSOR family of models for tellurium release, which are currently recommended for use, are based on an experimental correlation developed at ORNL. If it is assumed that tellurium is totally bound to the unoxidised zircaloy during the in-vessel phase, very little tellurium is released into the containment atmosphere until MCCI begins in the reactor cavity. In the other extreme, when tellurium is allowed to be released in-vessel, it is rapidly oxidised to TeO_2 which is readily deposited on primary circuit surfaces. Thus, the release of tellurium from the primary circuit is reduced, there is almost complete removal of gaseous Te_2 from the containment atmosphere and the release to the environment is reduced substantially, by about two orders of magnitude.

7.5.5 Examples of grouping schemes

7.5.5.1 <u>L2PSA for French 900 and 1300 MWe PWRs</u>

In the French PWR 900 MWe L2PSA by IRSN [259], the following fission product groups were used in the atmospheric source term assessment: (Note that the aerosol group consists of very different species [e.g. Cs and Pu]. These will behave in a similar way if they are in the form of aerosols, but – of course – their release from the fuel is very different)

Group	Elements in Group	Representative Element in Group
Noble gases	Xe, Kr	Xe
Iodine gas	I (I_2 form)	I
Organic iodine	I (ICH_3 form)	I
Aerosols	I (aerosol form), Cs, Te, Sr, Y, Zr, Nb, Mo, Tc, Ru, Re, Sb, Ba, La, Ce, Np, Pu	Cs

In the recent PWR 1300 MWe L2PSA (2010), several changes were made:

- Bromine (Br) introduced and combined with iodine (I) to form the halogen groups, due to similar chemical behaviours.
- Ruthenium (Ru) removed from aerosols group, due to its alleged importance. Several chemical forms are considered for this element (e.g Ru04, Ru02), which is now seen as important for the accident consequences assessment
- Aerosols group divided into two groups, one volatile and another with semi-volatile aerosols, according to the kinetics of release from the core.

Hence, the following fission product groups were used for the source term assessment:

- Noble Gases (Xe, Kr).
- Halogen gases (I_2, Br_2).
- Organic halogens (ICH_3, $BrCH_3$).
- Halogen aerosols (I, Br).
- Ruthenium gas (RuO_4).

- Solid ruthenium (RuO_2).
- Ruthenium aerosol (Ru).
- Volatile aerosols.
- Semi-volatile aerosols.

Reference

[259] N. Rahni, E. Raimond, K. Chevalier-Jabet and T. Durin, L'EPS de niveau 2 pour les réacteurs REP de 900 MWE - Du développement aux enseignements de l'étude, IRSN, Rapport Scientifique et Technique 2008.

7.5.5.2 Sizewell B L2PSA

In the Sizewell B L2PSA [267] the deterministic analysis of the characteristic sequences was initially performed using the MAAP3B severe accident integrated code. The grouping scheme and fission product release recommendations in this version of the code were different in several respects from the current grouping scheme of Table 42. The chemistry, thermal-hydraulic conditions and fission product release and transport mechanisms within MAAP3B were reviewed and several non-default assumptions used. These were primarily in respect of:

Group 2: The only iodine species considered was CsI and there was no mechanism to evaluate the production of more volatile gaseous forms of iodine. The production of volatile inorganic iodine species in the sump water is inhibited in the Sizewell B design by providing buffer chemicals to maintain the pH at a value greater than 8. Control of the formation of these species restricts the potential longer term production mechanisms for inorganic iodine. However, an additional allowance was made for this in the source term evaluation of 0.2 % of the Group 1 release fraction.

Groups 3 and 11: Currently the recommendation is for use of the CORSOR-O / CORSOR-M release correlations. These correlations were not developed at the time of the initial analysis and a Cubicciotti based correlation was used. It was further assumed that the tellurium was completely bound to the unoxidised zircalloy; which effectively prevented release of Group 3 until all the cladding had oxidised and, as a consequence, maximised the release of Group 11 in the ex-vessel phase during MCCI.

Minor modifications were also made to the grouping scheme of MAAP3B, as a result of the internal review on fission product release mechanisms in the light of potential chemistry effects. This effectively increased release fractions of several key elements, including ruthenium and plutonium, which were important in calculating the off site consequences. Similar changes have since been adopted as the standard grouping scheme in MAAP4.

7.6 SOURCE TERM ASSESSMENT BY INTEGRAL CODES

Source term calculations for a selected number of accident sequences should be performed using integrated and/or mechanistic codes. Typically, the accident progression / phenomenology for these sequences are described in detail in the L2PSA documentation. The results of these accident sequence source term calculations using integrated codes may be used in several ways:

- directly to represent the source terms for all accident sequences which have been grouped into the same release category . This is the conventional approach to source term assessment;
- as the basis for more detailed quantification of source term parameters and their uncertainties, e.g. in more mechanistic single effect codes;
- to quantify parameters and uncertainties to be input to fast running source terms codes or dedicated source terms models used to calculate very large numbers of accident sequence source terms (see Section 7.7).

7.6.1 Introduction to integral codes

Since the NUREG-1150 study [262] significant progress has been made in the integrated severe accident analysis codes to model fission product release and transport behaviour. Two specific codes are widely used in the current generation of L2PSA – MAAP (modular accident analysis program) and MELCOR [274]. Both codes have undergone significant validation (based on both integral and separate effect experiments) and benchmarking exercises. The ASTEC European accident evaluation code system, largely used for IRSN L2PSAs, is also an upcoming code for L2PSA analysis.

The application of integral codes for source term assessment should be validated to provide confidence in the results being produced. The users of an integral code should be: experienced in the use of the code; familiar with the phenomena being modelled by the code and the way that they interact; the meaning of the input and output data; and the limitations of the code. Since the phenomena in integral codes is broad, starting from thermal hydraulics through fission product chemistry to aerosol physics, it is very useful for the code user to consult specialists in the individual phenomena. If this is not possible in his/her organisation/country, it is recommendable to look for a specialist from different organisation/country.

The present integral codes are quite detailed or they allow using detailed models defined by the input data. This may be misleading for the user who may feel that the predictions are precise. In reality when using a detailed model in a validated code, the uncertainties are still very high:

- Simplifying assumptions have to be made in some areas – fuel degradation, fission product speciation, aerosol physics
- Even some lack of knowledge exists in the areas named above – eutectic reactions in the core, fission product speciation especially in the primary system, some aerosol phenomena other than agglomeration and sedimentation
- The code validation is limited to some phenomena, it is useful to check the published validation matrices before deciding on the level of validation for given purposes
- There may be some special assumptions in the code or the assumptions change with the code version occurs which may be overlooked by the user

When using integral codes the radionuclide release and transport information is typically tracked in terms of the fraction of the initial core mass inventory of one or more of the groups of radioactive elements or chemical compounds rather than individual radioisotopes. This simplification is necessary to reduce the number of species being tracked to a manageable number. However, some current versions of the integral codes can be used to track a number of important radionuclides as well. For example, MAAP4.07 can be run with either the element input option or the element-and-nuclide input option. For the element input option, core fission product mass inventory alone is tracked. For the element-and-nuclide option, additionally 65 radiologically significant nuclides can be tracked. In a same way, ASTEC is able to track more than 1000 nuclides, along with the possibility to model the chemical reactions between them. This additional information is useful if the L2PSA objectives are expressed in terms of a quantitative activity release or an off site dose target.

It is generally acknowledged that a significant level of uncertainty still exists in the prediction of offsite releases due to the issues discussed in previous sections and this should be addressed in uncertainty analysis / sensitivity analysis carried out as part of the L2PSA.

7.6.2 The conventional approach to source term assessment

Conventional L2PSA tends to use the integral codes for providing the baseline source term assessment of the Release Categories. Appropriate analysis should be carried out to provide confidence that the source term associated with each Release Category is adequately characterised. Additionally, such analyses should support the assessment of uncertainties in source terms for a given release category.

In principle, this means that a basic sequence for each type of release category should be selected and a source term assessment performed, e.g. for a Small LOCA with design leakage from the reactor building from the onset and late containment failure. Following this, enough variations could be performed in principle to provide information on the range of possible source terms which could be expected for the same sequence within the scope of possible run parameter variations. Repeating this for all sequences belonging to the release category, and enough variations to provide results for all the different containment failure modes that the release category could present in the course of the accident progression (e.g., a repetition of the analysis assuming containment failure at a penetration, one with failure assumed to occur when hydrogen concentration in the containment is at its peak, etc.) would be desirable, but not normally achievable with present day resources.

The number of integral code assessments indicated by this approach may become prohibitive. In practice, if the Release Category contains very similar sequences and the conditions are relatively stable, a small number of analyses may be acceptable. Typically, more analysis may be required for Release Categories defined by energetic or uncertain phenomena, e.g. if the source term for a given Release Category is particularly sensitive to a design feature of the plant or a specific transport mechanism for radioactive material. Pessimistic assumptions for the source term may be justified to address this issue to avoid undue resource needs.

It may also be possible, within certain limits and with much caution, to use analyses from reference studies or another plant where the design features are sufficiently alike [270]. This may be a viable approach for l2PSAs whose objective is limited to demonstration of LERF/LRF, but the results and assumptions must be carefully qualified.

7.6.3 Additional issues for predicting releases to the environment

The environmental releases, associated with accident scenarios, are usually calculated in the integral codes using user defined release path parameters. The most obvious being an equivalent leak size for containment failure sequences or a vent pathway size for vented containment sequences. It is not straightforward to extrapolate such parameters to cope with leakage through very small release pathways as would be expected in an intact containment boundary; however, it is common practice to use an equivalent leak size approach even for very small leak paths. This issue has been discussed at length in section 6.

Most modern reactor designs have an additional structure around some or all of the primary containment boundary. Release pathways from an intact primary containment will, in most cases, first enter the surrounding structure before they reach the environment. This structure may be qualified as part of the containment boundary, or as a secondary containment boundary or it may have a less defined role that is not credited in the safety case. Depending on the design, this structure may have a number of engineered safety features that would mitigate the environmental release; e.g. qualified ventilation systems with particulate or iodine filters, sprays of fire extinguishing systems, pressure tight doors, etc. The type and leaktightness of the building structure and the installed engineering safety features defines the retention potential in the structure. Many l2PSAs, pessimistically, do not consider transport and retention of fission products in such structures; but a realistic source term assessment should take these issues into account where they are significant. The associated physical and chemical processes are similar to the events inside the containment; hence, standard integral severe accident analysis codes (MELCOR, ASTEC, MAAP) provide features which could be used for modelling such structures. However, the effort for building an adequate input deck could be substantial and it may be more practical to consider the global impact of such effects on the source term prediction, rather than try to build a detailed model of partly speculative release pathways.

The total influence of such factors may be up to several orders of magnitude for some fission product groups. Additionally, groups are not affected in the same way, e.g. noble gases may be delayed, diluted or retained in dead-ended spaces, or may be released at a higher elevation because of ventilation systems, while other groups undergo other retention mechanisms, such as deposition. Such effects could be taken into account when estimating source terms. If this is not done explicitly, L2PSA overestimates releases in general. In this case a degree of uncertainty and conservatism is introduced into the source term prediction.

When considering additional possible retention mechanisms during transit along the release path out of the containment, the following aspects should be considered

- deposition of aerosols in release paths (particularly if these are long and tortuous with low flow rates);
- dilution in larger gas/air volumes in surrounding structures (e. g. turbine building in a BWR) may result in a lower release rate to the environment (though with a pro rate extension of the release duration);

- influence of ventilation systems (these may not be explicitly qualified for severe accident conditions but, if thermal and flow conditions are not extreme, there may be some mitigation);
- filtration (particularly important for secondary containment);
- scrubbing in water pools (may also be considered within containment);
- flows to adjacent spaces, etc may result in multiple release paths;
- natural atmospheric draft may direct releases through the stack even if exhaust fans are not active.

The issues associated with predicting releases to the environment for bypass pathways and two complex release pathways (intact containment and basemat melt through) are discussed below.

7.6.3.1 Release in containment bypass sequences

Containment bypass is often the dominating cause of large early releases in the results of L2PSA studies. It is very difficult to find credible mitigative mechanisms for these sequences, since the containment function is lost immediately. This issue was considered in the Source Term Uncertainties project [268]. The identified mitigative mechanisms (retention of fission products by deposition on pipework, pool scrubbing and potential variations in the release route), therefore, are potentially very important to take into account when striving to remove excessive conservatism from the PSA results.

The bypass sequence plant damage state definition (the sequence information input to the level 2) usually contains information on, for example, what systems are involved in an interfacing system LOCA. Thus it may be possible to fairly realistically determine the pipe geometry and thermohydraulic flow conditions, which serve as input information for estimation of the retention factor. Therefore, it is quite feasible and justified to give credit to this retention mechanism when modelling a bypass sequence in the PSA. If the pipe geometry is such that conditions are favourable to retention of fission products, even quite a short flow release path will be sufficient to ensure very efficient deposition. It is clear that the issue of subsequent resuspension of fission products also needs to be addressed, if the pipe deposition is significant.

In some bypass cases, locations where water pools will be formed may also be readily identified. In such cases pool scrubbing would serve as an important mitigative mechanism, e.g. with a leakage into rooms at a low elevation. Then the scrubbing mechanism should be given credit for reducing the source term. However, when there is the potential to have the release almost anywhere in the auxiliary building (e.g. from any location in an interfacing system), it is quite difficult to ensure that the fission product release location will indeed be submerged in water. Giving credit to this mitigative mechanism in a real PSA application for bypass sequences may, therefore, not be quite straightforward.

7.6.3.2 Release through an intact containment

In most designs a containment design leak rate is specified. This leak rate is normally related to a design basis accident, and not to a severe accident with core melt. Therefore, even if the containment remains “intact” in an accident sequence, it has to be checked whether the design leak rate is applicable.

Even if the actual leak rate is increased in a severe accident, an intact containment will provide significant protection against large releases. Therefore, if the scope of PSA is limited to large or large early releases, a simplified analysis may be admissible to show that the source term from an intact containment is below the "large" release.

If the PSA aims at producing realistic source terms for the complete set of accident sequences, the release from an intact containment has to be analysed in more detail. The release of fission products into the environment is significantly affected by the release pathway and multiple release pathways (e. g. at containment penetrations) may be developed for some accident scenarios with an intact containment. These release pathways are highly plant and scenario dependent and they have to be studied separately for each plant. Features representing particular sensitivity are the containment type, the engineered safety features and the severe accident loadings and their timing.

When the release path effects are introduced to the L2PSA studies, the difference between phenomenological uncertainty (e. g. deposition mechanisms in tortuous pathways) and system-related issues (e. g. leak rates of penetrations) should be kept in mind.

7.6.3.3 Releases in basemat failure sequences

The release of fission products to the atmosphere in case of basemat melt-through or basemat penetration has two components:

- The potential atmospheric release path, taking into account all release paths to the air. This release path has a similar timeframe to the accident timeframe.
- The potential release to the ground, transfer into the groundwater and subsequent transport to surface waters. This release path may be significantly delayed compared to the accident timeframe.

Only the first path, the atmospheric path, can be directly assessed in the same way as other release paths leading to environmental releases, and should, to some extent, be considered in a PSA. A key issue is the containment pressure when basemat failure occurs. If containment venting is foreseen as a mitigation measure, low containment pressure is very likely.

For reactor designs where no compartments are below the primary containment bottom the atmospheric path should not result in a large release, for two reasons:

- the release occurs at a rather late time after significant progress of the MCCI. At that time aerosol concentration within the containment is expected to be quite low;
- the atmospheric release path occurs after migration through a system of long paths through the underground with significant depletion potential.

For reactor designs where compartments exist below the primary containment bottom, the atmospheric path could result in a large release, because the bottom between the primary containment and the underlying rooms may not be very thick, leading to less depletion in the atmosphere before failure, and because the secondary containment may not be able to retain much activity, depending on the design.

The second release path, via ground water transport, is more speculative and cannot be compared quantitatively with the atmospheric path for two reasons:

- The final release to surface water or drinking water occurs very late, maybe years after the accident, but the situation may differ considerably from one site to the other,

– Because of this significant retardation, and because it is not a direct release to the atmosphere, counteractive measures to reduce human exposure are feasible. The potential long term impact of such releases on the environment and non-human species should not be dismissed, but it is not considered within the scope of conventional L2PSA.

Two processes are relevant for the prediction of source terms via ground-water transport:

– The transport of fission products from the melt to the ground water. The release of (low volatile) fission products from the melt may occur by leaching when water gets into contact with the melt. However, this effect is significantly limited by the heat-pipe effect: in case the surface of the melt is hot, water will evaporate before touching the melt, in case the surface is cold, leaching rates are very low.

– The transport within the groundwater. The transport within the ground water is usually very slow. Groundwater typically flows with a velocity of around 1 m per day. However, the effective transport velocity of fission products in ground water is smaller by many orders of magnitude, due to ion-exchange processes in the soil (note that many of the low-volatile fission products are alkaline earth metals being in abundance in the soil). Finally, the filtration and dispersion within the soil has to be taken into account. Thanks to some research in the frame of underground nuclear power plants, methods exist to estimate upper bounds for the transport of relevant elements with groundwater, provided that the composition of the soil is known.

For these reasons it is usually recommended to not consider quantitatively the source term from the ground water transport. It can be sufficient to provide the frequency of accident that may lead to ground contamination after basemat penetration.

7.6.3.4 Potential impact of severe accident management actions

Severe Accident Management (SAM) strategies with the potential to terminate or mitigate severe accidents are at various stages of development and implementation at NPPs within the European Union. The European Commission sponsored the OPTSAM study [277] to evaluate the impact of certain accident management strategies on the radionuclide behaviour. In total, 24 accident sequences covering a range of potential reactor faults were selected to provide the basis for over 130 detailed plant calculations performed using integral codes. Overall, it was concluded that no significant adverse influences on the in-containment fission product behaviour, as a result of implementation of SAM measures, were seen in the case studies examined. The following SAM objectives were considered:

- Protecting the integrity of the RCS, including the RPV:
  - Precautionary RCS depressurisation.
  - Extending the timeframe of RPV integrity, e.g. by measures such as recovery of safety injection systems and ex-vessel cooling of the RPV wall.
- Protecting the integrity of the containment:
  - Containment venting strategy, e.g. optimisation of filtered venting strategy to minimise the environmental releases (also to prevent loss of core cooling in BWRs).

- Containment spray strategy, e.g. recovery of (or use of alternative) water supplies for pressure control, restoring condensation pool cooling, achieving ex-vessel flooding, removal of fission products from the atmosphere.
- Hydrogen management strategy, e.g. use of passive autocatalytic recombiners (PARs), or combined use of PARs together with the containment spray system.
- Long term decay heat removal from the containment atmosphere.

7.6.4 Example discussion on uncertainties of source term assessment with integral code

GRS has performed several L2PSA, where source terms have been calculated with MELCOR 1.8.6. The following issues have been discussed in order to estimate the uncertainty of the MELCOR source terms:

Resuspension of aerosols

If hydrogen combustion leads to containment damage either by a missile or by effective pressure it is assumed that the release from the containment to the reactor building derived from MELCOR results would not sufficiently consider resuspension of aerosols. Compared to results without resuspension, the release can be a factor 1 - 2 higher according to a GRS report on a L2PSA for a German NPP.

Aerosol settling

In a PSA for a German BWR the analysis of aerosol release by MELCOR has been discussed. It turned out that the aerosol settling process in wet atmosphere is probably underestimated by MELCOR, which leads to an overestimation of the source term. Within that PSA a factor of 0.1 - 1.0 has been multiplied to MELCOR aerosol releases in order to consider this issue. It is unclear whether these analyses are also applicable for more recent MELCOR versions. The internal MELCOR modelling may have evolved since then.

Gaseous iodine

During a severe accident, a certain amount of iodine is expected to be released from the RCS or to be generated in the containment in gaseous forms like elementary I_2 or organic methyl iodide. The MELCOR modelling assumes that iodine radionuclides are always released as CsI aerosols from the core or the melt in the cavity and no conversion to gaseous forms is modelled. There is an iodine pool model existing in MELCOR, which however is not recommended for routine application. Therefore, engineering judgment was used as explained below to treat the subject of gaseous iodine in an acceptable way.

There is a general consensus [294] that a maximum of 5 % of the iodine inventory in the core could be released into the containment directly as I2. In agreement with this, the revised accident source term for US plants [295] assumes that at least 95% of the iodine reaching the containment is in aerosol form. Note that this applies for PWR fuel elements with SIC control rods whereas for BWR fuel elements with B_4C control rods I_2 releases of up to 80 % of the iodine inventory have been observed in the Phebus FPT3 experiment. Because no more specific estimate is available it has been assumed that between 0 % and 5 % (uniform uncertainty distribution) of the CsI aerosol reaching the containment actually occurs as gaseous iodine and can be further released as such, e.g. after containment failure.

In reality, a more complicated behaviour is expected in the gas phase (in the containment). Many experiments have been performed in the last more than 10 years f. i. at the Phebus FPT facility in France, at the THAI facility in Germany and as well in other research programs. A detailed re-evaluation of all these experiments is not yet available. Therefore, some main findings are discussed below.

Painted surfaces (f. i. all walls inside the containment and the annulus) act as a sink for I2 but also as a source for volatile organic iodine (depending on the radiation level). I2 can also react with air radiolysis products like ozone to form IOx aerosols, which we count here as gaseous iodine as well. In PHEBUS experiments [296], [297] the airborne concentrations of these three species (elementary iodine, organic iodine, IOx) were observed to have similar orders of magnitude (about 10^{-9} mol/l). The dose rate in the gas phase was about 0.3 Gy/s, the ratio between painted surfaces and containment volume about 0.33 1/m. If similar concentrations like in the PHEBUS experiments were assumed in the containment this could correspond to about 1 % of the iodine core inventory occurring as gaseous iodine in the containment.

Another possible source for gaseous iodine is the conversion of CsI aerosol into gaseous iodine upon passage through an operating PAR (see section 7.5.1). Recent THAI experiments indicate that conversion rates up to 3 % are possible under specific conditions. The following approach was used in the GRS PSA to obtain a rough estimate of the total amount of gaseous iodine thus produced. MELCOR calculates the hydrogen recombination rate for each PAR. The recombination efficiency is at least 50 %. Therefore, twice the hydrogen recombination rate should approximate the rate of hydrogen entering a PAR. Dividing this rate by the amount of hydrogen present in the corresponding control volume (also available as MELCOR result), an estimate of the volume fraction flowing through a PAR per time unit is possible. This result has to be multiplied by the mass of airborne (i.e. not yet deposited) CsI aerosol particles in the control volume in order to calculate the rate of airborne CsI aerosol particles entering the PAR. From MELCOR output files it could be determined that typically a fraction of about 5 % of all airborne aerosol particles are CsI particles. Using this value and assuming a conversion rate into gaseous iodine of 2 % , evaluations with two MELCOR calculations showed consistently that at the end of an accident about 2.5 % of the CsI aerosol reaching the containment could be converted into gaseous iodine by PAR operation. This estimate neglects that conversion occurs only for high PAR temperatures linked to high hydrogen concentrations at the PAR entrance (> 8 vol.%) and may also overestimate the flow through the PARs. In practice only a few PARs may see for a short period in time such high hydrogen loads, allowing an efficient CsI conversion. So this estimate seems to be conservative in a way of overestimating the amount of gaseous iodine.

Finally, iodine in gaseous form can be produced due to radiolysis in water pools (where according to MELCOR a large fraction of all released CsI aerosol ends up in the course of an accident). The various experimental and modelling aspects of iodine chemistry are very complex (cf. e.g. a recent state-of-the-art report [300] and it is very difficult to estimate the resulting iodine source term into the environment. A main uncertainty comes e.g. from the unknown pH value in water pools like the containment sump (a possible accident management procedure leading to an alkaline pH value > 8 could help to reduce the production of gaseous iodine). The treatment of these possible sources of gaseous iodine has been documented in a L2PSA by GRS for a German BWR (69 series) [293]. From that analysis an effective production rate of gaseous iodine from water pools in the

containment of the order of 0.01 % per hour (in terms of the iodine core inventory) can be derived although this estimate has to be associated with a very large uncertainty. In the event tree this rate is given a uniform uncertainty distribution between 0 % (e.g. in case of an alkaline pH value in the sump) and 0.02 % per hour.
The further release and transport of all gaseous iodine in the event tree is assumed to be similar to noble gases except for the retention in filters.
The estimate of releases of gaseous iodine presented above is rather crude given the complexity of issues to be considered and the large uncertainties in quantifying them. Therefore, the corresponding results should be considered more like an estimate of the right order of magnitude to obtain an idea of the relative importance of gaseous iodine releases compared to other source term contributions. Also, it is assumed to be conservative in the sense of overestimating the releases into the environment.

7.7 SOURCE TERM ASSESSMENT BY DEDICATED (FAST-RUNNING) SOURCE TERM MODELS

The approach described in this section is not recommended for PSAs with limited budget and is not necessary if the objectives are limited.

7.7.1 Introduction to fast running source term models

As mentioned in section 7.4, key parameters for calculating environmental releases are both physical and scenario based.

Firstly, even if, assuming that all physical phenomena are well known, the number of key parameters and the variation range of some of them may be quite large. As an example, their impact on the evolution of the accident scenario with time is considered for 3 situations:

- Core degradation may start right after scram (for a reactor at nominal power) or after a few days (e.g. 3 days in case of accident at shutdown); in consequence the specific activity for equivalent iodine masses that may be released can differ by a factor 2.
- Containment failure could occur during core degradation (for instance due to hydrogen combustion), or later (for instance due to containment pressurisation following corium concrete interaction). Thus, an identical containment leak cross-section could lead to a wide range of potential releases to the environment, due to the different amounts of airborne radionuclides in the containment atmosphere.
- Depending on the NPP, some systems providing capture mechanisms for radioactive material, like containment spray system, may have a large impact on the magnitude of radioactive releases. In order to have meaningful results, the system operation must be described with a reasonable precision along the accident, and at least during each main phase of the accident.

Multiplying all the possibilities of all the key parameters, one can see that the number of possible release sequences can be large, up to several thousands for a given PSA 2. Moreover, for a given release sequence, uncertainties about physical phenomena exist: for example, the occurrence of an ex-vessel steam explosion could lead to containment failure but, the precise consequences of the explosion on the containment integrity cannot be fully calculated and should be associated with a distribution of possible leak sizes (less than 1 cm^2, to few tens of cm^2, etc). In this case, the order of magnitude of radiological consequences may vary by a factor of 10 or more.

One could perform sensitivity analyses to approximate the lack of knowledge of each single phenomenon. But sensitivity analysis becomes more complex when dealing with multiple phenomena, each one potentially balancing the other one. For a given release sequence, the number of potential release calculations may become very high if a sufficient sample of the possible values of physical parameter is taken into account.

In other words, considering the number of different release scenarios and the existing uncertainties, a large number of calculations may be needed and it is rather useful to develop fast running source term models.

7.7.2 Global principles of dedicated (fast running) source term models

Simple source term models can be used to sort out the relative severity (in terms of magnitude of release and radiological consequences) of each situation, to give priority to sequences which are most challenging. Detailed kinetics results are less important for simple source term models providing results at a small number of predefined time frames. The source term model, for this kind of use, may be as simple as analytical functions or a neural network system, taking into account the parameters governing releases. For example of simplification in comparison with integral codes, there is no need to model precisely the in-vessel core degradation but only the fission production emission during this phase.

Furthermore, this kind of evaluation has the advantage of providing useful information to gather release sequences into wider categories, which can be easier to present.

One could aim to have a greater understanding of the impact of physical phenomena, or to retrieve the full kinetic data of the accident to get relevant information for emergency planning: for example, the time when emergency levels are reached. For these reasons, a more detailed model may be built, based on balance equations. This kind of model also needs to remain rather simple to retain good calculation performances:

- The physical modelling of the plant has to be rather simple. Each relevant part of the buildings of the plant, if their modelling is required, may be described as surfaces, so that the equations used have no space dimension; for each modelled compartment, however, the exchanges between the parts of the buildings have to be taken into account, one being potentially a source of activity for the other. For example, for a PWR, having one single containment volume, the containment building may be described as 3 interconnected parts: the atmosphere of the containment, the walls, and the sump. In the same way, the interconnections between the buildings have to be taken into account.

- The physical/chemical behaviour of fission products has to be modelled in a very simple way, and the phenomena may be modelled as independently as possible one from another. As an illustration, radioactive decay and chemical behaviour may be processed independently.

    - Radioactive decay / specific activities of radionuclide groups: this must be taken into account, because they may vary quite substantially along accident progression. If one chooses to take into account a very limited number of isotopes, radioactive decay may be computed in a classical way, provided that all the relevant decay chains of the selected isotopes are fully modelled. For large numbers of isotopes though, the number of calculations can become too large, and radioactive decay should be pre-calculated using specialised computer codes.

  - Release of the different species from the core can be modelled as a linear constant source term, the release intervals being specie-dependent.
  - Retention in the RCS may be modelled roughly with some coefficients.
  - Aerosols modelling: aerosol behaviour depends on the thermo-hydraulic conditions in the containment atmosphere, the space configuration, and their size distribution, and whether sprays are working or not. In a fast running code, the required information is the global quantity of aerosols present in the atmosphere, the walls, and the sump. Therefore, by performing evaluations with computer codes (ASTEC for example), for a limited set of situations, one can deduce simple correlations for global deposition of aerosols.
  - Transport to other buildings or to the environment: it can be modelled through a simple leak flow rate, being possibly a function of containment pressure.
  - Filtration systems: they may be modelled rather simply by using a simple retention factor.

- As we saw previously, the time frame of the accident can be important. The species amounts in each modelled part of the containment should be computed as a function of time. The different physical/chemical conditions (pressure, pH, T, dose rate…) may strongly influence the behaviour of the species. These conditions may evolve along accident progression. But, as a first approximation, some of them may be considered as constants during reasonably short time frames, that we may call sub-phases. The larger the number of sub-phases, the larger the number of evaluations of the physical/chemical conditions.

7.7.3 Validation of dedicated (fast running) source term models

The different physical / chemical models should be elaborated from reference data obtained through R&D activities but a simple way to validate the fast running source term model is to compare the results obtained on a representative set of transients with integral code calculations (ASTEC, MELCOR, MAAP or ECART).

The description of models, the link with R&D results and these comparisons with integral codes should be documented in the L2PSA.

7.7.4 Examples of fast source term calculations

7.7.4.1 The neural simulation of source term potential at ERSE

An innovative method for the fast running evaluation of the source term (named ECART) was proposed by ERSE in cooperation with the Politecnico di Milano developing a neural simulation model [278].

This method is aimed at evaluating quickly the airborne concentrations of representative radioactive element groups in the containment atmosphere, to calculate the environmental releases as a function of the containment leakage rate, accounting for all the aerosol and gases transport processes according to the state-of-the-art.

In this example, the radionuclides release into the containment is predicted for up to 72 hours following core-melt, and the thermo-hydraulic main variables are assumed to be known or are deduced through some simplifying hypotheses. In case of containment failure, the airborne concentrations are the starting point for the analysis of the environmental releases.

The example of severe accident scenario provided is representative of a small LOCA in a large PWR. The typical release pattern considered for the case study is a simplified one, featuring two distinct release phases to the containment: an in-vessel phase and an ex-vessel phase with possible MCCI. For the in-vessel phase, the case study simplification is based on the typical release trend shown in Fig. 57; for the ex-vessel phase, a constant release is assumed.

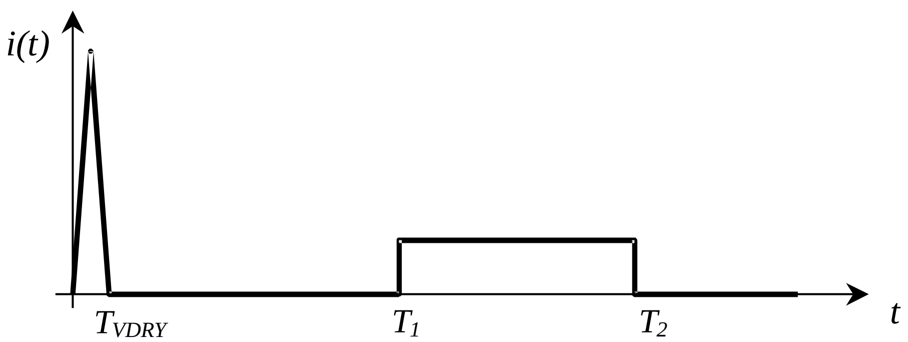

Fig. 57 Typical release pattern to the containment considered for the case study.

Each scenario is characterised by a different value of the following variables:

- - Accident times [s]:
  - START: the fuel begins to be uncovered by water;
  - T_{VDRY}: the vessel is completely dry;
  - T_1: the interaction between molten-core and concrete begins;
  - T_2: the interaction between molten-core and concrete ends.
- Containment atmosphere temperature at 6 different times (0h, TVDRY, 12h, 24h, 48h, 72h); an upper limit of 410 K can be assumed to avoid a containment failure due to the air and steam pressure.

- Release from primary loop factor R: depending on the accident scenario, a fraction (1-R) of the in-vessel release is retained in the primary loop and does not flow into the containment (typically 0.1 <R<0.8).

The radioactive elements accounted for in the analysis, are according to the WASH-1400 grouping: Iodine, Caesium, Tellurium, Ruthenium, Strontium and Lanthanum, in their molecular forms CsI, CsOH, Te_2, SrO, RuO_2, La_2O_3. According to this approximation, it is assumed that all Iodine reacts with Caesium, and the remaining Caesium forms CsOH. The first four radioactive elements appear as the most important for the thermal-hydraulic behaviour of the atmosphere and structures involved in the fission product transport: iodine, in particular, represents a high fraction of the decay power and, at the same time, is likely to be almost completely released. The latter elements have a lower potential of release from the degraded core because of their lower volatility, but are important from the radiological point of view.

Following the typical release pattern of Fig. 57, once the accident times (TVDRY, T1, T2) that characterise each scenario are known, a release mass flow function specific to each element is written.

Concerning the ex-vessel phase, in the case of some elements, the release can stop before T2. In fact, whereas T2 is the end time of the corium-concrete interaction, generally due to the intervention of some safety system, the ex-vessel release of an element can also stop because of its mass exhaustion.

Then, the physical processes of aerosol and gas transport and deposition are analysed taking into account mean containment atmospheric conditions, thus obtaining the containment concentrations of radionuclides as output variables.

An automatic procedure has been designed to run a high number of simulations with ECART. The automatic procedure performs an arbitrary number of simulations, of the order of several thousands, randomly extracting the input variables for each simulation from a range defined by the user [279].

Each ECART output file is read and an appropriate file, containing the case study system inputs and outputs, is prepared. This file represents the training field for one or more neural networks. It must be remarked that the artificial neural network is a non-linear interpolator that can be progressively adjusted (trained). It permits very fast-running calculations on the basis of a proper training on experimental or theoretical databases. It does not take care either of the physical phenomena or their complexity.

The example in Fig. 58 is a demonstration case of a-containment source term, where the calculations of each pattern are based on very simple assumptions. It is evident that much more complex cases, long-running, could be analysed through mechanistic codes and used to train the neural networks.

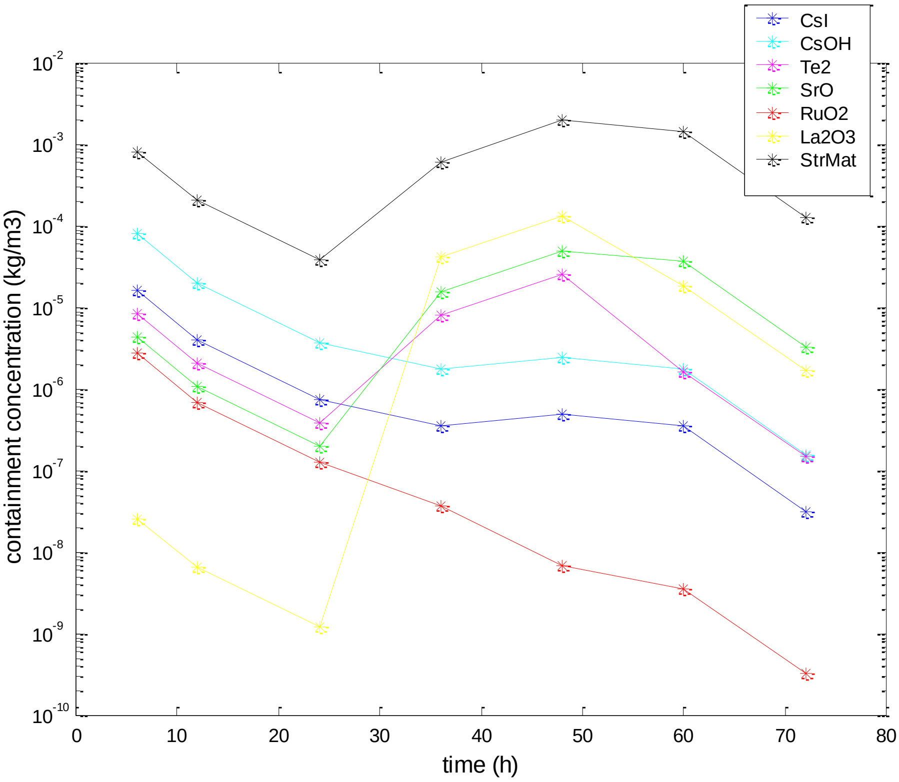

Fig. 58 Typical output pattern representing the containment concentration of the 7 key-elements

7.7.4.2 Release estimation in L2PSA at IRSN (example of 900 MWe PWR L2PSA)

IRSN uses software for fast release evaluations, called MER. This software may be run from KANT, the IRSN probabilistic software for APET development and quantification. Environmental releases for several thousands of release sequences can be computed with this software for a whole L2PSA. Additionally, MER can be run by EVARISTE, which is software able to produce some detailed post-processing and to calculate atmospheric dispersion and radiological consequences. Specific codes are used for atmospheric dispersion and doses evaluations (respectively pX and ConsX). These codes are coupled with MER within EVARISTE. The Fig. 59 shows the different steps for release evaluations.

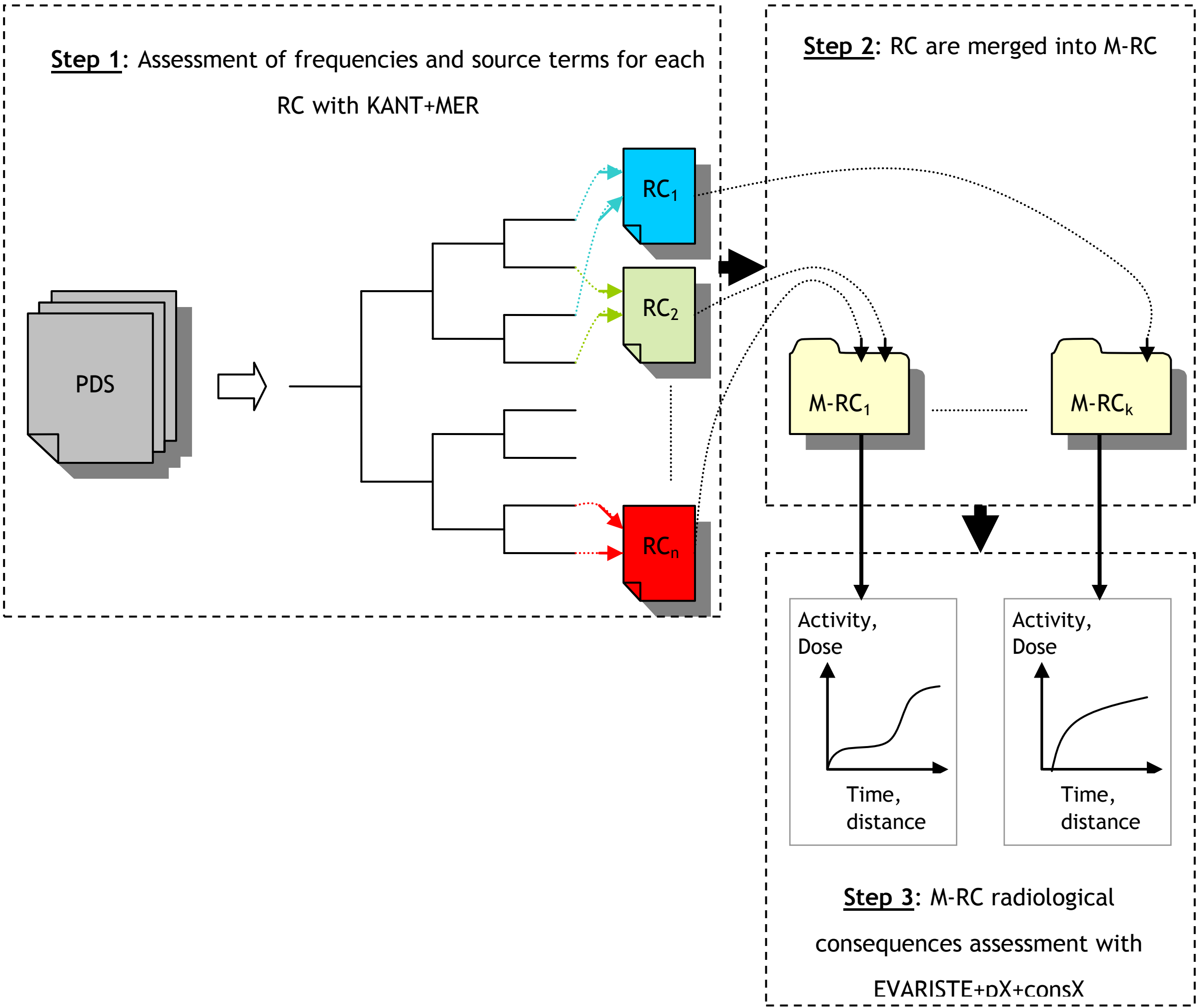

Fig. 59 Structure of L2PSA source term assessment in APET (From IRSN – L2PSA for French PWR)

The first step consists of a rapid global evaluation of the KANT software, for each PDS, to evaluate possible accident scenarios, and merge them into the RC1, RCi…RCn. Each RC is characterised by a set of relevant discrete variables, the values of which are computed at the end of APET calculations, according to scenario computed results. At this step, for each RC, KANT calls the MER software, which performs a fast calculation of environmental release for each fission product group and for each phase of the accident: during core degradation, at vessel rupture, after vessel rupture until final containment failure or until 2 weeks after vessel rupture. At this step, for the French 900 Mwe PWR PSA 2 study, about one thousand RCs are found, associated with an environmental release for each phase of accident.

At step 2, the RC are merged into M-RC (merged RC), according to these rules: in one M-RC, there is only one single containment failure mode type (all other modes being excluded) or one single containment failure mode combination type (all other combinations being excluded). For example, one M-RC must contain only early containment failure mode due to H_2 explosion, another M-RC must contain only initial leakage and early containment failure mode due to H_2 explosion. The second rule is that for a given M-RC, the environmental releases must be rather homogenous. As a consequence, investigations (based on the previous step applying MER) have to be done to retrieve what may be the influential variable(s) inside a single containment failure mode. For instance, for a given containment leakage phase, the influential variable is whether the auxiliary building filtration systems are functioning or not.

At the end of step 2, 44 M-RC were found for the 900 MWe PWR L2PSA. It should be noted that, as mentioned above, certain physical data are quite uncertain. At this stage of the process, uncertain parameters are set to their median value.

At step 3, M-RC characteristics are used to perform an environmental release calculation vs time for each species and each isotope. This source term is then applied in an atmospheric dispersion code, which is beyond the L2PSA realm.

Uncertainty evaluations are performed, using a Monte Carlo method, and sampling for uncertain parameters is made using the Latin Hypercube methodology.

The results obtained (beyond the normal scope of L2PSA) are the following:

- Dose equivalent and thyroid dose at given points (in the axis of the plume) versus time.
- Deposited activities 2 weeks from the beginning of the accident, among which ^{137}Cs deposited activity.
- Data linked to emergency preparedness: the instance when counter-measure levels are reached. At a given time, the geographical extent of each countermeasure.

For each of these numerical results, statistical distribution percentiles are evaluated: 5%, 50%, and 95%. The average value is also calculated.

Considering the calculated doses (2 km away from the plant, 15 days after the beginning of the accident) and the allocated frequency, a general and hierarchical idea of the accident sequences can be obtained. At this stage, the M-RC can even be merged into super categories to simplify the results presentation.

At the present time there are two MER models at IRSN, one for the 900 MW French PWR, and the other for the 1300 MW French PWR. Both have been used for a L2PSA. The 1300 MWe PWR model is more recent and includes recent results from the IRSN R&D on source term modelling, in full consistency with the ISTP programme. A model for EPR will be developed soon. Some information on the 900 version is presented below. The 1300 version is a little more sophisticated because it includes a modelling of double containment (single containment for 900 MWe PWR) and more detailed modelling for the chemical species. Table 43 summarises the attributes which have been used for defining the release categories in the first step.

Table 43 RC variables used to fully determine a release category (900 MWe PWR)

APET Phase	RC variable	Range
In-Vessel core degradation	CHRS availability	Yes/ partially/ No
	Containment breach size	0.065 cm2 to total failure
	Delay before containment failure	No failure At the beginning of the phase At the end of the phase

	Delay before vessel breach	From 1 hour to 16 hours
	RCS fission products retention	Yes/no
	Core reflooding	Yes/no
	Energetic phenomena in RCS (i.e. steam explosion)	Yes/no
	Average primary pressure	From 3 to 80 bar
	SGTR	No SGTR, SGTR with empty SG 1 or 2 tube(s) 10 tubes Induced SGTR
	Leakage by containment isolation failure (¤ mode)	0.065 cm2 to 50 cm2
	Venting system functioning	yes/partially/no
	Filtration type	Iodine filters High efficiency filters No filtering
	V-LOCA	Yes/no
	Energetic phenomena in containment	No energetic phenomenon ex-vessel steam explosion hydrogen combustion
Vessel rupture	Containment breach size	0.065 cm2 to total failure
	vessel rupture mode	No rupture Non energetic rupture Steam explosion
Ex-vessel phase	CHRS availability	Yes/ partially/ No
	Containment breach size	0.065 cm2 to total failure
	Leakage by containment isolation failure (¤ mode)	0.065 cm2 to 50 cm2
	First containment failure mode	No failure, H2/CO combustion, U5 filter, Containment slow pressurisation Failure of containment isolation valves
	Energetic phenomena in containment	hydrogen combustion, ex-vessel steam explosion,
	Delay before containment failure	5 to 24 hours
	Venting system functioning	yes/partially/no
	Filtration type	Iodine filters High efficiency filters No filtering

The following fission product groups are modelled:

- noble gases;
- aerosols (including aerosols containing I such as CsI);
- molecular iodine I_2;
- organic iodine ICH_3.

NB: It should be noted that, for the 1300 model, the aerosols are separated into three families: volatile, semi volatile, iodine aerosols; a specific family for ruthenium will be modelled; iodine includes 4 forms: aerosol, molecular, organic, oxide.

A part of the emitted iodine is supposed to be released in I_2 form directly to the containment (about 5% of the total amount of the iodine); the rest is in aerosol form.

Additional phenomena are modelled too:

- aerosols revolatilisation from boiling sumps;
- aerosols resuspension mechanisms inside the primary system or the containment building (in case of hydrogen combustion, in-vessel steam explosion, reflooding);
- Aerosols release at the beginning of the MCCI.

The main uncertain parameters are:

- Masses released from the fuel.
- Retention coefficients in secondary circuits, RCS for aerosols.
- Fraction of wall-deposited aerosols that subsequently undergo resuspension.
- I_2 adsorption rate on the containment walls.
- Molecular to organic conversion rate.
- For certain types of containment failures (early containment failure due to steam combustion, out-vessel explosion, direct containment heating), the containment break size.

As a conclusion, the IRSN model for environmental releases estimation and their consequences is fast-running with a precision fitted to purpose. It allows hierarchical representation of the plant safety as well as a comprehensive analysis of certain scenarios, such as the impact of iodine and aerosols uncertainties on emergency planning. IRSN has now initiated a specific project for a systematic validation/comparison of results obtained with this tool and with ASTEC on a limited number of accident scenario.

7.7.4.3 Support System for Radiation Experts in Loviisa L2PSA

Fortum, which owns and operates Loviisa NPP on the south-eastern coast of Finland, has developed their own source term model Support System For Radiation Experts (SaTu) [281],[282],[283],[284], which is used in L2PSA source term calculations. The system was originally developed to support decision making in an emergency situation and it is also used in emergency drills. Besides environmental releases, the system can also calculate radiation dose levels at the power plant for a given sequence and sequence input is easy to create. The SaTu system also includes a module for fast environmental dose calculations with statistical weather data. For L2PSA purposes, a sensitivity study module for calculating different parameter variations of the sequences as Monte Carlo is also included. The SaTu system is implemented as an Excel-application using Visual Basic extension.

The approach in SaTu is to describe the fission product release from the core and transport in the primary circuit, in the containment and in some other areas outside the containment with rather simple models. The source in a single compartment is formed as a combination of the release from the core and the fission product flows from other compartments, and averaged in each time step. The flow chart of the SaTu system is shown in

Fig. 60. The compartments described with boxes with orange bottoms may have fission product deposition on the floor and those with blue bottoms also in a water pool. The bold solid line surrounds the compartments within the containment. Stack is not an actual calculation compartment, and in the figure it only shows a release path through the ventilation system. The pipe systems, e.g. primary circuit, have zero gas volumes, but part of the material is assumed to deposit on pipe surfaces.

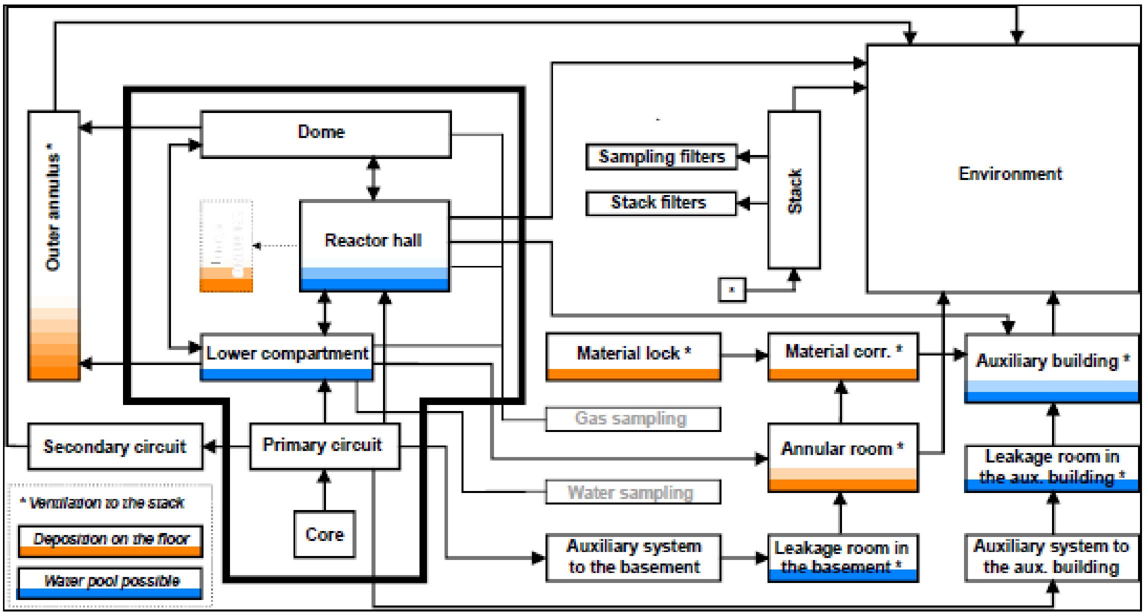

Fig. 60 Flow chart of the SaTu system

There are ten different fission product groups in the SaTu system (see [281]

Group	Elements in group
Noble gases	Xe, Kr
Iodine	I, Br
Cesium	Cs, Rb
Tellurium	Te, Sb, Se
Strontium	Sr
Rutenium	Ru, Rh, Pd, Mo, Tc
Lantanium	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
Cerium	Ce, Pu, Np
Barium	Ba
Uranium	U

Table 44 Fission product groups is SaTu system

The system calculates the fission product release from the reactor core during meltdown and the transport in the containment and in some other building areas. The deposition in the primary circuit and in the interfacing system piping is also taken into account. The fission product release from the core is estimated from the information on the core exit temperatures and depressurisation of the primary circuit. The release fractions are based on MELCOR calculations on several severe accident sequences for the Loviisa NPP. In SaTu, the release model from the core is described with constant release rates of fission products from the core. The release rate is assumed to be proportional to the decay heat of the core. Different constants are used for release during core degradation, from molten pools and from core material in the reactor cavity after the RPV failure.

Transport of fission products in the containment and areas outside the containment takes place mainly along with gas flows. The containment flows during severe accidents are driven by a global loop flow through ice condensers after forcing open the ice condenser doors. The stratification of the upper compartment (UC) between the dome and the reactor hall is modelled, as well. Leakages out of the containment are calculated from the containment pressure and leak sizes. Transport outside the containment is driven by the leak from the containment and by a ventilation system that forces the air flow into the exhaust stack. In addition to the stack, five other release locations are included in the model.

In the interface of Loviisa Levels 1 and 2 PSA, the sequences from Level 1 are grouped into core melt bins (CMB). From the CMB it can be seen whether the sequence is a containment sequence or a containment bypass sequence, what is the initial leak size for LOCA sequences, or in other cases what type of initiator is considered. It can also be seen whether the ECCS is available or not etc. These core melt bins are then calculated through the containment event tree (CET). In Loviisa L2PSA, CET end states are called accident progression categories (APC). With a screening frequency of 10-9, source terms and environmental releases are calculated for all the combinations of CMBs and APCs. For the Loviisa L2PSA study (power and shutdown) this means that the source term input will be created and source term will be calculated with the SaTu system for approximately 200 different sequences. Source term calculations and results handling have been made automatic, but the sequence input have to be made for each sequence separately. Fig. 61 illustrates the cesium release to the environment for one L2PSA sequence and the associated uncertainty and Fig. 62 illustrates the level 2 cumulative caesium source term results for a set of containment sequences.

Results from the SaTu system in Loviisa cases have been compared to respective results from MELCOR, CONTAIN and COCOSYS. All together some 20 cases were compared and results showed that SaTu is able to give reasonable results and it can be used for source term calculations for the Loviisa NPP.

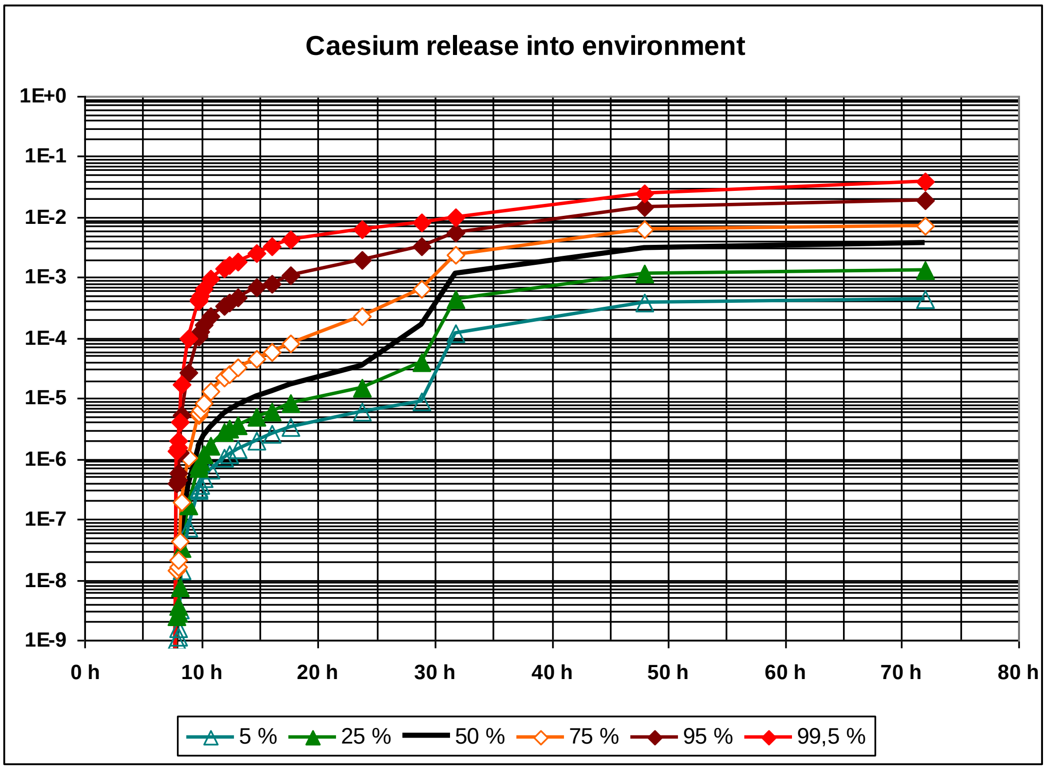

Fig. 61 Cesium release into environment, typical result from SaTu system. Sequence used as an example is a small LOCA, with ECCS injection failure and containment isolation failure

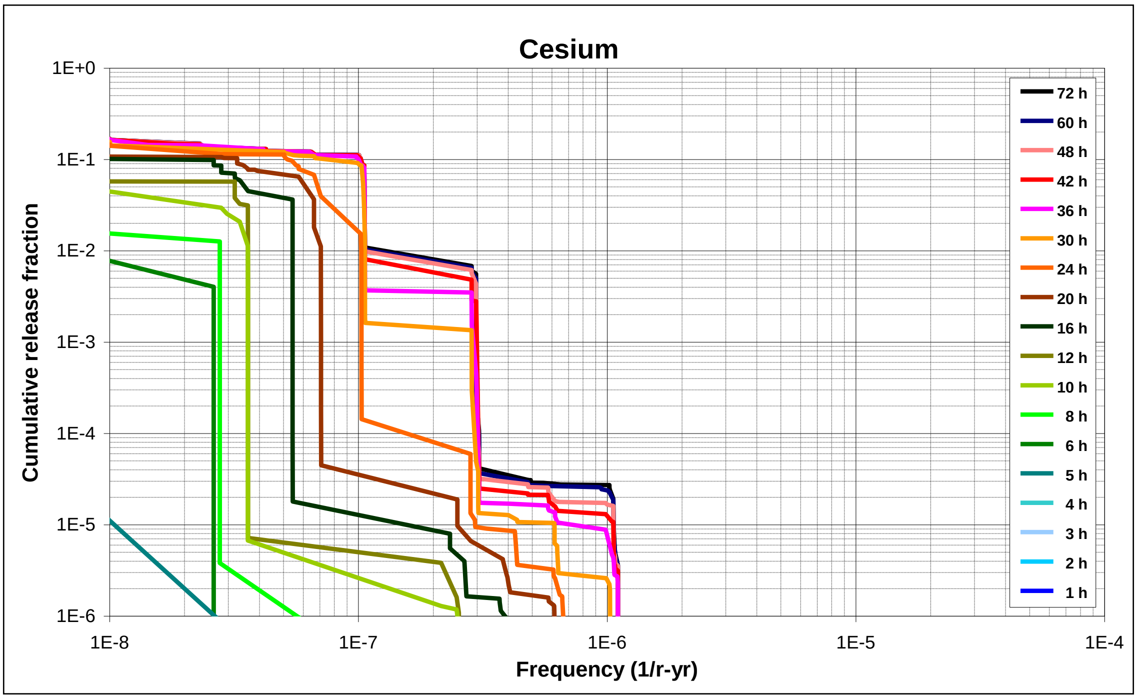

Fig. 62 L2PSA source term sample results for cesium releases in containment sequences (power operation)

7.7.5 Attribution of hazard to the public to source terms

Typically, a source term provides information about the release quantity and release mode of various isotopes into the environment. However, the risk to public health and the wider environment, which may be the ultimate interest in assessing safety, cannot readily be deduced from this information. To this end, probabilistic consequence calculations can be performed as part of a Level 3 PSA, covering a wide spectrum of weather, human health effects and environmental consequences.

However, some extended L2PSAs (sometimes called Level 2+) use source term information to estimate an indicative off site consequence, using a simplified effective dose calculation, without the effort of a complete Level 3 PSA. This enables an approximate ranking of different accident sequences, according to the impact on public health in the absence of off site countermeasures. Properly done, this approach can also be used as an estimation of the impact of accident management or plant modifications on the off site dose, which is used as a risk surrogate.

If the objective of L2PSA is restricted to qualitative characterisation of source terms (e.g. in order to identify “large” releases), it is not necessary to consider many different isotopes. Often PSAs just indicate releases of Iodine, Caesium and noble gases. However, if hazard to the public or to the environment has to be determined, all relevant nuclides have to be taken into account. From a technical point of view this does not require a large additional effort since most PSA apply integral computer codes for analysis, and these routinely provide source terms for all radionuclides of interest anyway. From a practical point of view, one could dismiss radionuclides which contribute in summary less than 10% to the total radiological effect. This is justified because the relative uncertainties in source term assessment are bigger than 10%.

The following concepts can be envisaged:

- Characterise the source term as the sum of released activity (in Bq). This can be easily done by adding the activities of the individual isotopes which contribute to the source term. This provides a very rough indication of the relative environmental effects of release categories.
- Characterise the source term as the sum of released activity (in Bq), weighed with appropriate biological toxicity of each isotope. This can be done by adding the activities of the individual isotopes which contribute to the source term and multiplying them with appropriate relative toxicity factors. Such factors can, for example be found in the INES manual [290]. This provides a very rough indication of the relative health effect of release categories.
- Characterise the source term as an integral indicative health effect, e.g. the effective dose in the absence of off site countermeasures, which is directly linked to the likelihood of health effects at a certain distance from the plant. This approach does not need to assess in detail the site specific issues. However, this assessment requires assumptions about the dispersion of activity in the environment and about the intake and effects of the activity in humans. Unless these assumptions are clearly indicated, this type of characterisation is not recommended. Note that any comparison between release categories, or risk based on such assessments, has to be consistent with regard to the environmental, social and medical assumptions in the absence of Level 3 PSA results.

For the last item, simplified deterministic calculations of offsite effective doses in the absence of off site countermeasures may be calculated in a manner similar to offsite doses as calculated for Design Basis Accidents. It is recommended that the IAEA guidelines for estimates of doses in safety demonstrations (see for instance among other IAEA safety series documents, IAEA-TECDOC 953/1997 [286] and IAEA-TECDOC-955/1997 [287]) should be followed. Such results can be used to assess:

- whether a Release Category falls above or below a pre-defined effective dose threshold for members of the public, e.g. for compliance with dose limits;
- ranking of Release Categories;
- sensitivity to specific systems, events and phenomena;
- potential impact of SAM measures.

However, care should be taken in quoting individual doses resulting from severe accidents as the concept of the dose unit Sievert breaks down at levels in the region of 1 Sv and the dose unit Gray applies at higher exposures. In addition, individual doses calculated for members of the public close to the site that are far above the threshold for early health effects, or even early fatalities, are not very meaningful. At these extreme levels the concept of societal harm (for example in terms of the expected number of fatalities) is more meaningful. This can only be addressed in Level 3 PSA.

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