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"NUCLEAR FISSION"

Safety of Existing Nuclear Installations

Contract 211594

Warning: this draft document is proposed by the ASAMPSA 2 Partners for an external review. The content may evolve in the next versions.

BEST-PRACTICES GUIDELINES FOR L2PSA DEVELOPMENT AND APPLICATIONS

Volume 1 - General

Reference ASAMPSA 2

Technical report ASAMPSA 2/WP2&3/ 2010-28

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Authors	E. Raimond (IRSN), F. Pichereau (IRSN), T. Durin (IRSN), N. Rahni (IRSN), H. Löffler (GRS), O. Roesch (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), C. Bassi (CEA), K. Peers (AMEC NNC), G. Hulqvist (FKA), F.Parozzi (ERSE), F. Polidoro (ERSE), E. Cazzoli (CCA), J. Vitazkova (CCA), L. Burgazzi (ENEA), H. Brinkman (NRG), A. Seidel (VGB), B. Schubert (VGB / VATTENFALL), R. Wohlstein (VGB / EON), S. Guentay (PSI), L. Oury (TRACTEBEL), C. Ngatchou (TRACTEBEL), S. Siltanen (Sipola) (FORTUM), I. Niemela (STUK), T. Routamo (STUK), L. Vinçon (AREVA SAS), P. Helström (SCANDPOWER)
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Summary :

The Volume 1 of the ASAMPSA2 guideline presents general recommendations regarding L2PSA development and applications

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

Name	Organisation
All ASAMPSA2 Partners.	
Specific list of organizations concerned by L2PSA development and applications for NPP.	

ASAMPSA2 PROJECT SUMMARY

The objective of this coordinated action was to develop best practice guidelines for the performance and application of Level 2 PSA with a view to achieve harmonisation at EU level and to allow a meaningful and practical uncertainty evaluation in a Level 2 PSA.

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 Level 2 PSA development and application. An international workshop was organised in Hamburg, with the support of VATTENFALL, in November 2008.

The Level 2 PSA experts from ASAMPSA2 project partners have proposed some guidelines 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).

There are a large number of technical issues addressed in the guideline which are not all covered with the same level of detail in the first version of the guideline. This version was submitted for external review in November 2010 by severe accident and PSA experts (especially from SARNET and OECD-NEA members).

The feedback of the external review will be discussed during an international open workshop planned for March 2011 and all outcomes will be taken into consideration in the final version of this guideline (June 2011).

The guideline 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 the guideline should not be considered as "mandatory" but should help Level 2 PSA developers to achieve high quality studies with limited time and resources. It may also help Level 2 PSA reviewers by positioning one specific study in comparison with some state-of-the art information.

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 Grou	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 Level 2 PSA 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 Level 2 PSAs 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 Level 2 PSA which have been performed and discussed in (6th EC FP) SARNET L2PSA work packages, have shown that the detailed criteria and methodologies of current Level 2 PSAs 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 to highlight common best practices, develop the appropriate scope and criteria for different Level 2 PSA 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 Level 2 PSA. 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 is 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 utilises 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 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 Level 2 PSAs 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. Level 2 PSA 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 not achievable</u> and it has been decided to present in the report, for each issue, some recommendations that may refer to simplified or detailed approaches.

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 Level 2 PSA, performed and discussed in SARNET L2PSA work packages and also in CSNI workshops (Koln 2004, Petten 2004, Aix en Provence 2005), have shown that the high differences in practical implementation of Level 2 PSAs 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 has strong links with the SARNET Network of Excellence since it will be based on the already published work on Level 2 PSA carried out by SARNET. However it has departed from the level of progress achieved by different countries and international organisations (e.g., in the USA, regulatory documentation has been developed as well as various industry standards (ASME, ANS) and the IAEA recently updated a Safety Standard on Level 2 PSA). Conclusions of SARNET activities on Level 2 PSA harmonisation and the last version of IAEA Safety Standard on Level 2 PSA have constituted the departure point of the coordination action.

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 ASAMPSA project was assumed by IRSN, including all administrative tasks and relationship with EC services.

VOLUME 1 CONTENT

GLOSSARY

AGR	Advanced Gas-cooled Reactor
AICC	Adiabatic Isochoric Complete Combustion
APET	Accident Progression Event Tree
ASEP	Accident Sequence Evaluation Program
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transient Without Scram
BDA	Boron Dilution Accident
BMMT	BaseMat Melt-Through
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAHR	Connectionism Assessment of Human Reliability
CCW	Component Cooling Water
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
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
ESF	Engineered Safety Feature
FCI	Fuel Coolant Interaction
FLIM	Failure Likelihood Index Methodology
FP	Fission Product

FSAR	??
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
IRWST	In-containment Residual Water Storage Tank
IVMR	In-Vessel Melt Retention
IVR	In-Vessel Retention ?? (remove)
L1PSA	Level 1 PSA
L2PSA	Level 2 PSA
LAM	Late Accident Management
LDW	Lower Drywell
LER	Large Early Release
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LHSI	Low Head Safety Injection
LMP	Larsson Miller Parameter
LWR	Light Water Reactor
MCCI	Molten Core Concrete Interaction
MCS	Minimal Cutset
MERMOS	Methode d'Evaluation de la Réalisation des Missions Opérateur pour la Sureté
NPP	Nuclear Power Plant
PAR	Passive Autocatalytic Recombiner
PDF	Probability Density Function
PORV	Power Operated Relief Valve
PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PWR	Pressurised Water Reactor
R&D	Research and Development
RCS	Reactor Coolant System
ROAAM	Risk Oriented Accident Analysis Methodology
RPV	Reactor Pressure Vessel

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 LOCA
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
SPAR-H	Standardized 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

DEFINITIONS

This part has to be completed

1 INTRODUCTION

The objective of the present guideline is to identify some best-practices regarding Level 2 Probabilistic Safety Assessment (L2PSA) development and applications. 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 Appendix 9.5 and can be used as a material to prepare the next review of the draft guideline.

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 are 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, 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 Level 2 PSA. The technical recommendations are based on the Authors experience (or open literature). They should not be considered as "mandatory" but are supposed to help the L2PSA developers or reviewers to improve the quality of the Level 2 PSA they consider.

The ASAMPSA2 guideline has 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 the Authors to define any quantitative safety requirement.

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 Level 2 PSA, 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 Level 2 PSA 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 Level 1 PSA,
- 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:

<u>Volume 1 - General considerations on L2PSA</u>

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.

<u>Volume 2 - Technical recommendations for Gen II and III reactors</u>

This volume provides recommendations regarding specific methods to be used in a L2PSA (Level1/Level 2 PSA 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).

<u>Volume 3 - Specific considerations for future reactor (Gen IV)</u>

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 Level 2 PSA 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 should provide useful information on all of these issues for either the L2PSA developers or the reviewers.

1.3 REFERENCE

[1] Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, 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

2 STRUCTURE OF A LEVEL 2 PSA AND RELATED ACTIVITIES

The intention within this chapter is to give an overview of a Level 2 PSA project. All details on the different elements that constitute a L2PSA can be found in the other chapters of the guideline.

2.1 OVERVIEW

Level 2 PSA aims to quantify source term risk distribution of a Nuclear Power Plant. For this objective, frequency distributions and associated source term distributions are calculated for a certain number of Release Categories that cover all potential release modes from the plant (in the case of an accident) either combined or separately. The methodology used is now standardised:

- L1PSA core damage sequences are gathered in Plant Damage States if they are equivalent in terms of severe accident progression and source term risk profile,
- For each selected Plant Damage State, several severe accident sequences paths are tracked with all their potential branching with the aid of an Accident Progression Event Tree (also called Containment Event Tree) to quantify the frequency distributions for each Release Category,
- These assumptions of the Accident Progression Event Tree, as well as the quantification of the associated source term distributions, are supported by deterministic calculations with integrated severe accident codes such as MAAP, MELCOR or ASTEC and with complementary codes such as MC codes to quantify source term or split fraction distributions, as well as dedicated codes for some specific issues (structural strength, steam explosion, hydrogen distribution in the containment …).

This methodology needs the following activities to be performed:

1. Plant familiarisation;
2. Definition of the L2PSA objectives;
3. Accident Sequence Analysis, Analysis of Phenomena, Source Term Analysis;
4. Containment Analysis;
5. Human Reliability Analysis;
6. Systems Analysis;
7. Event tree Modelling;
8. Quantification of Event Trees , Results, Presentation, and Interpretation;
9. Documentation.

Fig. 1 presents the different activities linked to Level 2 PSA.

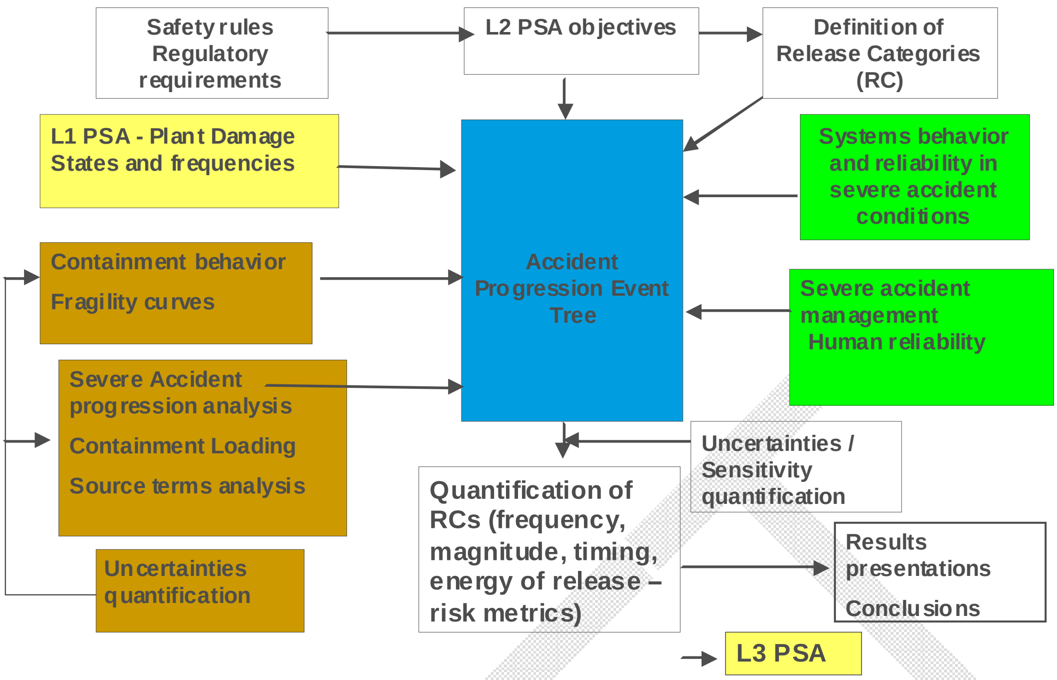

Fig. 1 Overview of a Level 2 PSA Project Activities

2.2 PLANT FAMILIARISATION

It is important that plant characteristics of significance for accident progression are identified and described in support of the Level 2 PSA. Reference [1] provides an example of key plant and/or containment design features that are significant to the progression and mitigation of severe accidents, which is reproduced and completed in Table 1.

Table 1 Example of key plant and/or containment design features [2]

Key plant and/or containment design feature	Comment
Reactor type	BWR/PWR/other
Power level	Actual thermal power
Fuel/cladding type and mix	Oxide, mixed oxide/Zr, etc.
Reactor coolant and moderator type	Water, heavy water, others
RCS coolant/moderator volume	As designed and fabricated
Accumulator volume and pressure setpoint	Actual operational values
Containment free volume	As built
Containment design pressure/temperature	As designed
Containment structure	Steel, concrete
Operating pressure, temperature	Actual operational values
Hydrogen control mechanisms	Inerted, ignitors, recombiners, others
Mass of fuel	Actual operational values
Mass of cladding material	Actual operational values
Control rod type and mass	Actual operational values
RCS depressurisation devices/procedures	Specify setpoint/procedures
Pressure relief capacity	Actual operational value

Key plant and/or containment design feature	Comment
Suppression pool volume	Water and atmosphere volumes
Containment cooler capacity and setpoints	Actual operational values
Concrete aggregate	Specify chemical content
Cavity/keyway, pedestal design	Dispersive, non-dispersive
Flooding potential of cavity/pedestal	Flooded, dry
Sump(s), volume and location(s)	Specify details
Proximity of containment boundaries	Relative to reactor vessel
Accident consequences limiting design features like venting procedure and vent location	Specify location/procedures
Containment geometry	Compartmentalisation
Description of containment penetrations	As designed and included operating experience
Description of containment isolation systems	As designed and included operating experience
Containment vulnerability to different phenomena	First by expert judgement then supported by specific studies
Basemat features (concrete composition, thickness, existence of bypass ways like control access)	This specific information may not be available in the basic documentation of the plant.
Design limits of materials	As designed, for comparison with severe accident conditions.
External events impact	Seismic, flooding and impact
Potential for bypass	Penetrations/interfaces

More data is needed to analyse the severe accident progression including Emergency Operating Procedures, Severe Accident Management Guidelines, systems, automatic actions, core composition, and containment integrity
Since Level 2 PSAs cover sequences beyond design, the plant's documentation sometimes does not easily reveal issues of interest in Level 2 PSA. A typical example is the existence of drain lines, pump sumps, ventilation ducts, concrete composition or penetrations in the bottom part of the containment where corium might be present. Such details are important for the containment's ability to withstand corium attack, but the documentation of details could be so poor that visiting critical areas is needed. It is very helpful to have a qualified system of photographs or videos to avoid time consuming plant inspections which may be difficult due to safety and security concerns.

2.3 DEFINITION OF THE L2PSA OBJECTIVES

The definition of the L2PSA objectives should be one of the first tasks to be performed before developing or updating a L2PSA. A list of general PSA applications has been proposed in the L2PSA IAEA safety standard [1] and is reproduced hereafter:

(1) to provide a systematic analysis to give confidence that the design will comply with the general safety objectives;

(2) to demonstrate that a balanced design has been achieved such that no particular feature or PIE (postulated initiating event) makes a disproportionately large or significantly uncertain contribution to the overall risk, and that the first two levels of defence in depth bear the primary burden of ensuring nuclear safety;

(3) to provide confidence that small deviations in plant parameters that could give rise to severely abnormal plant behaviour ('cliff edge effects') will be prevented;

(4) to provide assessments of the probabilities of occurrence of severe core damage states and assessments of the risks of major off-site releases necessitating a short term offsite response, particularly for releases associated with early containment failure;

(5) to provide assessments of the probabilities of occurrence and the consequences of external hazards, in particular those unique to the plant site;

(6) to identify systems for which design improvements or modifications to operational procedures could reduce the probabilities of severe accidents or mitigate their consequences;

(7) to assess the adequacy of plant emergency procedures;

(8) to verify compliance with probabilistic targets, if set."

The same IAEA safety standard [1] provides a formulation of general L2PSA objectives;

"*A Level 2 PSA covers the progression of events that would occur in nuclear reactors following accident sequences that have led to significant damage to the reactor core. The main objective of the analysis is to determine if sufficient provisions have been made to manage and mitigate the effects of such an accident. These provisions could include:*

- *Systems provided specifically to mitigate the effects of the severe accident such as molten core retention features, hydrogen mixing/recombiners or filtered containment venting systems;*
- *The inherent strength of containment structures or capability for radioactive material retention within a confinement building, and the use of equipment provided for other reasons for accident management;*
- *Guidance to plant operators on severe accident management."*

It also provides examples of more precise applications that could be assigned to a specific L2PSA:

- *To gain insights into the progression of severe accidents and containment performance;*
- *To identify plant specific challenges and vulnerabilities of the containment to severe accidents;*
- *To provide input into the resolution of specific regulatory concerns;*
- *To provide an input into determining whether quantitative safety criteria that typically relate to large release frequencies (LRFs) and large early release frequencies (LERFs) are met;*
- *To identify major containment failure modes and their frequencies and to estimate the corresponding frequency and magnitude of radionuclide releases;*
- *To provide an input into the development of off-site emergency planning strategies;*
- *To evaluate the impacts of various uncertainties, including assumptions relating to phenomena, systems and modelling;*
- *To provide an input into the development of plant specific accident management guidance and strategies;*
- *To provide an input into plant specific risk reduction options;*
- *To provide an input into the prioritisation of research activities for minimization of risk significant uncertainties;*

- *To provide an input into the Level 3 PSA consistent with the PSA objectives;*
- *To provide an input into the environmental assessment for the plant."*

It may be difficult to precisely define the objectives that could be assigned to a L2PSA because they must depend on the local regulatory context, the type of plant (Gen II, III, IV for example), and the specifics of the particular site. Many variations exist in the practical way of presenting the results of a L2PSA, as explained in Chapters 4 and 5. Chapter 3 presents information related to the practices of different countries and how they differ. Chapter 3 also describes the position of international organisations like WENRA.

This information could then be used to help define precise objectives associated with a L2PSA for a specific plant. Once these objectives have been defined the L2PSA scope, content, and methodology can be defined.

Chapter 6 proposes a tentative definition of a harmonised safety goal that may be applied for all plants.

2.4 ACCIDENT SEQUENCES ANALYSIS, ANALYSIS OF PHENOMENA, SOURCE TERM ANALYSIS

To develop a L2PSA, a good understanding of how the plant behaves in an accident is necessary. Deterministic calculations of accidental transients (thermal hydraulic and source term) may need to be performed to support the APET model development. Thermal-hydraulics calculations of accident transients can help to group Level 1 PSA sequences in the Plant Damage State that will show the same accident progression in the APET.

It is necessary to identify important phenomena for accident progression and release categories during the plant familiarisation phase. Some phenomena are a natural part of the sequence development whilst others are threats to the containment integrity. All must be taken into consideration in the development of the APET nodes. It is necessary to perform deterministic studies to quantify the impact of each event or phenomena on accident progression and containment integrity and some specific methodologies have to be used to correctly handle the dependencies between the events and to assess the uncertainties. The accident sequence analysis should provide enough information to design the APET. More specific methods, like Success Block Diagrams (SBDs), can also be used to help in this process.

More information and recommendations regarding accident sequence and phenomena analysis have been provided in Volume 2, chapter 4.

For a L2PSA it is necessary to estimate the amplitude and kinetics of radioactivity for all of the accident sequences considered in the study. This source term analysis needs development and the application of appropriate specifications for modelling of the plant and all release paths. Details have been provided in Volume 2, chapter 7.

2.5 CONTAINMENT ANALYSIS

The plant familiarisation should provide a general description of the containment and should help to define the different containment failure modes. The containment analysis should include:

- The potential for loss of containment leaktightness due to phenomena (pressure peak for example): fragility curves are generally applied for the intact containment shell as well as for all major imperfections (such as penetrations) and the associated break size,
- The potential for containment isolation failure,
- The potential for containment bypass (interfacing system-LOCA, steam generator tube rupture for PWRs).

The analysis of an un-isolated containment can be based on fault trees, identifying all penetrations and systems connected to these, availability of isolation valves, assessment of the reliability of the isolation signals and the isolation components, and considering the contribution from any inadvertent openings.

Information and recommendations regarding containment analysis have been provided in Volume 2, chapter 5.

2.6 HUMAN RELIABILITY ANALYSIS

The plant familiarisation will include information about the plant emergency organisation (operator, local emergency teams, national emergency teams) and important operator actions, related emergency operating procedures and response to severe accidents. Examples of areas of importance for accident management by the operators are:

- Pressure control/relief in the primary system before vessel failure,
- Containment cooling,
- Hydrogen management,
- Containment pressure relief strategy,
- Mitigation of radionuclide releases to environment.

The Human Reliability Analysis in L2PSA aims to quantify the probability of failure of each operator action that should be performed during a severe accident sequence.

Operator actions modelled in the L1PSA sequences have to be identified and the potential impact from a Level 2 perspective has to be investigated. There may be addition of more actions, change of time available or time windows for performing the actions. One factor to consider is if an action may prevent vessel failure but would not prevent core damage in a Level 1 PSA perspective.

Operator actions that are part of the Level 2 PSA accident sequences development affecting the timing, consequences, etc. are identified. The actions are described concerning their importance which is defined according to when they occur and the phase of accident sequence development. Factors which affect the probability of failure of the various actions are also identified and described.

The human error probabilities (HEPs) and related uncertainties are evaluated with a suitable consistent method for actions in the combined Level 1 and Level 2 PSAs.

Considerations of any dependencies are described – between events in both the Level 1 and Level 2 PSA, and between events in the Level 2 PSA.

The potential for recovery (repair) of failed equipment may be looked at. This may be more important for dominating sequences where the accident evolves slowly but radiological conditions have to be taken into account and modify the probability of success in comparison with assumptions that may be used in L1PSA.

The human actions basic events are introduced into the PSA model fault trees and event trees (Level 1, APET/CET) and should include consideration of any backup provided by a crisis team and the national organisation.

All details regarding Human Reliability Analysis have been provided in Volume 2, Chapter 3.

2.7 SYSTEMS ANALYSIS

Systems analysis is performed for Level 1 functions/systems that need to be updated with regard to Level 2 and for new functions and systems in the Level 2 PSA. The input to systems analysis is from the accident sequence analysis that identifies functions/systems and their success criteria in different accident sequences.

The systems analysis task also interacts with the human reliability analysis task for analysis of system specific operator actions. The specifics of each severe accident have to be taken account.
Details are provided in Volume 2, chapter 6.

2.8 EVENT TREE MODELLING

Once all information is available the event tree and fault tree models are created:

- Assignment of plant damage states to the Level 1 PSA sequences,
- Additional modelling of bridge trees (if bridge tree technique is used),
- Necessary updating of Level 1 PSA part of the model (event trees, system fault trees, basic events),
- Additional system fault trees development for the Level 2 PSA,
- Definition of release categories,
- Creation of APET/CET structure including release categories as end states in the Level 2 PSA event tree sequences.

All details have been provided in Volume 2, chapter 2 of the guideline.

2.9 QUANTIFICATION, RESULT PRESENTATION AND INTERPRETATION

The purpose of the quantification of the PSA model is to obtain results in terms of the frequency distributions for all release categories and any intermediate results of interest. This includes specific results such as:

- the plant damage states total frequency and contribution arising from different initiating events in the level 1 PSA part (minimal cutsets),
- the release categories of total frequency and contribution which have arisen from different initiating events / plant damage states and specific events resulting from the severe accident progression.

In some studies, the quantification can include the calculation of amplitude and kinetics of release for each individual sequence or for each release category.
The individual sequences from Level 1 or the PDS can be quantified separately which can help in determining which sequences that are most important for each plant damage state and release category.
It may also be of interest to calculate the fault tree top events representing functions and systems in the Level 2 PSA (1) event trees.
In addition to point values, both importance and uncertainty analysis and separate analysis of sensitivity cases should be quantified.
It must be noted that the setup of the quantification is intimately related to the PSA modelling approach and the software probabilistic tool being used as explained in Volume 2, chapter 2.
The results to be presented in a Level 2 PSA project depend on the objectives of the study. This aspect is detailed in Volume 1 chapters 5 and 6.

2.10 DOCUMENTATION

The documentation of a Level 2 PSA usually follows the different tasks and activities that are performed in the project. A considerable quantity of information can be associated with a L2PSA. For the sustainability of the study and also to allow external review, the documentation is considered a crucial element of the L2PSA quality.
A tentative outline for a L2PSA summary report is given below:

- Introduction,
- Plant Description,
- Methods/Procedures/General assumptions and limitations,
- Synthesis of Level 2 PSA Accident Sequences Analysis:
  - Level 1 / Level 2 Interface,
  - CET/APET Development,
  - Release categories definition,
- Synthesis of Containment Performance Analysis,
- Synthesis of Phenomena Analysis,
- Synthesis of integral accident progression Analyses,
- Synthesis of Systems Analysis,
- Synthesis of Human Reliability Analysis,
- Synthesis of Source Term Analysis,
- Synthesis of PSA Event Tree Modelling,
- Synthesis of the quantification of frequency and source term distribution,
- Results Presentation and Interpretation, including sensitivity studies/uncertainties treatments,
- Conclusions and Recommendations,
- Appendices with details on all different supporting analyses such as:
  - Thermal hydraulics,
  - In-vessel core degradation,
  - Hydrogen combustion,
  - Containment strength,
  - MCCI,
  - Source Term assessment.

Outside the L2PSA summary report, the supporting documentation should be drafted with the objective to maintain all knowledge and justifications of probabilistic assumptions during the plant life. Periodic update of this documentation should be managed in relation to the update of the L2PSA.

2.11 MANAGEMENT OF A PSA IN SUPPORT OF THE OBJECTIVES

The management tasks of a Level 2 PSA project are:

- Definition of scope and objectives of the Level 2 PSA,
- Planning. This includes resource allocation, securing of resources, and coordination of different specialists,
- Development of project specific instructions and methodology guidelines,
- Follow-up of project performance,
- Review.

The definition of scope and objectives of the Level 2 PSA project at the beginning of the project is of vital importance since it will have a major impact on the resources and competencies that are required, and also the time schedule and eventually the cost.

It is therefore very important to identify the objectives necessary to satisfy the stakeholders (the regulator, the owner, the local organisation). These objectives are then essential for defining the scope of the project:

- Plant status (the plant design at a specific date to be analysed, or several designs if the Level 2 PSA is an input to choice of design features),
- Sources of radioactivity (the core, spent fuel, fuel during transportation etc).
- The initial reactor states to be considered (operating modes, full power, partial power, different start up and shutdown states).
- Type of initiators included (basic loss of coolant and process related events, area events, external events any restrictions on which types of external events that shall be addressed).
- End states (definition of end states are part of the work, but may be a condition depending on the objectives and regulatory requirements).

A Level 2 PSA with the objective to show that the risk is below a certain safety goal (risk target) may require less effort compared to a study required to present realistic results on source terms and release frequencies.

The Level 2 PSA project needs a multidisciplinary team with experts covering many areas; PSA, source term prediction, accident progression, phenomena, plant behaviour during severe accidents, containment mechanical behaviour, containment systems, human reliability, data, and deterministic and probabilistic software. It may also include plant and site specialists.

The different activities in the project will need guidance and coordination between the activities. Examples are:

- PSA model naming and modelling conventions,
- Definition of accident progression analysis: a L2PSA could generate an infinite number of different accident scenarios. It is therefore necessary to define a method to limit the number of studies to support the L2PSA development,
- Human Reliability: a specific methodology is required to be applied to the quantification of all human failure events,
- Systems analysis: it is necessary to develop specific methodology or criteria to quantify the system failure and repair in a homogeneous way,
- Planning of the activities: the high level of coupling between the different topics can make the organisation of the different tasks difficult. It is highly recommended to identify all dependencies between the different activities in the L2PSA planning. However rules need to be defined to allow each task to progress in parallel,
- Quality Assurance Procedures: some specific procedures should be defined to assure the homogeneity of the study and to verify the relevancy of parts of the study. The verification process can be based on internal resources but can also rely on external contributions (experts for specific topics, reviews by other organisations having already developed L2PSA).
- Results communication: the summary L2PSA report should present all assumptions and results obtained. However when discussing specific applications of the L2PSA, an adapted communication between the L2PSA developers and the stakeholders (decision-makers) needs to be organised.

2.12 COMMUNICATION OF L2PSA RESULTS

The communication of the L2PSA results, which provide a global measurement (and induce judgement of the NPP level of safety when compared to other NPPs) of the safety level of a NPP, needs a prudent approach:

- The numerical results should always be accompanied by precise explanations, especially for the dominant risks,
- Specific warning related to the lack of knowledge on some parts of the plant behaviour in severe accident conditions should be provided. In cases where uncertainties are assessed in the L2PSA, this lack of knowledge should be introduced in the uncertainty band of distribution of frequency or amplitude of release,
- Specific warning related to L1PSA assumptions may be provided (quality of system reliability data, quality of the functional analysis) especially if a L2PSA dominant risk is linked to L1PSA sequences with a low quality of analysis.

In general, all limitations of the study should be provided in the summary report and need to be considered before any decision is made based on the L2PSA conclusions. The limitations can concern the data, the modelling, the state of knowledge and also the scope of the PSA. For example, if the L2PSA scope is limited to internal events, then the frequency of some release categories may be highly underestimated. All these aspects should be explained by the L2PSA developers to the stakeholders.

It is highly recommended to bring together numerical L2PSA results and all of the qualitative conclusions that have been obtained from the perspective of plant design and operation improvement.

3 THE CURRENT SITUATION REGARDING L2PSA ACTIVITIES AND APPLICATIONS

This chapter presents a review of the current background regarding L2PSA activities and applications. It introduces the general situation at international level without any additional input from the ASAMPSA2 project. This situation will certainly evolve in the near future and this information has to be used carefully. Nevertheless, the chapter provides some global views on the different stakeholders' positions.

3.1 IAEA REFERENCE DOCUMENTS AND ACTIVITIES

A recent overview of the IAEA reference documents and activities that can be useful for L2PSA development and applications has been provided in reference [3] and [4]. With the permission of the authors, the second article has been reproduced hereafter.

3.1.1 Introduction

Consideration of beyond design basis accidents of nuclear power plants (NPPs) is an essential component of the defence in depth approach which underpins nuclear safety ([5] to [7]). Beyond design basis accidents that may involve significant core degradation are of particular interest for accident management - a set of actions taken during the evolution of a beyond design basis accident made to prevent the escalation of the event into a severe accident; to mitigate the consequences of a severe accident and to achieve a long term safe stable state. The IAEA Safety Standards Safety Guide[1] "Severe Accident Management Programmes for Nuclear Power Plants" [8] provides recommendations on meeting the requirements of Refs. [9] to [11] for the establishing of an accident management

[1] The IAEA Safety Standards Safety Guides are publications that provide recommendations on different aspects of NPP design and operation. They are governed by the general principles and objectives stated in Safety Fundamentals (Ref. [5]) and safety requirements presented in Safety Requirements publications.

programme to prevent and mitigate the consequences of beyond design basis accidents including severe accidents. The guiding principles for design and operation of NPPs are deterministic requirements with the implications that if deterministic criteria are met, the plant would be safe enough, and the risk of unacceptable radiological releases would be sufficiently low. The PSA technology provides the possibility to assess the risk dealing with a particular NPP. The application of PSA techniques to severe accidents is of particular importance due to very low probability of occurrence of a severe accident, but significant consequences resulting from degradation of the nuclear fuel. To address the need for standardisation of the technical content of PSA the IAEA is developing two new Safety Guides: “Development and Application of Level-1 Probabilistic Safety Assessment for Nuclear Power Plants” [12] and “Development and Application of Level-2 Probabilistic Safety Assessment for Nuclear Power Plants” [13]. The Safety Guide on Level-2 PSA among others applications addresses the use of PSA for identification and evaluation of the measures in place and the actions that can be carried out to mitigate the effects of a severe accident after core damage has occurred.

3.1.2 The general process of development of IAEA Safety Standards

The general process of development of the publications in the IAEA Safety Standards Series foresees several stages that ensure close involvement of Member States, thorough review, and achieving a consensus position. Two safety Guides on PSA have been approved by the Commission on Safety Standards (CSS) in 2009.

3.1.3 The safety guide on severe accident management programme

The Safety Guide on Severe Accident Management Programme published in 2009 [8] provides recommendations on meeting the requirements for accident management, including severe accidents that are established in IAEA Safety Requirements [9] to [11] . The Safety Guide focuses on the development and implementation of severe accident management programmes for NPPs. Although the recommendations of this Safety Guide have been developed primarily for use for light water reactors, they are anticipated to be valid for a wide range of nuclear reactors, both existing and new.

The recommendations of this Safety Guide have been developed primarily for accident management during at-power states; however it is also applicable, in principle, to other modes of operation, including shutdown states. The Safety Guide consists of two main parts that are briefly described below.

3.1.3.1 Concept of the Accident Management Programme

A structured top down approach that should be used to develop the accident management guidance and main principles that should be followed while developing accident management guidance are presented in the Safety Guide. The top down approach should begin with the definition of objectives and strategies, follow a systematic process throughout the development course, and finally result in procedures and guidelines that generally should cover both the preventive and the mitigatory domains.

The Safety Guide presents recommendations to the structure and features of the accident management guidance for different possible domains (*Preventive, Mitigative or both Preventive and Mitigative domains)* and discusses the effective organisation of the accident management process, the roles and responsibilities for the different members of the emergency response organisation at the plant or the utility involved in accident management and communication between members of the emergency response organisation. General recommendations to the upgrade of the equipment that is necessary for the development of a meaningful severe accident management programme and recommendations to the update of the accident management guidance where existing equipment or instrumentation is upgraded are also given in the Safety Guide.

3.1.3.2 Development of an Accident Management Programme

The recommendations to the process of the development and implementation of an accident management programme are presented in the Safety Guide. A brief summary of the key aspects of the process is given below.

Identification of sufficiently comprehensive spectrum of credible beyond design basis accidents (BDBA) is the main goal of the process for the preventive domain. An effective tool achieve this goal is to use insights from Level 1 PSA.

Identification of the full spectrum of credible challenges to fission product boundaries due to severe accidents is the primary task for mitigative domain. Safety Guide recommends to use insights from Level 2 PSA for determination of the full spectrum of challenge mechanisms and to check whether risks are reduced accordingly after the severe accident management guidance has been completed. In view of the inherent uncertainties in determining the credible events, the PSA should not be used a priori to exclude accident scenarios from the development of severe accident management guidance. The Safety Guide considers the following main steps to set up an accident management programme:

1. Identification of plant vulnerabilities to find mechanisms through which critical safety functions may be challenged,
2. Identification of plant capabilities under challenges to critical safety functions and fission product barriers,
3. Development of suitable accident management strategies and measures and,
4. Development of the procedures and guidelines to execute the strategies.

STEP 1 The *identification of plant vulnerabilities* should be based on a comprehensive set of insights on the behaviour of the plant during a beyond design basis accident and severe accident, including identified phenomena that may occur and their expected timing and severity are discussed.

STEP 2 *Plant capabilities* available to fulfill the safety functions, including unconventional line-ups, temporary connections and adaptation of equipment necessary to use these capabilities should be identified. At this process, the capabilities of plant personnel to contribute to unconventional measures to mitigate plant vulnerabilities should be considered.

STEP 3 The *accident management strategies* should be developed for each individual challenge or plant vulnerability in both the preventive and mitigative domains. The development of strategies in the preventive domain should be aimed to preserve safety functions important to prevent core damage, and in the mitigative domain - to enable terminating the progress of core damage once it has started, maintaining the integrity of the containment as long as possible; minimising releases of radioactive material; and achieving a long term stable state. The systematic evaluation and documentation of the possible strategies that can be applied and particular consideration of the strategies that have both positive and negative impacts is essential. The overall goal of this systematic evaluation is to provide the basis for a decision about which strategies constitute a proper response under a given plant damage condition.

STEP 4 *Development of the procedures and guidelines* is the next step of the process. The strategies and measures should be converted to the Emergency Operating Procedures (EOPs) for the preventive domain and to the Severe Accident Management Guidelines (SAMGs) for the mitigative domain. Procedures and guidelines should contain the necessary information and instructions for the responsible personnel, including the use of equipment and associated limitations as well as cautions and benefits. The guidelines should also address the various positive and negative consequences of proposed actions and offer options. Interfaces between the EOPs and the SAMGs should be addressed, and proper transition from EOPs into SAMGs should be provided for, where appropriate. However, where EOPs and SAMGs are executed in parallel it is important that hierarchy between EOPs and SAMGs is established. The recovery of failed equipment and/or recovery from erroneous operator actions that led to a beyond design basis accident or severe accident should be a primary strategy in accident management, and this should be reflected in the accident management guidelines. The Safety Guide recommends that pre-calculated precalculated graphs be developed or to use simple formulas ('computational aids') to avoid the need to perform complex calculations during the accident. It is also recommended to define "rules of usage" for the actual application of SAMGs. The adequate background material that provides the technical basis for strategies must also be presented.

Hardware provisions for accident management (e.g. specific safety systems dealing with accidents) are essential to fulfil the fundamental safety functions (control of reactivity, removal of heat from the fuel, confinement of radioactive material) for beyond design basis accidents and severe accidents. For the new plants there are usually design features present that practically eliminate some severe accident phenomena; however, for existing plants, it may not be possible to develop a meaningful severe accident management programme that would make use of the existing hardware configuration; therefore, modification of the plant should be considered accordingly. Changes in design should also be proposed where uncertainties in the analytical prediction of challenges to fission product barriers cannot be reduced to an acceptable level. Equipment upgrades aimed at enhancing preventative features of the plant should be considered with high priority. For the mitigative domain, when upgrading equipment, the focus should be placed on preservation of the containment functions.

The role of instrumentation and control in the accident management is defined by the ability of the instrumentation to estimate the magnitude of key plant parameters needed for both preventive and mitigative accident management measures. The instrumentation qualified for global conditions may not function properly under local conditions; therefore its failures in severe accident conditions should be identified and methods should be developed which verify that the reading from the dedicated instrument is reasonable. In the development of the SAMGs, the potential failure of important nonqualified instrumentation during the evolution of the accident should be considered and, where possible, alternative strategies that do not use this instrumentation should be developed.

The functions and responsibilities in accident management, in both preventive and mitigative domains, need to be defined within the documentation of the accident management programme. A typical layout of the on-site emergency response organisation is shown in the Safety Guide. The Safety Guide gives detailed recommendations to the responsible persons for the decision making in different domains, and key recommendations to the technical support centre personnel, decision makers and implementers. In addition, the Safety Guide recommends that any involvement of the regulatory body in the decision making process should be clearly defined.

The verification and validation process of all procedures and guidelines is aimed:

- To confirm correctness of the written procedure or guideline,
- To ensure that technical and human factors have been properly incorporated and,
- To confirm that the actions specified in the procedures and guidelines can be followed by trained staff to manage emergency events.

The review of plant specific procedures and guidelines and proper quality assurance programme is an essential part of the process.

An important factor is the education and training. It is recommended that education and training should be given for each group involved in accident management, including the management of the operating organisation and other decision making levels, and, where applicable, safety authority personnel. The training should be in proportion with the tasks and responsibilities of the functions (e.g. in-depth training should be provided for those performing the key functions in the severe accident management programme; others should be trained so that they fully understand the basis of proposed utility decisions). The training programme should be put in place prior to the accident management programme being introduced. The results from exercises and drills should be fed back into the training programme and, if applicable, into the procedures and guidelines as well as into organisational aspects of accident management.

The next point emphasised in the Safety Guide is dealing with processing new information and supporting analysis. This is an essential part of the procedures and guidelines development process. The revisions of EOPs and SAMGs and organisational aspects of accident management should be made for any change in plant configuration or change in background information used in the development of the procedures and guidelines (e.g. update of the PSA that identifies new accident sequences that were not a part of the basis of the existing accident management guidance; new insights from the research on severe accident phenomena).

The key aspects of the analysis of a potential beyond design basis accident or severe accident sequences performed in support for SAMGs are considered in Safety Guide for three consequential steps. In the first step of the analysis a full set of sequences should be analysed that would, without credit for operator intervention in the beyond design basis accident or severe accident domain, lead to core damage (typically identified in the PSA). In the second step - the effectiveness of proposed strategies and their potential negative consequences should be investigated. In the third step of the analysis, once the procedures and guidelines have been developed, they should be verified and validated. It is generally recommended that supporting analysis should be of a best estimate type performed with the appropriate computer codes and a consideration should be given to uncertainties in the determination of the timing and severity of the phenomena.

Several examples and recommendations given for the practical use of severe accident management guidelines and categorisation scheme for accident sequences are presented in the Safety Guide (in Appendixes).

3.1.4 The safety guides on PSA performance and application

The Safety Guides on PSA ([12] and [13]) provide recommendations for performing or managing a Level 1 and Level 2 PSA for a NPP and for using the PSA to support the safe design and operation of NPPs. The recommendations aim to provide technical consistency of PSA studies to reliably support PSA applications and risk-informed decisions.

An additional aim is to promote a standard framework that can facilitate a regulatory or external peer review of a Level 1 and Level 2 PSAs and their various applications. The Safety Guides addresses the necessary technical features of a Level 1 and Level 2 PSAs for NPPs, as well as its applications, based on internationally recognised good practices. This paper briefly describes the Safety Guide on Level 1 PSA and with more details the Safety Guide on Level 2 PSA (with emphasis on application for severe accident management).

3.1.4.1 Safety Guide on Level 1 PSA and Applications

The PSA scope addressed in the Safety Guide [12] includes all plant operational modes (i. e. full power, low power, and shutdown), internal initiating events (i.e. initiating events caused by random component failures and human errors) internal hazards (e.g. internal fires and floods, turbine missiles) and external hazards, both natural (e. g. earthquake, high winds, external floods) and man-made (e.g. airplane crash, accidents at nearby industrial facilities). The Safety Guide is focused on the damage to the reactor core; it does not cover other sources of radioactive material on the site, e. g. the spent fuel pool. However, while considering PSA for low power and shutdown operational modes, the risk from the fuel removed from the reactor is also addressed. The consideration of hazards dealing with malevolent actions is out of the scope of the Safety Guide. In Level 1 PSA aimed at assessing the core damage frequency, the most common practice is to perform the analysis for different hazards and operational modes in separate modules having a Level 1 PSA for full power operating conditions for internal initiating events as a basis. The Safety Guide on Level 1 PSA and applications follows this consideration.

3.1.4.2 Safety Guide on Level 2 PSA and Applications

This Safety Guide [13] includes all the steps in the Level 2 PSA process up to, and including, the determination of the detailed source terms that would be required as input to a Level 3 PSA. Different plant designs use different provisions to prevent or limit the release of radioactive material following a severe accident. Most designs include a containment structure as one of the passive measures for this purpose. The phenomena associated with severe accidents are also very much influenced by the design and composition of the reactor core. The recommendations of this Safety Guide are intended to be technology neutral to the extent possible. However, the number and content of the various steps of the analysis assume the existence of some type of containment structure. General aspects of performance, project management, documentation and peer review of a PSA and implementation of a management system are described in the Safety Guide on Level 1 PSA [12] and are therefore not addressed here. This Safety Guide addresses only the aspects of PSA that are specific to Level 2 PSA. The Safety Guide describes all aspects of the Level 2 PSA that need to be carried out if the starting point is a full scope Level 1 PSA as described in Ref. [12]. The objective of this Safety Guide is to provide recommendations for meeting the requirements of Refs. [9] to [11] in performing or managing a Level 2 PSA project for a NPP. The Safety Guide is structured in accordance with the major tasks as discussed below.

PSA project management and organisation: Specific recommendations relating to the management and organisation of a Level 2 PSA project are provided in the Safety Guide. In particular the following aspects are addressed: definition of the objectives of Level 2 PSA; scope of the Level 2 PSA; project management for PSA; and team selection.

Familiarisation with the plant and identification of aspects important to severe accidents: The aim of this task should be to identify plant systems, structures, components and operating procedures that can influence the progression of severe accidents, the containment response and the transport of radioactive material inside the containment. Safety Guide provides detailed recommendations dealing with acquisition of information important to severe accident analysis.

Interface with Level 1 PSA: grouping of sequences: This task is aimed at establishing the interface between Level 1 and Level 2 PSAs to define plant damage states. The Safety Guide addresses recommendations for plant damage states definition for all initiating events and hazards, and plant operational states. The recommendations on how the existing Level 1 PSA should be expanded to address specific aspects of the Level 2 PSA (when it is an extension of a Level 1 PSA performed originally without the intention to perform a Level 2 or Level 3 PSA) are also provided.

Accident progression and containment analysis: The key recommendations regarding the analysis of containment performance during severe accidents, analysis of the progression of severe accidents, development and quantification of accident progression event trees or containment event trees, treatment of uncertainties, and interpretation of containment event tree quantification results are provided in Safety Guide.

Source terms for severe accidents: The important step in the Level 2 PSA is the calculation of the source terms associated with the end states of the containment event tree. Source terms determine the quantity of radioactive material released from the plant into the environment. Since the containment event trees have a large number of end states, for practical reasons this requires the end states to be grouped into release categories for which the source term analysis is then carried out. Safety Guide gives detailed recommendations for definition of the release categories, grouping of containment event tree end states into release categories, source term analysis, uncertainty evaluation, and interpretation of results of the source term analysis.

Documentation of the analysis: The specific issues related to the presentation and interpretation of results and to organisation of Level 2 PSA documentation are also focused in Safety Guide.

Use and applications of the PSA: The Safety Guide provides the key recommendations for a number of Level 2 PSA applications. The following applications are covered among others: design evaluation; severe accident management; emergency planning; off-site consequences analysis; prioritisation of research.

Three appendixes of the Safety Guide provide an example of a typical schedule for a Level 2 PSA, information on computer codes for severe accidents, and details on the severe accident phenomena.

3.1.4.3 Application of Level 2 PSA for Severe Accident Management

The Safety Guide [13] provides recommendations on the use of Level 2 PSA for the evaluation of the measures in place and the actions that can be carried out to mitigate the effects of a severe accident after core damage has occurred. The aim of mitigative measures and actions should be to arrest the progression of the severe accident or mitigate its consequences by preventing the accident from leading to failure of the reactor pressure vessel or the containment, and controlling the transport and release of radioactive material with the aim of minimising off-site consequences. In particular the Safety Guide recommends to use the results of Level 2 PSA to determine the effectiveness of the severe accident management measures that are described in the severe accident management guidelines or procedures, whether they have been specified using the Level 2 PSA or by any other method. In addition the Safety Guide emphasise that an accident management measure that is aimed at mitigating a particular phenomenon might make another phenomenon more likely due to the fact that the phenomena that occur in the course of a severe accident are highly uncertain and often interrelated. Therefore it is recommended to identify using the Level 2 PSA all interdependencies between the various phenomena that can occur during a severe accident to take them into account in the development of the severe accident management guidelines. Several examples illustrate this statement: depressurisation of the primary circuit may prevent high pressure melt ejection but might increase the probability of an in-vessel steam explosion; introducing water into the containment may provide a cooling medium for molten core material after it has come out of the reactor pressure vessel but might increase the probability of an ex-vessel steam explosion; and operation of the containment sprays may provide a means of removing heat and radioactive material from the containment atmosphere but might increase the flammability of the containment atmosphere by condensing steam. It is also recommended that the updates of the Level 2 PSA and updates of the severe accident management guidelines should be performed in an iterative manner to facilitate the progressive optimisation of the severe accident management guidelines. These recommendations correspond to those, provided in Ref. [8].

3.1.5 INSAG documents

The International Nuclear Safety Group (INSAG) is a group of experts with high professional competence in the field of safety working in regulatory organisations, research and academic institutions and the nuclear industry. INSAG is convened under the auspices of the International Atomic Energy Agency (IAEA) with the objective to provide authoritative advice and guidance on nuclear safety approaches, policies and principles. In particular, INSAG will provide recommendations and opinions on current and emerging nuclear safety issues to the IAEA, the nuclear community and the public.

The list of existing INSAG reports is provided hereafter. Some of these documents (e.g. INSAG-2, 3, 10, 12) provide useful positions on the role of PSA in the Safety of NPP.

INSAG-1: (revised as INSAG-7): Summary Report on the Post-accident Review Meeting on the Tchernobyl Accident
INSAG-2: Radionuclide Source Terms from Severe Accidents to Nuclear Power Plants with Light Water Reactors
INSAG-3: (revised as INSAG-12): Basic Safety Principles for Nuclear Power Plants
INSAG-4: Safety Culture
INSAG-5: The Safety of Nuclear Power
INSAG-6: Probabilistic Safety Assessment
INSAG-7: The Tchernobyl Accident: Updating of INSAG-1
INSAG-8: A Common Basis for Judging the Safety of Nuclear Power Plants Built to Earlier Standards
INSAG-9: Potential Exposure in Nuclear Safety
INSAG-10: Defence in Depth in Nuclear Safety
INSAG-11: The Safe Management of Sources of Radiation: Principles and Strategies
INSAG-12: Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev.1
INSAG-13: Management of Operational Safety in Nuclear Power Plants
INSAG-14: Safe Management of the Operating Lifetimes of Nuclear Power Plants
INSAG-15: Key Practical Issues in Strengthening Safety Culture
INSAG-16: Maintaining Knowledge, Training and Infrastructure for Research and Development in Nuclear Safety
INSAG-17: Independence in Regulatory Decision Making
INSAG-18: Making Change in the Nuclear Industry: The Effects on Safety
INSAG-19: Maintaining the Design Integrity of Nuclear Installations Throughout Their Operating Life
INSAG-20: Stakeholder Involvement in Nuclear Issues
INSAG-21: Strengthening the Global Nuclear Safety Regime
INSAG-22: Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles
INSAG-23: Improving the International System for Operating Experience Feedback
INSAG-24: The Interface between Safety and Security at Nuclear Power Plants

3.1.6 Related IAEA services

The IAEA mandate authorises the IAEA to develop Safety Standards and to provide support for the application of these standards. A number of Services are made available by the IAEA for the Member States; amongst them there are also those related to severe accident management and Level 2 PSA.

The IAEA RAMP service is an activity to support individual Member States with the Review of Accident Management Programmes at their plants. Review of AM programme at particular plant is performed on request by a Member State. The review team usually includes four experts plus an IAEA staff-member. The review focuses on studying the relevant documents, interviews with plant staff and regulators. The output of the review is a detailed report with assessment and recommendations for the improvements/refinements to the existing Accident Management Programme. IAEA has prepared a manual in support of RAMP service [16] that contains a detailed questionnaire for the self assessment of the existing accident management programme. The following topics are covered in the manual:

- Selection and definition of AMP,
- Accident analysis for AMP,

- Assessment of plant vulnerabilities,
- Development of severe accident management strategies,
- Evaluation of plant equipment and instrumentation,
- Development of procedures and guidelines,
- Verification and validation of procedures and guidelines,
- Integration of AMP and plant Emergency Arrangements,
- Staffing and qualification,
- Training needs and performance.
- AM Programme revisions.

Several successful RAMP missions have been already conducted during which extensive review activities have been performed, feedback has been provided, and findings have been discussed with the plant specialists. A formal review report was produced by the IAEA and forwarded to the counterpart.

Numerous workshops, training seminar and expert missions were provided by IAEA to China, Romania, Russia, Ukraine, Pakistan, Slovakia, Lithuania, etc. before the RAMP mission. The first RAMP mission was held at Krisko NPP in Slovenia in 2001, and other missions to Chinese PWR in China and Ignalina NPP in Lithuania were also conducted in 2006 and 2007, respectively. In 2009 the RAMP was performed for KANUPP (Pakistan). So far the mission has been conducted for PWR, PHWR and RBMK. The RAMP for Cernavoda NPP (Romania) etc are expected for future service.

- For Ignalina NPP, several design modifications (core exit temperature measurement and an additional shutdown system) were made during the establishment of SAMP. It is the first SAMGs for RBMK reactors. It will therefore constitute a source of valuable information for other RBMK reactors.
- For Krisko NPP, assessing the possible impact of non-uniform hydrogen distribution and of the adequacy of the hydrogen source term and reconsidering the availability of the systems due to their potential failure during scenarios dominating core damage frequency were recommended during the mission.

An International Probabilistic Safety Assessment Review Team (IPSART) service was established in 1988. The dedicated guideline [17] is used to conduct the review missions. A Review of PSAs for plants from different countries, of various designs, and all PSA levels, hazard scopes, and operational modes is performed on specific request submitted to the IAEA by the Member State. Depending on the scope of the PSA the review duration is 1 to 2 weeks and the review team composition is from four to seven international independent experts plus an IAEA staff-member. The review focuses on the check of methodological aspects, completeness, consistency, coherence, etc. of the PSA. The output of the review is the IPSART Mission Report that describes the review performed, the review findings, the technical aspects of the PSA study, strengths and limitations, and provides suggestions and recommendations for improvement of the PSA quality and its sound use for enhancing plant safety and risk management applications.

The IPSART service helps to achieve high quality of PSA and therefore assists in further enhancing the nuclear safety. More than 60 IPSART mission have been conducted so far in many countries all around the world helping to achieve high quality PSA and to transfer advanced methodology and knowledge in nuclear safety assessment.

3.1.7 Conclusions

The IAEA has developed a comprehensive set of new Safety Standards including Safety Guides for Level 1 and Level 2 PSAs and severe accident management. The Safety Guides provide a common standardised platform for safety assessment and severe accident management that represent widely accepted good practices and consensus amongst Member States. These publications will promote a consistent development of the severe accident management programme, and development, application and review of PSA studies, as well as the use of PSA results and insights in different applications, including application for severe accident programme development.

3.1.8 References

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[6] INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Defence in Depth in Nuclear Safety, INSAG-10, IAEA, Vienna, 1996.

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[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Operation, IAEA Safety Standards Series No. NS-R-2, IAEA, Vienna, 2000.

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA, GSR-4, Vienna, 2009.

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level-1 PSA, IAEA, Vienna (DS394, in preparation).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level-2 PSA, IAEA, Vienna (Specific Safety Guide SSG-4).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna, 2006.

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna, 2006.

[16] NTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for the Review of Accident Management Programmes in Nuclear Power Plants, Services Series No. 9, IAEA, Vienna, 2003.

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, IPERS guidelines for the international peer review service. TECDOC 832. Second Edition, IAEA, Vienna, 1995.

3.2 OECD/NEA/CSNI REFERENCE DOCUMENTS AND ACTIVITIES

Many collaborative actions related to severe accident and L2PSA are conducted through the OECD/NEA, especially by the CSNI Risk and GAMA working groups. The present chapter provides some of the recent references that may be of key importance for the development of L2PSAs. It is of course highly recommended to connect the development of a NPP L2PSA to the international experience shared through the OECD activities.

Table 1 OECD references on severe accidents, severe accident management and Level 2 PSA

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NEA/CSNI/R(2000)10. Carbon Monoxide - Hydrogen Combustion Characteristics in Severe Accident Containment Conditions.
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NEA/CSNI/R(2000)8. Impact of Short-Term Severe Accident Management Actions in a Long-Term Perspective.
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NEA/CSNI/R(2000)19. Technical Notes on Ex-vessel Hydrogen Sources.
NEA/CSNI/R(2000)18R. Proceedings of the Workshop on Ex-vessel Debris Coolability, 15-18 November, 1999, Karlsruhe,

Germany.
NEA/CSNI/R(2001)5. Status of Degraded Core Issues - Synthesis Paper, October 2000.
NEA/CSNI/R(2001)7. Severe Accident Management - Operator Training and Instrumentation Capabilities, Proceedings, 12-14 April 2001, Lyon, France.
NEA/CSNI/R(2001)16R. Severe Accident Management - Workshop on Operator Training and Instrumentation Capabilities, Summary and Conclusions, 12-14 March 2001, Lyon, France.
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NEA/CSNI/R(2007)16 Recent Developments in Level 2 PSA and Severe Accident Management.
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NEA/CSNI/R(2009)3 Ability of Current Advanced Codes to Predict Core Degradation, Melt Progression and Reflooding - Benchmark Exercise on an Alternative TMI-2 Accident Scenario.
NEA/CSNI/R(2009)16 Probabilistic Risk Criteria and Safety Goals

Note: **R** at the end of the report code means that the report has a limited distribution.

3.2.1 Technical Opinion Paper on Level 2 PSA

A significant publication is the Technical Opinion Paper (TOP) on Level 2 PSA [18].

The CSNI TOPs are short statements giving a summary and a position of WGRISK concerning an important topic, generally written after a State-of-the-Art Report or after a Workshop. The Level 2 PSA TOP was published in 2007 and its conclusion is recalled hereafter.

"The main message of this Technical Opinion Paper is that the Level 2 PSA methodology may now be seen as mature. This is reflected by the large number of high quality analyses that have been performed in recent years and used to identify the potential vulnerabilities to severe accidents and the accident management measures that could be implemented.

The Level 2 PSA is now seen as an essential part of the safety analysis that is carried out for all types of nuclear power plants worldwide. The information provided by the Level 2 PSA is being used by plant operators and Regulatory Authorities as part of a risk informed decision making process on plant operation and more specifically on issues related to severe accident management.

A consistent framework has been established with the development of the individual components of the Level 2 PSA methodology and guidance has been produced by international organisations for carrying out the analysis. In practice, however, there are still differences in the approach and the level of detail in the individual steps that have been carried out in different analyses, partly due to the different objectives that have been defined for these studies. Quality standards and guidelines are currently being developed for Level 2 PSA which should address many of these differences.

The acceptability of the methodology since the early studies in the 1980s is due largely to the significant progress made in the understanding of severe accident and source term phenomenology and in the model development in the current generation of integrated severe accident analysis codes. The research and development activities have continued internationally, albeit at a reduced scale, with emphasis on improving the state of knowledge and providing further data for model validation and improvement.

Further development in Level 2 PSA is likely to see its integration within a Living PSA and its use for risk-informed applications. This requires improvement in the Level 2 PSA methodology in a number of areas, including: the Level 1/ Level 2 PSA interface, the modelling of safety system recovery and human reliability analysis.

The epistemic uncertainty related to some Level 2 PSA issues is regarded as being quite large. The impact of this on risk-informed decision making will also require further consideration of uncertainty treatment in a more integrated manner.

Finally, given the role that integrated severe accident codes (supported by research) have played in the acceptance of Level 2 PSA, future Level 2 PSA research and development activities should be aimed at making these codes play a more central and integral role in the PSA quantification process. Such a shift is likely to alter (and quite possibly diminish) the role of expert judgement and phenomenological event tree modelling in the quantification."

3.2.2 Probabilistic Risk Criteria and Safety Goals

Another important document for the ASAMPSA2 project is the NEA/CSNI report on "Probabilistic Risk Criteria and Safety Goals" [19]. Some extracts of the executive summary has been reproduced hereafter:

"*Probabilistic Safety Criteria, including Safety Goals, have been progressively introduced by regulatory bodies and utilities. They range from high level qualitative statements (e.g., "The use of nuclear energy must be safe") to* *technical criteria (e.g., probability of fuel cladding temperature being higher than 1204 °C).They have been published in different ways, from legal documents to internal guides. They can be applied as legal limits (not meeting them is an offence) down to "orientation values".*

The questionnaire produced for this tasks requested information on the above issues, with added questions on the basis for the criteria, the way they are applied and experience on their use.

Answers have been received from 13 nuclear safety organizations (Canada, Belgium, Chinese Taipei, Finland, France, Hungary, Japan, Korea, Slovakia, Sweden, Switzerland, UK and USA) and 6 utilities (Hydro-Québec, Fortum, OKG, Ontario-Power-Generation, Ringhals and TVO). Two of the regulatory bodies (Belgium and Chinese Taipei) declared they have not set (and do not intend to set) any Probabilistic Safety Criterion. Some supplementary information (three countries) has been taken from a questionnaire on Safety Goals during the 20-24 November 2006 IAEA Technical Meeting on the development of draft DS-394. This report is based on information given in the annexed questionnaire. More information that could be found in other CSNI reports is not considered here.

The reported Probabilistic Safety Criteria can be grouped into 4 categories, in relation with the tools to be used for assessing compliance:

- *Core Damage Frequency (CDF) - Level 1 PSA - 16 respondents.*
- *Releases Frequency (LERF, LRF, SRF) - Level 2 PSA - 14 respondents.*
- *Frequency of Doses - Level 3 PSA - 4 respondents.*
- *Criteria on Containment Failure - System level - 2 respondents.*

Several respondents use more than one criterion (e.g., CDF and LERF) while some others use a range of values for a given criterion (e.g., frequency of doses to the public, to the workers, during accidents, during normal operations).

While originally set considering the state of the art of PSA, the CDF criterion is presently considered as based on Defence-In-Depth. Also, the Criteria on Containment Failure, newly introduced in Japan and USA, is an expression of Defence-In-Depth as new designs could meet the LERF without taking containment into account.

Releases Frequency and Frequency of doses address public safety. However, while the frequency of doses addresses directly public health, Releases Frequency considers that public safety is achieved for a given release (within a given time for LERF), taking into account Emergency Measures (such as evacuation).

The values associated with CDF vary from 5 E-4 per year to 1 E-5 per year. When indicated, this spread is reduced when considering new plants where all respondents but 2 set the CDF to 1 E-5.

The values associated to releases frequency show a wider spread, from 1 E-5 per year to 1 E-7 per year. As for the CDF, the spread is reduced when considering new plants, where all respondents but one set the LRF (or LERF) to 1 E-6 per year. It has to be noted that the results are highly related to the scope and detail of the reference PSA, so the numerical values cannot be compared without a complete definition of the scope covered by the PSA.

Generally, all respondents considered introduction of Probabilistic Safety criteria resulted in safety improvements.

Opinion is widespread on the benefits of using Probabilistic Safety Criteria for communication with the public, ranging from bad to good experiences. It seems that there is a strong relation with each country culture and the circumstances.

The responses to the questionnaires suggested that more work should be considered in the definition of Releases Frequencies: some regulators include a time range (generally 24 hours) in the criterion while others do not limit the time to be considered. It is suggested that, in the first case, the existing PSAs should be revisited to assess if long development accident sequences were considered."

3.2.3 References

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3.3 EU REFERENCES DOCUMENTS

3.3.1 WENRA

The WENRA (Western European Nuclear Regulator's Association) is a network of Chief Regulators of EU countries with nuclear power plants and Switzerland as well as of other interested European countries which have been granted observer status. The main objectives of WENRA are to develop a common approach to nuclear safety, to provide an independent capability to examine nuclear safety in applicant countries and to be a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues.

Two WENRA documents are particularly important in the context of L2PSA development and applications, because they precise the orientations defined by the European Safety Authorities:

- The Reactor Safety Reference Levels [20],
- The Safety Objectives for new Power Reactors [21].

The first document defines some Safety Reference Levels that are supposed to be demanding for the existing reactors. Concerning the Chapter O ("Probabilistic Safety Analysis"), the following Safety Reference Levels have been defined [20].

« 1. Scope and content of PSA

1.1 For each plant design, a specific PSA shall be developed for level 1 and level 2 including all modes of operation and all relevant initiating events including internal fire and flooding. Severe weather conditions and seismic events shall be addressed[2].

1.2 PSA shall include relevant dependencies[3].

1.3 The basic Level 1 PSA shall contain sensitivity and uncertainty analyses. The basic Level 2 PSA shall contain sensitivity analyses and, as appropriate, uncertainty analyses.

1.4 PSA shall be based on a realistic modelling of plant response, using data relevant for the design, and taking into account human action to the extent assumed in operating and accident procedures.

1.5 Human reliability analysis shall be performed, taking into account the factors which can influence the performance of the operators in all plant states.

2. Quality of PSA

2.1 PSA shall be performed, documented, and maintained according to requirements of the management system of the licensee.

2.2 PSA shall be performed according to an up to date proven methodology, taking into account international experience currently available.

3. Use of PSA

3.1 PSA shall be used to support safety management. The role of PSA in the decision making process shall be defined.

2 This means that these two hazards shall be included in the PSA, except if a justification is provided for not including them, based on site-specific arguments on these hazards or on sufficient conservative coverage through deterministic analyses in the design, so that their omission from the PSA does not weaken the overall risk assessment of the plant.

3 Such as functional dependencies, area dependencies (based on the physical location of the components) and other common cause failures

3.2 *PSA shall be used[4] to identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant.*

3.3 *PSA shall be used to assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff-edge effects"[5].*

3.4 *PSA shall be used to assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational occurrences.*

3.5 *Insights from PSA shall be used as input to development and validation of the safety significant training programmes of the licensee, including simulator training of control room operators.*

3.6 *The results of PSA shall be used to ensure that the items are included in the verification and test programmes if they contribute significantly to risk.*

4. Demands and conditions on the use of PSA

4.1 *The limitations of PSA shall be understood, recognised and taken into account in all its use. The adequacy of a particular PSA application shall always be checked with respect to these limitations.*

4.2 *When PSA is used, for evaluating or changing the requirements on periodic testing and allowed outage time for a system or a component, all relevant items, including states of systems and components and safety functions they participate in, shall be included in the analysis.*

4.3 *The operability of components that have been found by PSA to be important to safety shall be ensured and their role shall be recorded in the SAR. »*

The second document on the Safety Objectives for new Power Reactors ([21], which is a draft for external review) indicates that:

"These "Safety Reference Levels" were designed to be demanding for existing reactors. However, in line with the continuous improvement of nuclear safety that WENRA members aim for, new reactors are expected to achieve higher levels of safety than existing ones, meaning that in some safety areas, fulfilment of the "Safety Reference Levels" defined for existing reactors may not be sufficient.

Hence, it has been considered timely for WENRA to define and express a common view on the safety of new reactors, so that:

- *new reactors to be licensed across Europe in the next years offer improved levels of protection compared to existing ones;*
- *regulators press for safety improvements in the same direction and ensure that these new reactors will have high and comparable levels of safety;*
- *applicants take into account this common view when formulating their regulatory submissions.*

In addition, this common view could provide insights for the periodic safety reviews of existing reactors."

The following safety objectives (linked to PSAs) are proposed:

"Compared to currently operating reactors, new ones are expected to be designed, sited, constructed, commissioned and operated with the objectives of:

O1. Normal operation, abnormal events and prevention of accidents

- *reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation;*
- *reducing the potential for escalation to accident situations by enhancing plant capability to control abnormal events.*

O2. Accidents without core melt

4 It is intended that such analyses will be done on a continuous basis, not just every ten years during the Periodic Safety Review.

5 Small deviations in the plant parameters that could give rise to severely abnormal plant behaviour

- *ensuring that accidents without core melt[6] induce[7] no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation[8]);*
- *reducing, as far as reasonably achievable:*
    - o *the core damage frequency taking into account all types of hazards and failures and;*
    - o *combinations of events;*
    - o *the releases of radioactive material from all sources;*
- *providing due consideration to site and design to reduce the impact of all external hazards[9] and malevolent acts.*

O3. Accidents with core melt

- *reducing potential radioactive releases to the environment from accidents with core melt, also in the long term[10], by following the qualitative criteria below:*
- *accidents with core melt which would lead to early[11] or large[12] releases have to be practically eliminated[13];*
- *for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures. (…)*

Regarding the quantitative safety targets to drive the compliance with proposed safety objectives, the WENRA document provides the following comments:

"*The RHWG considers that there is merit for countries to use quantitative safety targets along with the proposed qualitative safety objectives. As safety targets, these values are useful to drive in-depth technical discussions with the applicants aimed at identifying real safety improvements, rather than being used as stand-alone acceptance criteria.*

Candidate quantitative safety targets to drive compliance with the proposed safety objectives are discussed below. However, no consensus values were identified at this stage. The RHWG emphasises the need to be aware of differences in methodologies as well as terminology when making comparisons between numerical results in different countries.

Normal operation, abnormal events and prevention of accidents (O1)

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[6] *For new reactors, the scope of the defence-in-depth has to cover all risks induced by the nuclear fuel, even when stored in the fuel pool. Hence, core melt accidents (severe accidents) have to be considered when the core is in the reactor, but also when the whole core or a large part of the core is unloaded and stored in the fuel pool.*

[7] *in a deterministic and conservative approach with respect to the evaluation of radiological consequences.*

[8] *However, restriction of food consumption could be needed in some scenarios.*

[9] *As defined in Reference Level E 5.2., January 2008 version*

[10] *Long term: considering the time over which the safety functions need to be maintained. It could be months or years, depending on the accident scenario.*

[11] *early releases: situations that would require off-site emergency measures but with insufficient time to implement them.*

[12] *large releases: situations that would require protective measures for the public that could not be limited in area or time*

[13] *In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA NSG1.10).*

Safety indicators on abnormal event occurrences are sometimes used for the supervision of operating nuclear power plants.

No reference numerical value having practical application for improving safety of new reactors as regards objective O1 was identified among WENRA countries. However, RHWG recommends European licensees to have their own ambitious quantitative safety targets[14] on the reliability of systems and components involved in normal operation.

The compliance with the qualitative safety objective O1 is expected to be appreciated through:

- *the demonstration that all operational experience feedback has been used to identify the safety issues of existing plants that could be relevant for the envisaged new design;*
- *the verification that appropriately validated means have been designed to address these issues;*
- *the implementation of extended operational margins.*

Accidents without core melt (O2)

Reducing the core damage frequency

WENRA countries already make a large use of level 1 PSA and widely refer to the core damage frequency (CDF) as a probabilistic safety target for currently operating plants. Some WENRA countries refer to a CDF target less than 10-5 per year for new reactors. This is in line with INSAG-12 recommendations, which state that the CDF target for new reactors should be reduced by a factor of at least ten compared to the target for existing ones (10-4 per year as recommended by INSAG), all plant states and all types of initiating events being taken into account.

However, two arguments were put forward not to adopt such a common target:

- *in some countries, this value is considered as being already reached by some existing reactors;*
- *the methodologies to calculate the CDF may differ from one country to another.*

No or only minor off-site radiological impact

(…) A significant number of WENRA countries use dose / frequency criteria as design targets.

To achieve the objective O2, it is expected that off-site radiological impact of accidents without fuel melt is less than the intervention levels for iodine prophylaxis, sheltering and evacuation.

These intervention levels, which are used in the 5th level of the defence in depth, have already been enforced by EU members in their national regulation to comply with Directive 96/29/Euratom - 13 may 1996 - article 50.2., and are consistent with the ICRP recommendations. For instance, in ICRP-63, the intervention level for sheltering is 5-50 mSv in 2 days.

Design targets should be set below these intervention levels.

Accidents with core melt (O3)

Practical elimination

The possibility of certain accident conditions to occur can be considered as practically eliminated "if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise".[15].

As regards conditions that can not be physically excluded, it must be underlined that a justification for extreme unlikelihood has to be provided with high confidence. This means that the practical elimination of a condition cannot be claimed solely based on compliance with a general cut-off probabilistic value. Even if the probability of a condition is very low, any additional reasonable design features to lower the risk should be implemented.

[14] *Not to be mistaken with a plant availability criterion for electricity production.*

[15] IAEA document NS-G-1.10, para 6.5, footnote 14.

The justification should include demonstration that there is sufficient knowledge of the accident condition analysed and of the phenomena involved (e.g. DCH, steam explosion, hydrogen behaviour).
Furthermore, uncertainties associated with the data and methods should be quantified.

Limited protective measures in area and time

Regarding radiological criteria associated with core melt accidents, a significant number of WENRA countries use release / frequency criteria. Some WENRA countries refer to Caesium release criteria in case of a severe accident. The aim of such criteria is to require that accidents have a limited impact on food consumption and land use. However, it is not easy to make a link between a relevant numerical value for Cs releases and the safety objective O3.

To achieve the objective O3, it is expected that the off-site radiological impact of accidents with core melt only leads to limited protective measures in area and time (no permanent relocation, no long term restrictions in food consumption, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering).

These protective measures are associated with intervention levels, which are used in the 5th level of the defence in depth. Such intervention levels have already been enforced by EU members in their national regulation to comply with Directive 96/29/Euratom - 13 may 1996 – article 50.2., and are consistent with the ICRP recommendations. For instance, in ICRP-63, the intervention level for sheltering is 5-50 mSv in 2 days.

Considering these intervention levels, design targets should be set so that only limited protective measures in area and time are needed. These design targets should take due account of the uncertainties associated with the use of best estimate methodologies for core melt accidents. (…)"

3.3.2 European utilities requirement for LWR reactors (EUR)

The European electricity producers involved in the making of the European Utility Requirements (EUR) document aim at harmonisation and stabilisation of the conditions in which the standardised LWR nuclear power plants to be built in Europe in the first decades of the century will be designed and developed. This is expected to improve both nuclear energy competitiveness and public acceptance in an electricity market unified at European level. Beyond Europe, the EUR utilities also promote world-wide harmonisation of the design bases of the next nuclear power plants.

The EUR ([22], Revision C 2001) includes some Probabilistic Safety Targets that may be taken into account by the L2PSA analyst. Some extracts are provided here:

Probabilistic targets

"The design shall meet the following probabilistic design targets:

- *a Core Damage cumulative frequency of less than 10^{-5} per year and;*
- *a cumulative frequency of less than 10^{-6} per year of exceeding the Criteria for Limiting Impact*;*
- *a significantly lower cumulative frequency to get either earlier or much larger releases.*

These targets are broadly in line with the developing consensus as expressed, for example, in the IAEA document INSAG-3. They are aimed at achieving an acceptable level of risk to the public and limiting the extent of offsite measures in the case of Severe Accidents. The targets are considered to represent a good balance between accident prevention and mitigation.*

These frequency Targets shall include shutdown states which have been shown to be a significant contributor in assessments of present reactor designs."*

Release targets for Severe Accidents

« Thresholds of activity release into the atmosphere are given in the EUR document that shall be used as criteria for *Severe Accidents* and PSA studies. They are referred by Criteria for Limiting Impact* (CLI) in the EUR document.*
The CLI thresholds are set in order to limit the societal consequences resulting from effects on public health and contamination of soil and water. The following objectives have been included in the criteria:
Three objectives that support simplification of the emergency planning and off-site countermeasures:

- *minimal Emergency Protection Action* beyond 800 m from the reactor during early releases from the containment;*
- *no Delayed Action* (temporary transfer of people) at any time beyond approximately 3 km from the reactor;*
- *no Long Term Action*, involving permanent (longer than 1 year) resettlement of the public, at any distance beyond 800 m from the reactor.*

A fourth objective deals with limitation of the potential economic impact of a severe accident. Restriction on the consumption of foodstuff and crops should be limited in terms of timescale and ground area. The fourth component of the CLI is related only to the potential economic impact of a Severe Accident and to public acceptance. It is not related to the safety of the public, which is assured by the implementation of the national and international rules and standards on trade restrictions for contaminated food.
The following tables provide the numerical data associated to the four Criteria for Limiting Impact.

Table 2 Coefficients for Criterion for Limited Impact for no Emergency Action beyond 800m from the reactor

Isotope group	Coefficients for ground level releases C_{ig}	Coefficient for elevated releases C_{ie}
Xe_{133}	$6,5.10^{-8}$	$1,1.10^{-8}$
I_{131}	$5,0.10^{-5}$	$3,1.10^{-6}$
Cs_{137}	$1,2.10^{-4}$	$5,4.10^{-6}$
Te_{131m}	$1,6.10^{-4}$	$7,6.10^{-6}$
Sr_{90}	$2,7.10^{-4}$	$1,2.10^{-5}$
Ru_{103}	$1,8.10^{-4}$	$8,1.10^{-6}$
La_{140}	$8,1.10^{-4}$	$3,7.10^{-5}$
Ce_{141}	$1,2.10^{-3}$	$5,6.10^{-5}$
Ba_{140}	$6,2.10^{-6}$	$3,1.10^{-7}$

The acceptance criterion for the criterion for limited impact for no emergency action beyond 800 m from the reactor is that:

$$\sum_{i=1}^{9} R_{ig} \cdot C_{ig} + \sum_{i=1}^{9} R_{ie} \cdot C_{ie} < 5 \cdot 10^{-2}$$

R_{ig} and R_{ie} (expressed in TBq) are the cumulated releases respectively for ground level and elevated releases during the first 24 hours after the initiation of the Design Extension Condition (DEC). C_{ig} and C_{ie} can be found in Table 2.

The acceptance criterion for the criterion for limited impact for no delayed action beyond 3 km from the reactor is that:

$$\sum_{i=1}^{9} R_{ig} \cdot C_{ig} + \sum_{i=1}^{9} R_{ie} \cdot C_{ie} < 3 \cdot 10^{-2}$$

R_{ig} and R_{ie} (expressed in TBq) are the cumulated releases respectively for ground level and elevated releases during the first 4 days after the initiation of the DEC. C_{ig} and C_{ie} can be found in Table 3.

Table 3 Coefficients for Criterion for Limited Impact for no Delayed Action beyond 3 km from the reactor Isotope Group

Isotope group	Coefficients for ground level releases C_{ig}	Coefficient for elevated releases C_{ie}
Xe_{133}	0	0
I_{131}	$1,2.10^{-6}$	$3,5.10^{-7}$
Cs_{137}	$5,6.10^{-6}$	$8,9.10^{-7}$
Te_{131m}	$3,8.10^{-6}$	$7,0.10^{-7}$
Sr_{90}	$9,9.10^{-7}$	$3,2.10^{-7}$
Ru_{103}	$1,3.10^{-6}$	$2,2.10^{-7}$
La_{140}	$2,9.10^{-6}$	$4,8.10^{-7}$
Ce_{141}	$4,5.10^{-6}$	$8,1.10^{-7}$
Ba_{140}	$1,5.10^{-6}$	$2,5.10^{-7}$

Table 4 Coefficients for Criterion for Limited Impact for no Long Term Actions beyond 800 m from the reactor

Isotope group	Coefficients for ground level releases C_{ig}	Coefficient for elevated releases C_{ie}
Xe_{133}	0	0
I_{131}	$1,2.10^{-5}$	$7,8.10^{-7}$
Cs_{137}	$6,5.10^{-5}$	$3,4.10^{-5}$
Te_{131m}	$2,6.10^{-5}$	$1,3.10^{-6}$
Sr_{90}	$1,4.10^{-5}$	$7,2.10^{-7}$
Ru_{103}	$2,3.10^{-5}$	$1,2.10^{-7}$
La_{140}	$7,9.10^{-5}$	$4,1.10^{-6}$
Ce_{141}	$7,6.10^{-5}$	$4,0.10^{-6}$
Ba_{140}	$1,1.10^{-5}$	$5,9.10^{-7}$

The acceptance criterion for the criterion for limited impact for no long term action beyond 800 m from the reactor is that:

$$\sum_{i=1}^{9} R_{ig} \cdot C_{ig} + \sum_{i=1}^{9} R_{ie} \cdot C_{ie} < 1 \cdot 10^{-1}$$

R_{ig} and R_{ie} (expressed in TBq) are the cumulated releases respectively for ground level and elevated release. C_{ig} and C_{ie} can be found in Table 4.

Reference Source Term

"The reference Severe Accident shall be design-specific, since it is required to be a mechanistic sequence which is treated realistically. Therefore Best Estimate Analysis shall be considered.

Before PSA is finalised, engineering judgement may be used to identify the adequate reference sequence.
The identification of the reference Severe Accident for the determination of the RST shall be made among those Severe Accidents with higher contribution to Core Damage frequency. One reference Severe Accident shall be selected, as that sequence which leads to the most representative Source Term among the Severe Accident sequences with higher contribution to Core Damage frequency.
The term "most representative" is used in the sense that the reference Source Term should bound the releases associated to the dominant, from Core Damage frequency point of view, Severe Accident sequences.
In the hypothetical case that the second probabilistic target (cumulative frequency of exceeding the CLI) would be met without any mitigation feature, at least one sequence shall be selected for the RST identification.
If the Core Damage frequency would be lower than 10-6 per year, and therefore the second probabilistic Target (cumulative frequency of exceeding the CLI) would be already met, the most representative low-pressure Severe Accident shall be selected for RST identification. »

Required applications of RST

The RST shall be used by the Designer as the reference for design purposes such as:

- *demonstration of the capability of equipment to survive the environmental conditions associated with a Severe Accident and to still operate as required;*
- *evaluations of dose to control room Operators and in all other locations where Operator activities may be required;*
- *definition of equipment and system design requirements;*
- *verification of compliance with the plant release Targets.*

PSA evaluation of Source Term

On the basis of Level 2 PSA, releases associated with each sequence family shall be assessed. The Designer shall compare each of these releases with that associated with the RST. Cases where the release exceeds the RST release shall be reported and explained for sequence families with probabilities in the range of 10^{-7} per year and higher.
These sequences should be binned in families according, at least, to the mode and time of the postulated containment failure. PSA calculations might show that some particular values considered in the RST are exceeded. If these Deviations are minor for the design purposes mentioned in the previous paragraph, the RST should not be re-evaluated.
The use of the RST for checking design compliance with the release limits is intended only as a provisional assessment, where PSA identifies other sequences above the 10-7 per year cut-off. The RST remains the design-verification value if all PSA Severe Accident sequences families are below the probabilistic cut-off (10-7 per year). The cumulative probability of all sequences that exceed the RST releases or are not evaluated shall be less than 10-6 per year. Otherwise either the RST shall be revised or a design modification shall be introduced.

3.3.3 The Severe Accident Research NETwork of Excellence (SARNET)

In the European context, the Severe Accident Research NETwork of Excellence (SARNET, [23]) gathers a large part of activities concerning severe accident issues. A first project was initiated in 2004 with 51 organisations involved in severe accident research in Europe plus Switzerland and Canada. A second project, started in 2009, gathered 41 organisations from 21 countries (Europe plus Switzerland, Canada, USA and Korea).

The objective is to perform the common research programmes defined in the network first phase and to continue to improve the common computer tools and methodologies for NPP safety assessment. It will consolidate the sustainable integration of the European SA research capacities. These research programmes essentially concern the six highest priority safety issues that were identified after ranking in the first phase of the network: in-vessel core coolability, molten core-concrete interaction, fuel-coolant interaction, hydrogen mixing and combustion in containment, impact of oxidising conditions on source term, and iodine chemistry. The SARNET Joint Programme of Activities includes the following main tasks:

- Performing new experiments on the above mentioned issues and jointly analysing their results to elaborate a common understanding of the concerned physical phenomena,
- Continuing the development and assessment of the ASTEC integral computer code (jointly developed by IRSN and GRS to predict the NPP behaviour during a postulated SA), which capitalises the knowledge produced in the network for its models. In particular, efforts are being extended to its applicability to BWR and CANDU NPP types,
- Continuing the storage of SA experimental results in a scientific database, based on the STRESA JRC tool,
- Promoting educational and training courses, ERMSAR (European Review Meeting on Severe Accident Research) international conferences (to be held once a year) and mobility of young researchers or students between the various European organisations.

Activities concerning L2PSA were performed within the first project in 2004-2008 (general methodology, uncertainties assessment and dynamic reliability methods, [24]) and have been used to define and initiate the ASAMPSA2 project of the 7th EC Framework Programme that has produced the current guideline.
A detailed presentation of SARNET outcomes during the first phase of the project can be found in [25]. Other references on SARNET are provided in Refs. [26] to [34]).

Technical exchanges between SARNET and L2PSA2 analysts are crucial for updating the knowledge of severe accident physical phenomena, not only in the L2PSA modelling but also on the L2PSA requirements for computer codes such as ASTEC.

3.3.4 Nordic nuclear safety research (NKS) and Nordic PSA Group (NPSA) – Safety goals

Research activities are also conducted within the Nordic Nuclear Safety Research (NKS) and the Nordic PSA group (NPSA). A recent and still on-going project concerns the probabilistic safety goals ([35], [37], [38]). This project aims to provide the status, concepts and history of probabilistic safety goals for nuclear power plants and to provide some guidance for their definitions and applications.
Reference [35] gives a general definition related to risk that have been reproduced hereafter.

"Probability and risk concepts
Probability expresses quantitatively the uncertainty related to an event. Mathematically, it is a measure that assigns a number [0,1] to a subset of a given set, and it follows the axioms of the probability theory. In practical application, the interpretation of a subset can be an event, so that the assigned probability represents the uncertainty of the event.
When using probabilities and probability models in decision making, it is important to agree with the interpretation of the probability. The two main interpretations are the <u>subjective interpretation</u> (also called <u>Bayesian</u>), and the <u>frequency interpretation</u>.
According to the frequency interpretation, the probability of an event is the relative frequency with which the event occurs in an infinitely long experiment. This means that the probabilities cannot be known exactly, since in practice

there are no infinite series of experiments. However, the frequency interpretation makes it possible to estimate probabilities and to determine confidence bounds for unknown probabilities.

According to the subjective or Bayesian interpretation, probability is a rational degree of belief about the occurrence of an event. The probability depends on the information which the observer has about the occurrence of an event, which means that the assumed probabilities of different observers may be different. The Bayesian approach requires that all uncertainties are modelled with probabilistic concepts, and that the rules of probability calculus are followed in all inference.

Two types of <u>uncertainties</u> are distinguished: <u>epistemic</u> and <u>aleatory</u>. Epistemic uncertainty is attributable to incomplete knowledge about a phenomenon that affects our ability to model it. Aleatory uncertainty is caused by the non deterministic (stochastic, random) nature of phenomena.

<u>Risk</u> is defined relative to hazards or accidents. A hazard is something that presents a potential for health, economical or environmental harm. Risk associated with the hazard is a combination of the probability (or frequency) of the hazardous event and the magnitude of the consequences. The consequences can be represented in several dimensions. A usual engineering definition of risk associated with an event i is:

Risk(event i) = "the probability of an event i" x "the consequences of an event i".

<u>Risk measure</u> and <u>risk metrics</u> are two concepts used in the presentation and interpretation of results from a risk assessment. The risk measure is an operation for assigning a number to something, and the risk metrics is our interpretation of the assigned number. In the PSA context, the various numeric results obtained from the quantification of the model are risk measures. The interpretations of these numbers as core damage risk, plant risk profile, safety margin, etc., are risk metrics.

<u>Risk criteria</u> refer to any quantitative decision making criterion used when results of risk assessment are applied to support decision making. Various types of criteria can be used.

Risk acceptance concepts

Risk is acceptable if it is tolerated by a person or group. Whether a risk is "acceptable" or not, will depend upon the advantages that the person or group perceives to be obtainable in return for taking the risk, whether they accept whatever scientific and other advice is offered about the magnitude of the risk, and numerous other factors, political, social, and psychological.

Risk acceptance is often presented using the ALARP (As Low As Reasonably Practicable) framework. ALARP divides levels of risk into three regions:

1. *Unacceptable (intolerable) region. Risk cannot be justified on any grounds.*
2. *The ALARP or tolerability region. Risk is tolerable if the benefit is desired. Tradeoff analysis is made to evaluate the need for risk reductions.*
3. *Broadly acceptable region. Risk is negligible. No need for further risk reduction.*

ALARP can be applied to a single risk metric. It can be also defined with an F-N curve. Figure 2 presents the risk acceptance criteria for major industrial accidents defined by the Dutch safety authority [VROM-1988].

$$F(N) = 10^{-3} \cdot N^{-2}.$$

A risk neutral acceptance criterion has the form k ▫ N-1, where k is a non-negative factor. Thus, the Dutch criterion for unacceptable risk has an added aversion to large accidents.

While the F-N curve represents a high level safety goal, the CDF and LERF criteria used for interpreting PSA results can be regarded as surrogate safety goals of the high level safety goals. By using surrogate safety goals, which are easier to address, the role and importance of individual safety barriers can be assessed.

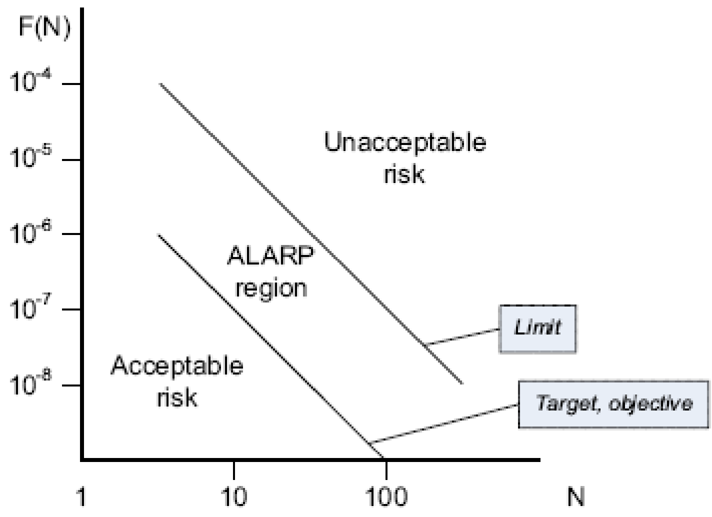

Fig. 2 Societal risk curve with ALARP region as defined by VROM [39]

Residual risk is the remaining risk which cannot be defined in more detail after elimination or inclusion of all conceivable quantified risks in a risk consideration.

Reactor vessel rupture is often given as an example of a residual risk. Based on [WASH-1400], this has been interpreted to correspond to an event with a frequency of approximately 10-7 per year. The residual risk concept is applied in safety analysis as a screening criterion, e.g., as defined in [SKIFS 2004:2].

Safety objectives are the objectives to be achieved, e.g., for safe operation of nuclear power plants (see e.g. [IAEA_INSAG-12]). In the implementation of safety objectives, quantitative targets called (quantitative) safety goals or numerical safety objectives need to be defined.

Regarding safety goals, the terminology varies between different references and countries. For instance, EUR, the European utility requirements document for new light water reactors use the concepts "safety targets" and "probabilistic design targets" [EUR_2002]. EUR defines "targets" as values established by the utilities (e.g. related to the frequency of release of radioactivity), which are more demanding than current regulatory limits, but which are considered reasonably achievable by modern, well designed plants. On the other hand, the UK NII translates the risk acceptance criteria (limit of tolerability) into a Basic Safety Limit (BSL), which has the function of the upper bound of the ALARP region. The lower bound of the ALARP region is called Basic Safety Objective (BSO)".

The references [35], [37] and [38] highlight some important characteristics and difficulty regarding safety goals. An extract of the summary of [38] has been reproduced here with permission of the Authors.

"The outcome of a probabilistic safety assessment (PSA) for a nuclear power plant is a combination of qualitative and quantitative results. Quantitative results are typically presented as the Core Damage Frequency (CDF) and as the frequency of an unacceptable radioactive release. In order to judge the acceptability of PSA results, criteria for the interpretation of results and the assessment of their acceptability need to be defined.

Safety goals are defined in different ways in different countries and also used differently. Many countries are presently developing them in connection to the transfer to risk-informed regulation of both operating nuclear power plants (NPP) and new designs. However, it is far from self-evident how probabilistic safety criteria should be defined and used. On one hand, experience indicates that safety goals are valuable tools for the interpretation of results *from a probabilistic safety assessment (PSA), and they tend to enhance the realism of a risk assessment. On the other hand, strict use of probabilistic criteria is usually avoided. A major problem is the large number of different uncertainties in PSA model, which makes it difficult to demonstrate the compliance with a probabilistic criterion. Further, it has been seen that PSA results can change a lot over time due to scope extensions, revised operating experience data, method development, or increases of level of detail, mostly leading to an increase of the frequency of the calculated risk. This can cause a problem of consistency in the judgements."*

3.3.5 References

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3.4 NATIONAL SITUATION (ASAMPSA2 PARTNERS)

3.4.1 Belgium

In the nineties, the first Level 2 PSA was performed for certain Belgian NPPs but it was limited to the analysis of containment response with the aim of investigating dominant containment failure modes. There was no source term analysis and it considered full power operational state only.

The previous Level 2 PSA has supported the implementation of Passive Autocatalytic Recombiners in all Belgian NPPs to reduce the risk of containment failure due to H_2 burn. Sensitivity studies considering some severe accident management actions have shown their beneficial impact on containment failure probabilities.

In the framework of the present Periodic Safety Review of the Belgian NPPs and considering the WENRA Reference Levels, Level 2 PSA update is underway in Belgium.

The WENRA Reference Levels should be implemented into the Belgian regulations soon. The WENRA Belgian action plan was established in 2007 [42] and includes Level 2 PSA related actions. The present Level 2 PSA update takes into consideration most of these actions. Accordingly, Level 2 PSA is performed for all Belgian representative NPPs and it includes the source term analysis and the shutdown states (not considered in previous Level 2 PSA).

The Level 2 PSA update consists of the extension of the previously developed Accident Progression Event Tree (APET): the new APET is generic for all Belgian NPP (specificities of all units are included), considers the implemented Severe Accident Management Guidance and is extended for source term analysis [43]. It is based on the NUREG-1150 large event tree approach. It is implemented in EVNTRE. The containment fragility curves are established for every representative unit. The supporting calculations are performed with MELCOR 1.8.6. Methodology for basic event quantification has been developed with detailed sections on the use of expert judgement (based on NUREG-1150) and HRA methodology (based on level 1 HRA methodology, THERP and SPAR-H methodologies). Homemade tools to help quantification have also been developed (regarding hydrogen risk analysis for example).

The Level 2 PSA aims to be used in some applications. Presently, the main applications for Level 1 PSA are related with modification (procedures and equipment), support for the training and events analysis. The extension of these applications to Level 2 PSA is under consideration. However, Level 2 PSA will be used to support Belgian NPPs lifetime extension project.

3.4.2 Czech Republic

There are two different types of nuclear units in Czech Republic, VVER-440/213 – 4 units at Dukovany and VVER-1000 – 2 units at Temelin. Historically, performing Level 1 and Level 2 PSAs was an initiative of the plant operator – CEZ, at the beginning with the support of US organisations – for VVER-1000 the first Level 2 PSA in 1996 was prepared by plant personnel and Halliburton NUS company, for VVER-440 in 1995-1998 it was SAIC (Science Applications Int. Corp.) with UJV Rez and financed by US DOE. The update for VVER-1000 from 2003 is again from plant personnel and Scientech, Inc., for VVER-440 the updates from 1998, 2001 and 2005 were performed by UJV Rez under a contract from the plant operator CEZ. Both PSA cover all power and shutdown states for Level 1 PSA and power states only for Level 2 PSA. Extending Level 2 PSA to shutdown states is planned in the near future. In case of VVER-440 it is spoken about a « living » L1+PSA with L2PSA elements updated every year.

The operator – CEZ – made a commitment to the regulatory body to present Level 1 and 2 PSAs in connection with PSR (Periodic Safety Review) to obtain plant operation permit, as this is not required by law. This was applied in 2004 for VVER-1000 and in 2005, 2006, 2007 for VVER-440 (in connection with plant upgrade). The PSR is every 10 years. The regulatory body is preparing a legislation that would require PSA as a part of PSR. The PSA results, particularly Level 1, have been used by the plant operator to identify plant vulnerabilities and performing some upgrades, especially for VVER-440 which is older (the first unit operating from 1985). The regulator, besides assessing the impact of such upgrades, uses the PSA results to check the fulfilment of IAEA INSAG-3 safety goals. There are no quantitative risk limits to compare with PSA results at present.

3.4.3 Finland

The general requirements of PSA and the frequency targets for CDF and large releases are given in the following (Guide YVL 2.8).

"The risks of operation of nuclear power plants are quantitatively analysed by probabilistic safety analysis (PSA). Safety functions for preventing or mitigating accidents and the associated systems necessary to carry out the safety functions are evaluated by these analyses. PSA supports both the design of a nuclear power plant (NPP) and the safety management and control of a NPP all through its service life.

The following numerical design objectives cover the whole nuclear power plant:

- *The mean value of the probability of core damage is less than 10^{-5}/a.*
- *The mean value of the probability of a release exceeding the target value of 100 TBq of ^{137}Cs must be smaller than $5\cdot10^{-7}$/a.*

The design phase PSA shall be used for its part to demonstrate that the plant design basis is adequate and design requirements are sufficient.

The design phase PSA shall be used to demonstrate that the plant meets the numerical design objectives.

Safety classification shall be assessed by PSA. The assessment shall be used to demonstrate that the requirements for quality management system concerning the safety classification of each component are adequate compared with the risk importance of the component.

The purpose of the level 1 and 2 construction phase PSAs is to ensure the conclusions made in the design phase PSA on the plant safety and to set a basis for risk informed safety management during the operation phase of the plant. The level 1 and 2 PSAs shall be based on the plant specifications submitted in conjunction with the application for an operating license.

PSA results shall be applied to the enhancement of safety and to the manifestation of needs for plant changes and to the evaluation of their priority. PSA methods shall be applied to evaluating the optional solutions of the design of system changes.

The results of PSA shall be applied to the assessment of needs for technical specifications changes in conjunction with extensive plant changes in a corresponding way as in the construction phase."

As the mean value of the frequencies above is required, the uncertainty analysis has to be carried out. If only the point estimates of the individual sequences are applied, the inherent uncertainty of the parameters and the model itself cannot be evaluated. The uncertainties may result in very wide release fraction distributions, and this may lead to mean values above very high, e.g. 95th percentiles.

Furthermore, there are more specific requirements on Level 2 PSA:

"The Level 2 PSA shall determine the amount, probability and timing of radioactive substances to be released out from the containment. The assessment shall cover the leaks, damage, controlled releases of radioactive substances and bypass sequences of the containment. The Level 2 PSA shall assess the physical progress and timing of a reactor accident in various accident sequences which endanger the integrity or functional tightness of the containment or in which a release from the primary circuit takes place through systems outside the containment (containment bypass).

The Level 2 PSA shall introduce the following issues:

- *interface between level 1 and 2: description of plant damage states used at level 2, division of level 1 minimal cutsets to level 2 plant damage states, and dependences of level 2 systems and functions from level 1 systems model;*
- *containment event trees;*
- *analysis of the interactions between safety systems and the processes taking place in the containment in the course of an accident;*
- *reliability analysis of the systems used for severe accident management taking into account the conditions prevailing in the containment during an accident and the possibility of erroneous measures;*
- *estimation of the amounts of radioactive substances released from the damaged reactor core into the containment and estimation of the transportation and retention of radionuclides;*
- *estimation of the amounts, quality, height and timing of various radioactive substances released to the environment, and estimation of the respective probability with associated uncertainties;*
- *assessment of the appropriateness and efficiency of the strategy of accident management and the balance between systems (by the aid of e.g. a containment matrix);*
- *expert judgements with related grounds;*
- *results and their evaluation with respective conclusions.*

In the Level 2 PSA, the following issues, among other things, shall be analysed:

- *leak or bypass of the containment e.g. due to a fault in the isolation of the containment, steam generator tube ruptures, systems interfacing LOCAs, or due to seal failures of wall penetrations or access locks;*
- *impact of reaction forces and missiles during different phases of accidents, especially in conjunction with the burst of reactor vessel or other damage to primary circuit;*

- *amount and timing of occurrence of hydrogen generated in various accident sequences, the spreading of hydrogen in the containment, and the likelihood and impact of hydrogen combustion or burning;*
- *steam spiking and steam explosion due to interactions between molten corium and coolant;*
- *melt-through mechanisms of the reactor vessel, their timing and the impact of bursting materials on the integrity of the containment;*
- *other factors endangering the integrity of primary circuit;*
- *rapid growth of pressure in the containment due to e.g. damaged primary circuit, hydrogen combustion or interactions between molten corium and coolant;*
- *recriticality of the reactor core;*
- *slow growth of pressure in the containment due to decay heat or generation of non-condensable gases;*
- *melt-through of the containment due to interactions between molten corium and structures."*

The limit for large release of 100 TBq of ^{137}Cs is less than 0.1% of Cs inventory of the reactor core. As caesium is almost totally released from fuel during the course of core meltdown, the containment has to be very efficient in retaining the fission products, although some of fission products may be deposited on RCS surfaces. The containment design leak rates are generally less than 1% per day at the design pressure. Thus, the release limit requires that in mitigated sequences the containment leaktightness is to be maintained, while natural removal processes of airborne fission products are usually adequate for attaining the requirement. Leakage rates higher than the design value may result in releases below the limit set for large release, provided the leakages can be collected and directed into the stack via a filtering system.

Let us consider natural removal processes in the containment with the removal rate (k_1) of the order of 1/h that is rather high. Now the leakage rate of the containment (k_2) of 1% per day would result in release of around 0.04% (= $k_2/(k_1 + k_2)$) of the fission products released into the containment. This appears to be around the limit of 100 TBq of ^{137}Cs for large NPP units, if the entire caesium inventory is released into the containment. The removal rate could be lower than proposed, which implies that the containment performance should be better than the proposed leak rate of 1% per day. Furthermore, if the leak rate of the containment was set to 10% per day, the leak fraction would become 0.4% that is clearly above the limit of 100 TBq of ^{137}Cs. The leak rate of 10% per day is not usually considered as a very good containment. Of course the possible containment leakage collection and filtering of the releases would decrease the release fraction significantly. Furthermore, if the release limit would be e.g. of an order of magnitude higher, the accuracy of the source term evaluation would become a key issue, and since it involves large uncertainties, it would be very difficult to show the acceptability of the design. The limit of 100 TBq of ^{137}Cs can be reduced to availability of the containment function, which is more straightforward than release evaluation.

The Finnish legislation also includes the requirement of avoiding acute health effects as a result of a severe reactor accident. However if the ^{137}Cs release limit above can be met, it is most probable that there are no acute health effects either. Thus, this does not bring much additional information for Level 2 PSA source term evaluation.

3.4.4 France

<u>A – General</u>

Level 2 PSAs for French NPP are developed by the French utility (EDF) and IRSN (French technical safety organisation). Both organisations develop their L2PSA models independently, with own methods and tools. The L2PSAs developed by

the utility are considered as the reference reactor studies and have now to be provided by the utility at each periodic safety review. The L2PSAs developed by IRSN are used for the review of the utility's conclusions. This approach has been firstly applied for the 900 MWe series during the third decennial periodic safety review (2004-2005) and is being applied for the 1300 MWe series (third decennial periodic safety review) and EPR (final safety report). The 1450 MWe series will be concerned for the second periodic safety review in near future.

The rules for development and application of L2PSA in France have not yet been described in an official text. The existing PSA Basic Safety Rule [44] concerns mainly Level 1 PSA and a decision to extend this rule to Level 2 PSAs has not yet been taken. The IRSN review of L2PSA for 900 MWe PWR has conducted the Safety Authority to make some specific requirements regarding both the L2PSA assumptions and the general methodology. These requirements drive the progress required to be done by the utility for the next versions of L2PSA.

B – Probabilistic Safety goals

The French Safety Authority (ASN) has always kept open the possibility to identify new plant improvements regarding safety, regardless of the accident frequency that can be calculated by PSAs. It is considered that if quantitative probabilistic criteria were provided, and if the compliance with these probabilistic criteria was demonstrated, this could lead to a low motivation for supplementary safety improvements. In that context, the French rules for PSA do not include any quantitative probabilistic criteria that should be strictly demonstrated by the utilities.

For example, the PSA Basic Safety Rule [44] does not give any numerical criterion, but indicates nevertheless that case by case orientation values can be defined. An example is provided hereafter.

- In the letter 1076/77 of the Nuclear Safety Division published in 1977 during the examination of the major technical options for the 1300 MWe plants, the Safety Authority set an overall probabilistic objective expressed as follows: "*In general terms, the design of a plant which includes a pressurised water nuclear reactor should be such that the overall probability that the plant could be the source of unacceptable consequences should not exceed 10^{-6} per year. This implies that, whenever a probabilistic approach is used to assess whether a family of events must be taken into account in the reactor design, the family must effectively be taken into account if its probability to lead to unacceptable consequences exceeds 10^{-7} per year (…)*." The 10^{-6} value is considered an "objective" for a PWR plant, and the utility has not been required to demonstrate that this objective has been achieved. The overall objective is stipulated in terms of "unacceptable consequences", but these "unacceptable consequences" are not specified by legislation or regulation.

C – Definition of "large release" and "large early release"

In the applications for French Gen II PWRs, it is considered that "large release" situations include all situations that could lead to worse consequences than a severe accident with a late filtered release (late opening of the containment filtered venting system). The release situations are called "early" if the delay before release is short regarding the possibility of emergency preparedness. An indicative value of 24 hours is used in the practical applications.

For the EPR reactor, the Technical Guidelines for Future PWRs [45] requires that accident situations with core melt which would lead to large early releases have to be "*practically eliminated*" and that "*low pressure core melt sequences have to be dealt with so that the associated maximum conceivable releases would necessitate only very limited protective measures in area and in time for the public. This would be expressed by no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in consumption of food.*" The last sentence may define the bounding limit for the large release regarding the EPR reactor.

For Gen II reactors, it is now considered as “an objective” that situations leading to “large release” should also be “practically eliminated”.

D –A new tool for the safety regulation: the severe accident safety standard (EDF)

The severe accidents were not included in the initial design of the French Gen II PWR. Nevertheless, some specific plant modifications are implemented to improve the plant robustness in case of accident (mainly for the mitigation of the consequences of a severe accident). Progressively the situation became difficult to manage in terms of safety regulation due to the lack of clear safety requirements that should be applied for the operated plants for the severe accident issues, while much progress was made on the severe accident phenomenology knowledge.

In that context, and after several meetings of the “French Advisory Group”, in 2001 the French Safety Authority asked EDF to propose a severe accident safety standard containing as a minimum the approach and objectives for prevention and mitigation of risks associated with serious accidents, the studies necessary to demonstrate compliance with the objectives and the practical provisions and their design basis. This standard should also take into account aspects related to radiation protection of workers and rely on the initial results of Level 2 PSA to prioritise requirements with regard to the level of potential releases for the accidental scenarios considered.

Several versions of this standard have now been established by EDF and successively reviewed by IRSN. The last version of the safety standard includes two parts:

- The safety requirements (approach and safety objectives in terms of prevention and mitigation of severe accident, the studies necessary to demonstrate compliance with the objectives, the current practical provisions and their design basis, the requirement applied to materials),
- The synthesis of the operated plants status related to severe accident (synthesis of existing knowledge on severe accident progression, the status of material behaviour in severe accident conditions, a demonstration that the probabilistic safety goals are achieved and the results of radiological consequences assessment for reference scenarios); this synthesis is supposed to show that the safety requirements are met.

The last review by IRSN and positions of the “French Advisory Group” have conducted the Safety Authority to ask for some complements:

- The continuous improvement of plant safety should be indicated as a key objective, in particular for radiological consequences or probabilistic safety goals,
- Some requirements linked to the long term management of the plant in case of severe accident, materials classification…) should be added.

E – Other applications

The main applications of L2PSA concerns the NPP periodic reviews and plant safety improvement but some other applications are conducted: the identification of priorities for the severe accident R&D efforts, the severe accident knowledge management (in relationship with the emergency organisation).

EDF has also recently proposed a cost-safety benefit method based on L1 and L2PSA to discuss the ranking of potential plant modifications during a periodic safety review.

In the near future, the conclusions of L2PSA are supposed to be used in relation with the future examination of plant lifetime extension for the French Gen II PWR.

3.4.5 Germany

Every ten years, a periodic safety review has to be performed by the licensees of NPPs in Germany. Level 1 PSA has been part of the periodic safety review for many years. A few Level 2 PSAs were performed prior to 2005, exploring

L2PSA methodology within R&D projects, but outside of the periodic safety review. In 2005 Level 2 PSA became part of the periodic safety review, and the licensees now have to submit a PSA (including Level 1 and Level 2) to the licensing authority. The scope of Level 1 PSA is normal operation and shutdown states, while Level 2 PSA has to be performed for normal operation only. A guideline (including Level 1 and Level 2 PSA) has been published by the Bundesamt für Strahlenschutz (BfS) on behalf of the federal ministry for environment, nature conservation and reactor safety (BMU). This guideline comprises a volume on methods [46] and a volume on data [47]. A working group has been installed which will probably propose an updated guideline in 2012.

As of February 2010, the following conclusions can be made:

- Performing and reviewing Level 2 PSA has become a routine task, but knowledge on production and review is not widespread,
- Level 2 PSA have been performed for PWR and BWR,
- The production is done by experienced companies on behalf of the utilities,
- The review is done in parallel to or after the production,
- Review is done by a group of experts (sometimes including experts from abroad) on behalf of the responsible licensing authority of the state where the plant is located,
- The guidelines are helpful, nevertheless the submitted L2PSA are still very different; based on the experience with recent PSA activities the guidelines are currently being updated,
- Since no quantitative probabilistic safety criterion exists, as frequencies of large releases are very low and Level 2 PSA issues are considered beyond design, the L2PSA results only have a direct impact on plant improvements in certain cases,
- Most (but not all) Level 2 PSA apply the "integrated" probabilistic approach, i.e. they use one single computer tool for Level 1 and Level 2 PSA,
- Most Level 2 PSA apply MELCOR as key tool for accident analysis and RiskSpectrum for the probabilistic analysis.

3.4.6 Hungary

During the decision making process in all of its regulatory areas, the Hungarian Atomic Energy Agency Nuclear Safety Department (HAEA NSD) follows deterministic principles and examines if rules and criteria derived from deterministic safety analyses performed with conservative assumptions are met. For many years, the HAEA NSD has also been referring to the application of PSA results in many of its safety policy articles, to the consistent consideration of risk aspects during the regulatory decision making. The HAEA NSD has decided to follow good international practices, therefore an Implementation Plan was developed to define the necessary steps towards risk-informed regulation and to co-ordinate its realisation. The second phase of this implementation plan was started in 2008. The focus is on PSA applications and on tools in support of regulatory decision making and utility risk management.

The nuclear safety requirements related to a nuclear power plant are collected in the first four volumes of the Nuclear Safety Codes (NSC) in Hungary. Volume 3 deals with the design requirements of a nuclear power plant and it contains several prescriptions in relation to the PSA. In its Chapter 3.5.4. "Probabilistic Safety Assessment" it contains requirements providing the framework of constructing a PSA model. Level 1 and 2 PSAs are required for a NPP covering all operational states, modes and initiating events. It is stated that in PSA analyses best estimate approach shall be followed and where it cannot be applied reasonable assumptions shall be considered. General requirements are given related to the data, human failure and common cause modelling applied in the PSA. According to the requirements, uncertainty and sensitivity analysis of the results shall be performed. However, no requirements are contained on the quality of PSA and on the use of PSA and its applications.

HAEA NSD produced and published a regulatory guideline on PSA in Sept. 2006. The guideline describes acceptable methodologies and data to be used for Level 1 and Level 2 PSA studies. Additionally, it describes attributes by which PSA quality can be assessed and it defines regulatory expectations on how changes to PSA models and data can be made and managed.

Presently no numerical criteria are in use in the Hungarian nuclear safety regulation. One Probabilistic Safety Goal (PSG) is stated in the NSC Volume 3 in relation to Level 1 PSA: the total CDF value shall not exceed 10^{-5}/reactor-year considering all initiating events and all operational states. This PSG is very challenging and in reality it is far from being met by the Paks NPP, which is a VVER-440/V-213 type reactor built to earlier standards. No explicit safety goals are present for Level 2 PSA in the current safety regulation.

The Level 2 PSA study was performed from 2001 to 2003 and the uncertainty analysis was finished at the end of 2004. The analysis was basically done by Hungarian research organisations and by Paks NPP. Containment fragility curves were made available as a result of a separate study performed by a US company.

The main objectives of the Level 2 PSA study carried out for a reference unit were: (1) to provide a basis for the development of plant specific accident management strategies, (2) to provide a basis for the plant specific back fit analysis and evaluation of risk reduction options, and (3) to provide a basis for the resolution of specific regulatory concerns.

A Level 2 PSA was performed for all types of initiating events and plant operational states that were included in the Level 1 PSA analysis at the time of launching the Level 2 PSA project. Subsequently, the Level 2 PSA analysis was extended to cover seismic event at full power mode. Currently the Level 2 PSA covers internal events, internal fires and flooding and seismic events during full power operation, internal events in low power and shutdown modes as well as accidents of the spent fuel pool due to internal events, internal fires and internal flooding.

The results of L2PSA were probabilities of the different status/failure of the containment, of the release including timing and height and of consequence categories, according to the activity of Cs released into the environment. As the quantitative results, the annual frequencies of large radioactive releases for 13 different predefined release categories were calculated. The severity of the categories was correlated to the amount of the caesium released. Events of only three release categories may have severe consequences (releases higher than 1000 TBq of Cs).

The risk reduction capability of different accident management possibilities has been assessed. The accident management program is submitted to the regulator and the review process is ongoing. This program comprises hydrogen treatment by using recombiners, flooding of the reactor shaft for the external cooling of the reactor pressure vessel or for protecting the basemat from melt through, filtered venting and prevention of the reactor shaft door damage as mitigative measures. A number of other improvements, mostly preventive measures, are suggested to decrease the frequencies of bypass sequences (i.e. blowdown of the secondary side of the SGs directly to the containment) and decrease the accident initiating frequencies in the shutdown states and in the spent fuel pools.

There is no living PSA programme in place for the Level 2 PSA of NPP Packs. However, a complete revision and update of the initial analysis is planned in a 2-3 year timeframe.

3.4.7 Italy

Regarding the current background of development and applications of L2PSA at a national level, to date no L2PSA criteria have been issued applicable for the risks of operation of NPP in Italy.

3.4.8 Nederland

To be completed

3.4.9 Spain

In Spain, the nuclear rulemaking is developed by the Ministry of Industry and Energy, which delegates the enforcement to the State Organisation, Nuclear Safety Council (CSN), as well as the adoption of instructions, circulars and guidelines of technical nature relating to nuclear and radioactive facilities and activities related to nuclear safety and radiological protection.

Until now, the Spanish Nuclear Regulatory only indicated the need to maintain an adequate level of safety in NPPs [48]. The technical aspects of security requirements have followed a path parallel to the regulations of the country of design origin (USA and Germany). Thus in the late 90s, just as it was done in USA, the CSN and the NPP agreed to develop a program for the creation and use of PSA in Spain [49], which covers power and shutdown states for both internal and external events. In turn, the CSN has developed a series of Safety Guides (GS), which specify the criteria and mechanisms that form part of the review process of the PSA:

- The GS1.10 [50], which regulates the processes of regular review of safety of NPPs, setting a frequency of 10 years and the necessary update of the full PSA Program,
- The GS1.14 [51], which establishes the basic criteria for the performance of the PSA applications through two risk measures: Frequency of large early releases (FGLT) and frequency of major releases (FGL), the latter is applicable only on permanent PSA application,
- The GS1.15 [52] which establishes the criteria for updating and maintenance of the PSA, which vary according to whether or not plants have implemented monitoring and maintenance programs based on risk. As a general rule, apart from significant changes to the Plant, the internal PSA is required to be updated due to refuelling, using the criteria of the RPS for the rest of analysis to complete the PSA.

A new Law for nuclear installations [53] has incorporated criteria of the safety culture in the regulatory requirements for the harmonisation of the safety regulation of NPPs European. Now, the CSN is developing the basic safety requirements applicable to nuclear facilities in Spain [54], containing the recommendations of the IAEA and WENRA reference levels. This document, still in draft, will govern the future scope and development of PSA in Spain.

3.4.10 Sweden

The Authority

The Swedish Radiation Safety Authority (SSM, until summer 2008 two separate organisations SKI – Nuclear Power Inspectorate and SSI – Radiation Protection Inspection Authority) is an authority under the Ministry of the Environment with national responsibility within the areas of nuclear safety, radiation protection and nuclear non-proliferation.

The Regulatory Framework with regard to safety assessment

The basic regulatory statute to be followed by the licensees is SSMFS 2008:1 Regulation and advice on safety in Nuclear facilities. Chapter 4: "Assessment and reporting of the safety of facilities, Safety analysis" give advice on what has to be done by the licensee; "shall" statements. In addition, there is a section with general advice on the interpretation of the "shall" statements. This section uses the wording "should".

- SSM FS 2008:1 Chapter 4

- **The capacity of a facility's barriers and defence-in-depth system to prevent nuclear accidents and mitigate the consequences in the event of an accident shall be analysed by deterministic methods before the facility is constructed, changed and taken into operation.**
- **The analyses shall subsequently be kept up-to-date….**
- **In addition to deterministic analyses in accordance with the first section, the facility shall be analysed by probabilistic methods in order to obtain as comprehensive a view as possible of safety.**

- SSM FS 2008:1 General Recommendations to chapter 4
  - When applying probabilistic analysis for the evaluation of a facility´s design and operation, one aim should be to obtain a safety level without dominating weaknesses.
  - PSA should include level 1 and level 2
  - Operating states should include
    - Power operation
    - Low power and shutdown
    - Fuel reloading/loading
  - The PSA should be as realistic as possible with regard to models and data, e.g. all initiating event categories of importance should be considered
    - LOCA
    - Transients
    - Area events
    - External events
  - Importance of uncertainties in scope, model and data should be evaluated
  - PSA should be used for evaluation of the safety importance of events (LERs) and plant changes

It has been a tradition that Swedish regulatory requirements regarding the performance of PSA and PSA activities at the utilities have been more descriptive than prescriptive. This means that the regulator has described what is to be done rather than how it is to be done, based on the fact that the full responsibility for the safety at the NPPs, including any analysis activities needed to evaluate or develop the safety, lies with the utilities.

SSM also have a Handbook concerning inspection of the PSA activities of the licensees. This "PSA Review Handbook" (in Swedish) is intended to be a support in the regulators supervision of the PSA activities of the licensees. The term PSA activities is to be interpreted in its widest sense, and includes both the underlying organisation and working procedures of the licensee, the layout and content of the PSA, and its areas of application. The handbook also describes regulators procedures for inspection and review of SAs and PSA activities covering three basic types of review activities:

1. Full PSA review, i.e., the review of a first-time PSA or of a major update or extension of an existing PSA
2. Review of PSA Application, i.e., review of applications where PSA is used as an analysis or decision tool, including risk-informed activities
3. PSA Inspection on site, with the focus on work procedures, management, quality and organisation

For each of these activities, the handbook describes how the review is planned and performed as well as how it is to be documented. The review handbook can be seen as describing the regulators expectation on the scope, objectives, methods, content and format of a PSA that is developed by the licensee.

Safety Goals

SSM does not provide any probabilistic safety goals (target values) for Level 1 PSA or Level 2 PSA. There is a design target regarding the accepted release through the filter or scrubber in case of a sever accident involving core damage. This criteria is a release of a maximum 0.1% of core equivalent to the Barsebäck NPP (now no longer in operation).

Current status of PSAs with regard to Level 2 PSA

All ten operating NPPs have both Level 1 PSA and Level 2 PSA. These PSAs are kept updated on a yearly basis.

References

[40] Regulations concerning Safety in Nuclear Facilities, SSM statutes SSMFS 2008:1, SSM, 2008 (in Swedish)

[41] SKI PSA Review Handbook, Report 2003:48, SKI May 2004 (in Swedish).

3.4.11 Switzerland

To be completed

3.4.12 UK

Regulatory Framework

The UK Health and Safety Executive (HSE) Safety Assessment Principles (SAPs) [55] provide UK nuclear inspectors with a framework for making consistent regulatory judgements on nuclear safety cases presented by duty holders. The SAPs also provide duty-holders with information on the regulatory principles against which their safety provisions will be judged.

HSE's SAPs [55] include the following fundamental principles (paragraph 42):

- FP.3 Protection must be optimised to provide the highest level of safety that is reasonably practicable,
- FP.5 Limitation on risks to individuals: "Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm",
- FP.6 Prevention of accidents: "All reasonably practicable steps must be taken to prevent and mitigate nuclear or radiation accidents",
- FP.8 Protection of present and future generations: "People, present and future, must be protected against radiation risks".

The SAPs are consistent with "Reducing risks protecting people: HSE's decision making process" (R2P2, [56]) which provides an overall framework for decision making based on the demonstration by the duty-holders that the risk is as low as reasonably practicable (ALARP), as required by UK Health & Safety Law. The structure of the targets included in the SAPs is based on the Tolerability of Risk (TOR) framework [57] which has been extended in the more recent R2P2.

Detailed numerical targets are established in the UK for judging whether the duty holder is controlling radiological hazards adequately and reducing risks ALARP. These are described in paragraphs 568 to 638 of the SAPs. These targets are further explained in "Numerical targets and legal limits in Safety Assessment Principles for Nuclear Facilities, An explanatory note" [58].

Of particular relevance here are:

Target 5: Individual risk of death from on-site accidents – any person on the site

Target 6: Frequency dose targets for any single accident – any person on the site

Target 7: Individual risk to people off the site from accidents

Target 8: Frequency dose targets for accidents on an individual facility – any person off the site

Target 9: Total risk of 100 or more fatalities

It should be noted that these targets apply to all fault conditions ranging from the most frequent design basis faults to very low frequency severe accidents. Core damage faults, analysed in the Level 2 PSA, are not assessed in a separate framework and have no subsidiary numerical targets.

The concepts of a Basic Safety Level (BSL) and Basic Safety Objective (BSO) are used in translating the TOR (R2P2, [56]) framework into numerical targets. The BSO marks the lower edge of the broadly acceptable level in R2P2 and the BSL marks the upper edge. These targets are not mandatory but, rather, they are guides to inspectors to indicate where there is the need for consideration of additional safety measures by the duty holders.

1. Individual risk of death from on-site accidents – any person on site (Target 5).

The targets for the individual risk of death to a person on the site, from on-site accidents that result in exposure to ionising radiation, are per annum (pa):

BSL: 1×10^{-4} pa

BSO: 1×10^{-6} pa

2. Frequency dose targets for any single accident – any person on the site (Target 6)

Table 5 Frequency dose targets for any single accident – any person on the site (Target 6 – UK rules)

The targets for the predicted frequency of any single accident in the facility, which could give doses to a person on the site, are: Effective dose, mSv	Predicted frequency per annum	
	BSL	BSO
2 – 20	1×10^{-1}	1×10^{-3}
20 – 200	1×10^{-2}	1×10^{-4}
200 – 2000	1×10^{-3}	1×10^{-5}
> 2000	1×10^{-4}	1×10^{-6}

3. Individual risk to people off the site from accidents (Target 7)

The targets for the individual risk of death to a person off the site, from on-site accidents that result in exposure to ionising radiation, are:

BSL: 1×10^{-4} pa

BSO: 1×10^{-6} pa

4. Frequency dose targets for accidents on an individual facility – any person off the site (Target 8)

Table 6 Frequency dose targets for accidents on an individual facility – any person off the site (Target 8 – UK rules)

The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site, are: Effective dose, mSv	Total predicted frequency per annum	
	BSL	BSO
0.1 – 1	1	1×10^{-2}
1 – 10	1×10^{-1}	1×10^{-3}
10 – 100	1×10^{-2}	1×10^{-4}
100 – 1000	1×10^{-3}	1×10^{-5}
> 1000	1×10^{-4}	1×10^{-6}

5. Societal risk – total risk of 100 or more fatalities (Target 9)

The targets for the total risk of 100 or more fatalities, either immediate or eventual, from on-site accidents that result in exposure to ionising radiation, are:

BSL: 1×10^{-5} pa

BSO: 1×10^{-7} pa

PSA Scope

There is an expectation that duty-holders will present PSA analysis compatible with good industry practices. For modern Nuclear Power Plants this implies a Level 1, 2, 3 PSA framework as presented in IAEA Guidance. The SAPs state that a suitable and sufficient PSA should be performed. The scope and depth of PSA may vary depending on the magnitude of the radiological hazard and risks, the novelty of the design, the complexity of the facility, and the nature of the decision that the safety case is supporting. For example, for certain facilities, qualitative arguments, application of good practice, and DBA may be sufficient to demonstrate that the risk is ALARP. However, for a complex facility such as a power reactor or a reprocessing facility, a comprehensive PSA should be developed.

Therefore, the PSA for NPPs should include internal and external events, full power and shutdown operating modes. It is noted that for the older Advanced Gas-cooled Reactors (AGR) and Magnox designs in the UK, there has been no regulatory insistence on Level 2 and Level 3 PSA.

Paragraph 12 of report on numerical targets and legal limits [58] indicates that the BSLs and BSOs in Targets 5 to 8 have been set at a level judged appropriate for a full-scope PSA (i.e. one in which all qualifying faults at the site/facility are included). If a reduced-scope PSA is to be assessed then these BSLs and BSOs will need to be adjusted accordingly.
As previously stated, these targets apply to all fault conditions ranging from the most frequent design basis faults to very low frequency severe accidents. Core damage faults, analysed in the Level 2 PSA, are not assessed in a separate framework and have no subsidiary numerical targets. The concept of large release frequency, which appeared in the previous version of the SAPs, has been superceded by Target 9. It is acknowledged that additional figures of merit including core damage frequency and large release frequency are useful in demonstrating acceptability against international probabilistic criteria, e.g. as proposed by INSAG [59]. However, there are no UK regulatory targets for these.

3.4.13 References

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3.5 NATIONAL SITUATION (OTHER COUNTRIES)

3.5.1 USA

US NRC

The US Nuclear Regulatory Commission (US NRC) has a number of ongoing activities related to Level 2 PSA, accident management, and consequence analysis, which are either performed in collaboration with the international community or are of interest to the international community. Each of these activities is highlighted below.

The US NRC's State-of-the-Art Reactor Consequence Analysis (SOARCA) project [60] involves the reanalysis of severe accident progression and consequences to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. In addition to incorporating the results of more than 25 years of research, the objective of this updated plant analysis is to include the significant plant safety improvements and updates, which have been made by plant owners but were not always reflected in earlier assessments by the US NRC. In particular, these plant safety improvements include system enhancements, improved training and emergency procedures, and offsite emergency response. In addition, these improvements include the recent enhancements in connection with security-related events.

The goal of SOARCA is to generate realistic estimates of the offsite radiological consequences for severe accidents at U.S. operating reactors using a methodology based on state-of-the-art analytical tools. These estimates account for the full extent and value of defence in depth features of plant design and operation, as well as mitigative strategies implemented in the form of Severe Accident Management Guidelines or other procedures. Results of the SOARCA project may also impact the application of deterministic calculations of severe accident behaviour and offsite consequences in Level 2 and Level 3 PSA. For example, comparisons of radiological release estimates from SOARCA to those from past analyses that were based on older modelling technology or that incorporated selected conservatisms, illustrate the extent to which these results impact numerical estimates of risk or revise the understanding of the characteristics of accident sequences that impact offsite radiological consequences.

In the US, a consensus standard exists for the application of an at-power Level 1 and limited Level 2 (large early release frequency - LERF) probabilistic risk assessment (PRA) for internal and external hazards for light-water reactors [61]. The US NRC's position on this standard is articulated in Regulatory Guide 1.200 [62]. There are three additional light-water reactor standards that are under development that are of interest to the Level 2 PSA community. These involve low power and shutdown PRA, Level 2 PRA, and Level 3 PRA. The second item is the focus of this discussion. This standard is being developed to provide requirements for a full Level 2 PRA, as opposed to a limited Level 2 PRA sufficient to estimate LERF. The standard is intended to integrate well with the existing Level 1/LERF standard as well as the Level 3 standard under development. This means that Level 1/2 and Level 2/3 interface issues are being addressed. The standard is also intended to be applicable to both existing and advanced light-water reactors, and will accommodate the differences in the Level 2 PRA risk surrogates used for each type. The target date for providing a draft Level 2 standard for public review is 2011. Subsequent to its issuance, the US NRC will issue supporting

implementation guidance. This activity shares some commonalities with other recent and ongoing international activities such as the ASAMPSA2 project itself, and the 2010 IAEA Specific Safety Guide on the development and application of Level 2 PSA [63].
The US NRC is also participating in an ASME-led effort aimed at developing a PRA standard for advanced non-light water reactors. This standard is intended to cover Level 1, Level 2, and Level 3 PRA for all potentially significant onsite sources of radioactivity, and for all potentially significant initiators and hazards.

The US NRC is also reviewing a number of applications for design certification and combined license for advanced light-water reactors. These reviews include deterministic severe accident analysis, probabilistic Severe Accident Mitigation Design Alternative (SAMDA) analysis, and Level 2 PRA development [64]. In addition, the US NRC is developing the necessary guidance for operational oversight of these new reactors, including risk-informed regulatory guidance and the associated risk metrics (e.g. large release frequency) and target values to be used [65]. The US NRC is also interacting with the international community on new reactor issues through the Multinational Design Evaluation Program (MDEP).
For operating reactors, the US NRC continues to conduct safety and environmental reviews that include Level 2 PRAs. A key example of such an activity is the review of license renewal Severe Accident Mitigation Alternatives (SAMAs, [66]). In addition, limited Level 2 PRAs (quantifying LERF) are a routine part of risk-informed application reviews (e.g. risk-informed changes to the licensing basis).
Recently, the US NRC's Office of Nuclear Regulatory Research announced plans to conduct the first NRC sponsored Level 3 PRA since the late 1980's, when a set of five Level 3 PRAs were conducted as part of the NUREG-1150 study [67]. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," provided a set of PRA models and a snapshot in time (circa 1988) assessment of the severe accident risks associated with five commercial nuclear power plants of differing major reactor and containment designs. Since then, NRC has used the landmark NUREG-1150 results and perspectives in a variety of risk-informed regulatory applications. The vision for the new project is to conduct a comprehensive, integrated Level 3 PRA that evaluates site accident risk to both onsite and offsite populations from all radiological hazards, while considering all plant operating states, all initiating event hazards, and multi-unit effects for sites with multiple units. The main objective of this project is to update and improve our understanding of site accident risk by:

- Incorporating plant safety improvements, insights from SOARCA, and advances in PRA methods, models, tools and data that have occurred in the two decades since NUREG-1150 was published, and
- Integrating the risk from additional radiological hazards (e.g. spent fuel pools, radioactive waste streams, etc.) using consistent assumptions, methods, and tools to enable a meaningful comparison and ranking of risk contributors.

Presently, a scoping study is underway to identify various options for a pilot Level 3 PRA with regard to the following project elements: (1) site selection; (2) project scope; (3) PRA methods, models, tools and data to be used; (4) new research needed to accomplish the project's objectives; and (5) resource estimates and information needs to better understand and address potential challenges. Once approved, the plan is to begin the pilot study in late 2011 or early 2012.
Finally, as part of an exploratory long-term research project, the US NRC is developing a tool for conducting dynamic PRA for postulated severe accident scenarios, by coupling and extending existing capabilities in hardware/phenomena simulation and operator response simulation [68]. Motivations for this activity include a desire to reduce reliance on modelling simplifications, improve treatment of human interaction and mitigation, and leveraging of advances in

computational capabilities and technology developments. Selected developments that are being leveraged include dynamic event tree generation and management tools, the US NRC's severe accident simulation tool (MELCOR), and the IDAC (Information, Decisions, and Actions in a Crew context) operator response model developed by the University of Maryland.

3.5.2 OTHER COUNTRIES

Can be completed during the guideline external review

3.5.3 References

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4 RISK MEASURES / SAFETY INDICATORS, PRESENTATION AND COMMUNICATION OF L2PSA RESULTS

The following chapters 4 and 5 introduce the different risk measures that are considered as state-of-the-art for Level 2 PSA. For all risk measures the analyst must be able to check that its quantification through the APET is relevant. All risk measures may be of interest depending of the final L2PSA applications. It is recommended that several risk measures (multi-criteria risk analysis provide more complete information to the final decision-maker) be used. The definition of risk measure is a key issue for the communication of the L2PSA results (see chapter 2.12).

The present chapter describes the different risk measures / safety indicators that may be calculated by a L2PSA and considered as state-of-the-art. This list has been be built on the basis of ASAMPSA2 partners' experience and completed with other reference documents.

4.1 INTRODUCTION

Before discussing the L2PSA results presentation and the way of obtaining some final conclusions, it might be useful to remind the relationship between the severe accident sequences, the release categories and the source term assessment.

In a "perfect" L1-L2PSA model, each "individual" accident sequence (defined by a list a components and success/failures of human missions) would be associated to one source term (kinetics and amplitude release of each fission product). In such a "perfect" study, millions of couples (frequency x source_term) would be generated. The calculation of so many couples is not currently possible with modern software. Therefore the use of computers and simplification are required and are provided by gathering the individual L1PSA sequences in PDS and the individual severe accident in Release Categories.

The L2PSA analyst or the reviewer must be aware of this limitation and must take it into consideration when presenting final conclusions. The Appendix 9.3 provides some details on this aspect of L2PSA and tries to explain the interest of introducing the source calculation directly in the APET to keep as much information as possible in the final result. Such an approach is possible with tools like EVNTRE, KANT or SPSA.

The following subchapters do not develop this topic but do formulate recommendations on how to use results presentation based on release categories. These recommendations are significant when the source terms of accident sequences gathered in the same release category are homogeneous in terms of amplitude and kinetics.

4.2 FREQUENCIES OF THE FAILURE OF CONTAINMENT FUNCTIONS

In the following paragraphs, the term "containment failure mode" concerns all release paths in the case of an accident, for example, a steam generator tube rupture is considered as a "containment failure mode" although in reality it is the bypass of an intact containment.

4.2.1 First containment function failure

An approach for presenting the results of a L2PSA consists of defining the APET outputs (release categories) with the *first* failures of a containment function during the accident progression. This approach is simple to perform with APET tools that take into account the chronology of the accident but may be more difficult if the chronology is not explicitly addressed (L1PSA APET tools).

In this case, the L2PSA results may be presented by a table as shown in Table 7.

Table 7 Table of result : first containment function failure

First containment function failure	**Frequency (point, fract 5%, 50%, 95%)**
Cont function failure tim1	
Cont function failure tim2	
Cont function failure tim3	
…	

Cont function failure timn	
No Cont function failure	

For example, the frequency of an accidental sequence that leads to the containment failure modes tim1 and tim2 will exclusively contribute to the frequency of the containment failure mode tim1 because it occurs before failure mode tim2.

For each quantification (or each Monte Carlo run), it can be checked that the sum of each first containment failure frequency plus the frequency of situations without containment failure is equal to the L1PSA total frequency.

This presentation may not be correlated to the severity of the accident (if the worst containment failure is the second one, it will not appear) and must be used carefully. The main point of interest is the possibility to check the consistency of the final results.

4.2.2 Dominant containment failure mode

If the L2PSA results exhibit sequences including several containment failure modes (for example a leak through a penetration followed by a basemat penetration), it may be useful to define a scaling of the different containment failure modes related to their severity. The definition of severity may consider both the amplitude of release and the accident kinetics. For example an induced steam generator tube rupture is often considered as one of the worst situations for a PWR as it may combine a short delay before atmospheric radioactive release and high amplitude of release. In this case, the L2PSA results will be presented by a table such as Table 8.

Table 8 Table of result : dominant containment failure mode

Dominant containment failure	**Frequency (point, fract 5%, 50%, 95%)**
Cont Failure mode dom1	
Cont Failure mode dom2	
Cont Failure mode dom3	
…	
Cont Failure mode domn	
No Cont Failure	

As an example, if the containment failure mode dom2 is considered to be more dominant than dom1, then the frequency of an accidental sequence that leads to the containment failure modes dom1 and dom2 will exclusively contribute to the frequency of the containment failure mode.

In that case, for each quantification (or each Monte Carlo run), it can be checked that the sum of each dominant containment failure frequency plus the frequency of situations without containment failure is equal to the L1PSA total frequency.

This presentation can be considered as the standard way for a result presentation of a L2PSA. However a clear definition on the scale of “dominant” may not be easy. For example, it is not obvious how to compare an early containment failure with limited leak size to a late containment failure with large leak size. The main limitation is that the dominant containment failure modes mask other containment failures in a sequence. This can bias the L2PSA applications, especially if some conservatism has been introduced in the APET assumptions related to some "dominant" containment failure modes.

4.2.3 Individual containment failure mode

For the Level 2 PSA applications, it may be useful to separately calculate the frequency obtained for each containment failure mode in order to discuss the interest of specific plant improvements regarding the specific contribution of the considered containment failure modes to the risk.

This should be also used to demonstrate that some specific risks can be excluded: for example, if the frequency of late containment failure by hydrogen combustion during MCCI phase was found to be very low, it should be checked that this result is not obtained because the previous modes have masked it.

The quantification of each individual containment failure mode frequency also allows the analyst to check the consistency of its model.

In this case, the L2PSA results are presented by a table such as Table 9.

Table 9 Table of result : individual containment failure mode

Individual containment failure	Frequency (point, fract 5%, 50%, 95%)
Cont Failure mode mod1	
Cont Failure mode mod2	
Cont Failure mode mod3	
…	
Cont Failure mode modn	
No Cont Failure	

For example, the frequency of an accidental sequence that leads to the containment failure modes mod1 <u>and</u> mod2 will contribute to both of the frequencies of the containment failure modes mod1 and mod2. In addition it may be of interest to document the combinations of failures that occur. For example, if a containment bypass is combined with a basemat melt through, the frequency of simultaneous occurrence for both failure modes should be given to complete the information.

For each quantification (or each Monte Carlo run), the sum of each individual containment failure frequency plus the frequency of situations without containment failure, may largely exceed the L1PSA total frequency if the APET allows the quantification of multiple containment failures in each accident sequence. This result has to be clearly explained to the final L2PSA user.

4.2.4 References

[69] M. Villermain, E. Raimond, K. Chevalier-Jabet, N. Rahni and B. Laurent , Method for Examination of Accidental Sequences with Multiple Containment Failure Modes in the French 900 MWe PWR Level 2 PSA, PSAM9, Hong-Kong, China, May 18-23, 2008.

4.3 FREQUENCY OF RELEASES BASED CATEGORIES

A Level 2 PSA provides information related to the failure of the different containment functions during a severe accident. This is a "system-oriented" presentation of results.

Another approach is to present the results through the level of consequences, for example the total atmospheric release of activity (Bq).

4.3.1 L2PSA with release calculations included in the APET

When the probabilistic tools used for the L2PSA APET quantification allow a direct calculation of release for each sequence (or a fine grouping of sequences) (e.g. SPSA developed by STUK or KANT developed by IRSN), it is possible to obtain, as a final result, several thousands of couples of frequency x amplitude of release. The amplitude of the release may be defined by the total atmospheric release activity or any other measure (for example total release activity of ^{137}Cs or ^{131}I or equivalent ^{131}I …).

During the results post-processing phase, it becomes possible to group the different scenarios obtained by their level of consequence. For example, such methods have been used by IRSN for the 900 MWe PWR L2PSA, and it has been conducted for the seven categories of consequences described in Table 10 for general presentation of results. The order of magnitude of the release obtained in this study has been provided but will be updated in the near future to take into account more recent results.

Table 10 Level of consequence defined for the French 900 MWe PWR Level 2 PSA by IRSN [70]

Level of consequence	Example of situation	Quantity of release (order of magnitude)
1 – Release after a major containment failure	Containment initially open Containment failure induced by prompt criticality (dilution accident)	Noble gases: 5 E+18 Bq Aerosols: 4 E+19 Bq Iodine gas: 2 E+17 Organic iodine: 0
2 – Release by containment bypass	SGTR	Noble gases: 2 E+17 Bq Aerosols: 1 E+19 Bq Iodine gas: 2 E+15 Bq Organic iodine: 3 E+13 Bq
3 – Release after containment failure due to energetic phenomena	Hydrogen combustion Direct Containment Heating	Noble gases: 4 E+18 Bq Aerosols: 3 E+18 Bq Iodine gas: 2 E+15 Bq Organic iodine: 3 E+14 Bq
4 – Release through a containment (reactor building) leak	Late containment failure due to slow overpressurisation and no containment venting Containment leak induced by ex-vessel steam explosion	Noble gases: 3 E+18 Bq Aerosols: 1 E+18 Bq Iodine gas: 1 E+15 Bq Organic iodine: 5 E+14 Bq

5 – Release through a leak on containment penetration	Initial or induced penetration leak and release through the auxiliary building	Noble gases: 3 E+17 Bq Aerosols: 3 E+15 Bq Iodine gas: 1 E+16 Bq Organic iodine: 2 E+13 Bq
6 – Late filtered release	Release induced by filtered containment venting and/or after basemat penetration	Noble gases: 5 E+18 Bq Aerosols: 2 E+15 Bq Iodine gas: 6 E+14 Bq Organic iodine: 8 E+14 Bq
7 – Release through nominal containment function	Accident progression stopped in-vessel with no containment failure.	Noble gases: 5 E+16 Bq Aerosols: 1 E+13 Bq Iodine gas: 1 E+12 Bq Organic iodine: 8 E+10 Bq

The main interest in using tools such as direct release calculations for each sequence quantified in the Level 2 PSA is to avoid any mistake in an "a priori" binning of sequences in release categories.

4.3.2 L2PSA with release calculations performed outside the APET quantification

When the L2PSA probabilistic tool does not allow the release calculation within the APET quantification, the analyst has to define the release categories outside the APET. Some sensitivity studies (source term calculations) may help in understanding what the key parameters for the release scenarios are. They can help to define the different scales of consequences to be considered. The final RC definition may include both containment failure modes and amplitude of release.

The quality and the necessary resources for this approach depend on the tool which is applied for the release calculation. One advanced approach is to use state-of-the art accident simulation codes (see Volume 2, section 7) for each characteristic sequence up to the calculation of the releases. Another method comprises combination of sophisticated source term codes, such as MELCOR, COCOSYS, ASTEC backed up by a fast running MC source term code, like the US XSOR code, to get distributions of the source terms for a number of Release Categories, covering epistemic uncertainties (e.g. release from the fuel, depletion phenomena) and aleatoric uncertainties (precise path of fission products through the plant). The simplest approach would be assessments by expert judgement or the transfer of results from comparable analyses.

In practice, both approaches (advanced and simple) may be encountered in a single PSA for different release categories. Reasons for such a choice may be that a detailed analysis seems to be unnecessary for very unlikely sequences, or that even detailed analyses have such a high uncertainty that a large effort is not justified.

4.3.3 References

[70] 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.

4.4 FREQUENCY OF “KINETICS BASED” RELEASE CATEGORIES

4.4.1 Based on containment failure time

The delay before containment failure or delay before the beginning of the release is of high importance when the L2PSA results are used regarding the emergency preparedness. Many degrees in the precision of the results can be defined:

- A simple approach can consider that containment failure during the in-vessel phase of accident leads to “early release” and that containment failure during the ex-vessel phase of accident leads to “late release”. This approach may be used as a first evaluation but it cannot cope fully with the reality of accidents. For example, there is no difference between a scenario with a large or short delay before core uncovery; for some very specific sequences, the containment failure may occur during ex-vessel phase and in a short delay (e.g. hydrogen combustion at the beginning of MCCI phase). Table 11 provides an example of the presentation of results for the simple approach:

Table 11 Table of results based on accident kinetics (function of accident progression phases)

Accident phase (containment failure)	Sub-categories	Frequency (point, fract 5%, 50%, 95%)
In-vessel phase	Cont Failure mode 1	
	Cont Failure mode 2	
	Cont Failure mode 3	
Vessel failure phase	Cont Failure mode 4	
	Cont Failure mode 5	
	Cont Failure mode 6	
Ex-vessel phase	Cont Failure mode 7	
	Cont Failure mode 8	
	Cont Failure mode 9	

- A more precise approach is to consider the delay between the initiation time of the emergency planning (activation of the local and national crisis organisation) and the release start time; this delay may be part of the release category definition. Table 12 provides an example of result presentation for the more precise approach.

Table 12 Table of results based on accident kinetics (function of delay)

Delay between emergency planning activation and containment failure	Sub-categories	Frequency (point, fract 5%, 50%, 95%)
[0-2h]	Cont Failure mode 1a	
	Cont Failure mode 1b	
[2-5h)	Cont Failure mode 2a	
	Cont Failure mode 2b	
[5h-10h]	Cont Failure mode 3a	
	Cont Failure mode 3b	
[10h-24h]	Cont Failure mode 4a	

	Cont Failure mode 4b	
[1 day-2days]	Cont Failure mode 5a	
	Cont Failure mode 5b	
[2 days-4 days]	Cont Failure mode 6a	
	Cont Failure mode 6b	

4.4.2 Based on the delay before obtaining an activity release limit

When using L2PSA regarding emergency preparedness criteria, it may be easier to characterise the kinetics of accidents by using some criteria directly connected to emergency zoning. For example, an order of magnitude of the activity of ^{131}I that would lead to iodine prophylaxis at a distance of 10 km for standard meteorological conditions could be used as criteria to identify the severity of the accident in terms of kinetics. Table 13 provides such an example.

Table 13 Table of results based on accident kinetics (function of delay)

Delay between emergency planning activation and achieving a threshold of activity release	**Sub-categories**	**Frequency (point, fract 5%, 50%, 95%)**
[0-2h]	Cont Failure mode 1a	
	Cont Failure mode 1b	
[2-5h)	Cont Failure mode 2a	
	Cont Failure mode 2b	
[5h-10h]	Cont Failure mode 3a	
	Cont Failure mode 3b	
[10h-24h]	Cont Failure mode 4a	
	Cont Failure mode 4b	
[1 day-2days]	Cont Failure mode 5a	
	Cont Failure mode 5b	
[2 days-4 days]	Cont Failure mode 6a	
	Cont Failure mode 6b	

Many possibilities can be defined depending on the final applications and tools used.

4.5 PRESENTATION OF RESULTS – CONTAINMENT MATRIX

The containment matrix presents the distribution of Level 2 APET analysis results for each PDS. The distribution can be introduced e.g. as release categories or APET end branches describing the different containment failure mechanisms. The result can be shown as frequencies of each PDS leading to different release categories (see Table 14). This kind of matrix is very helpful in judging the rationality of the results as it can be considered whether the consequences of a specific PDS are reasonable or not. To make this easier, the results may be further developed to show the distribution of frequencies of release categories for individual plant damage states (Table 15), or to show the contribution of the PDSs to different release categories (Table 16).

Table 14 Frequencies of different release categories (RC) for each plant damage state (PDS)

	PDS1	PDS2	…	PDSm	sum
RC1	$f_{1,1}$	$f_{2,1}$	…	$f_{m,1}$	f_{RC1}
RC2	$f_{1,2}$	$f_{2,2}$	…	$f_{m,2}$	f_{RC2}
…	…	…	…	…	…
RCn	$f_{1,n}$	$f_{2,n}$	…	$f_{m,n}$	f_{RCn}
sum	f_{PDS1}	f_{PDS2}	…	f_{PDSm}	f_{tot}

Table 15 Fractions of different release category frequencies of the total frequency of the PDS

	PDS1	PDS2	…	PDSm	sum
RC1	$f_{1,1}$ / f_{PDS1}	$f_{2,1}$ / f_{PDS2}	…	$f_{m,1}$ / f_{PDSm}	f_{RC1} / f_{tot}
RC2	$f_{1,2}$ / f_{PDS1}	$f_{2,2}$ / f_{PDS2}	…	$f_{m,2}$ / f_{PDSm}	f_{RC2} / f_{tot}
…	…	…	…	…	…
RCn	$f_{1,n}$ / f_{PDS1}	$f_{2,n}$ / f_{PDS2}	…	$f_{m,n}$ / f_{PDSm}	f_{RCn} / f_{tot}
sum	100%	100%	…	100%	100%

Table 16 Fractions of different PDS frequencies of individual release categories. The last row already shows the fractions of different PDSs of the total frequency results from the level 1 and 2 interface.

	PDS1	PDS2	…	PDSm	sum
RC1	$f_{1,1}$ / f_{RC1}	$f_{2,1}$ / f_{RC1}	…	$f_{m,1}$ / f_{RC1}	100%
RC2	$f_{1,2}$ / f_{RC2}	$f_{2,2}$ / f_{RC2}	…	$f_{m,2}$ / f_{RC2}	100%
…	…	…	…	…	…
RCn	$f_{1,n}$ / f_{RCn}	$f_{2,n}$ / f_{RCn}	…	$f_{m,n}$ / f_{RCn}	100%
Sum	f_{PDS1} / f_{tot}	f_{PDS2} / f_{tot}	…	f_{PDSm} / f_{tot}	100%

The same arrangement of results can be applied for initiating events leading to different release categories and this may give more insight into the interpretation of the results. Of course, separate studies can be applied e.g. for large releases, if it is considered necessary.

4.6 DIAGRAMS FREQUENCIES-CONSEQUENCES

In the late 1960's, F.R. Farmer [71] proposed the visualisation of PSA results in probability of occurrence / extent of consequence diagrams (Fig. 3). The advantage of such a diagram is to place all contributors to the risk in the same figure to allow visual comparisons. There are two ways to build such a diagram:

- Approach 1 : the probability can be expressed in terms of "cumulative probability for exceeding a certain consequence"; this approach can be considered as state-of-the-art,
- Approach 2: each RC is positioned in the graphic with a point (frequency x extent of Consequences).

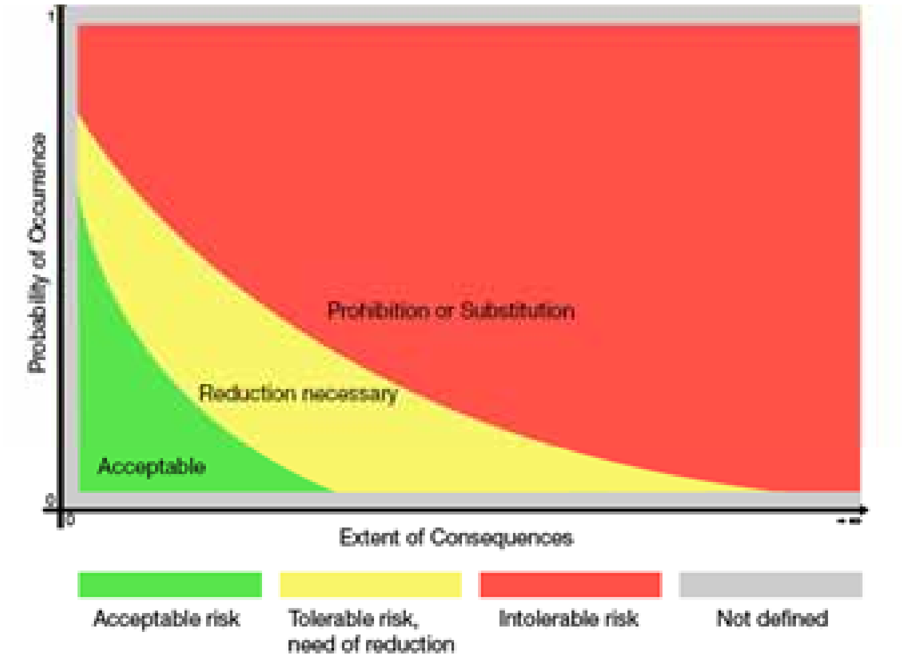

Fig. 3 Farmer's probability of occurrence / extent of consequences diagram

Nevertheless, it should be emphasised that although this type of representation seems to be a useful tool to help in decision-making, some difficulties have been encountered in its practical application:

- The definition of zone (acceptable, reduction necessary, prohibition or substitution) may be extremely difficult to justify regarding the subjective judgements about admissible consequences and the large uncertainties associated to accident consequence analysis and the probabilities of accidents,
- In the second approach, the way of grouping the different accident scenarios may impact their position in the figure and influences their "acceptability".
- The graph can only sort individual events into the acceptance regimes. It cannot provide a measure for the complete set of events. Therefore, in practice, the maximum number of events (= number of points in the graph) has sometimes been defined in a way which may be admissible. A more rigorous approach is to integrate the consequence-risk curve and compare it to a limit or target.

This approach can be recommended as a way to present and discuss the global results of a L2PSA (communication tool) but the notion of "acceptability limit" should be used very carefully. The extent of consequence scale can be presented with different measures of accident consequences (Total Activity Release, ^{131}I release in Bq, Fraction of core inventory etc) or any other qualitative metrics (see chapter 5).

4.6.1 References

[71] F.R. Farmer, Siting Criteria – a new approach, IAEA SM-89/34, 1967, reprinted in Nuclear Safety, 8; pp.539-548, 1967.

4.7 RANKING THE RISK

4.7.1 Frequency X Consequences

A measure of the "source term" (see comment below) risk can be obtained by a formula like:

Total risk = F_1xA (RC_1) + F_2xA (RC_2) + ….. + F_nxA (RC_n)

where F_i is the frequency of the release category RC_i and A(RC_i) is the amplitude of the consequence calculated for the release category RC_i.

This type of evaluation may be applied whatever the nature of consequence calculated but this has significance only if release categories are defined such as:

F1 + F2 + ….. FN = Total Level 1 PSA CDF.

This can be applied for each "point" of APET quantification, or each run in the case of Monte-Carlo simulation.

Comment: in the case of state of the art L2PSA (consequences are calculated through release amplitude), the calculated risk is a "source term" risk to distinguish from the more relevant risk as result of L3PSA considerations. This aspect is discussed in section 5.

4.7.2 Individual Contribution to the "source term" Risk

It may be very useful for the understanding of the Level 2 PSA results to provide the conditional contribution of each release category to the global risk:

Individual contribution of RC_i = F_ixA (RC_i) / (F_1xA (RC_1) + F_2xA (RC_2) + ….. + F_nxA (RC_n)).

The calculations of the individual contributions of each RC_i allow the classification of the RC_i (or containment failure situations) according to their contribution to the global risk. This can be applied for each "point" of APET quantification, or each run in the case of Monte-Carlo simulation.

The classification of the different RC_i contributions can help the analyst to present a scale of containment failure scenarios that takes into account both the frequency and the severity of the consequence.

For example, it may be found that the probability of a severe accident in shutdown state with an "open containment" is very low (e.g. 10^{-8} per year) but the severity of the consequence may require such a sequence to be placed at a high level in terms of risk.

4.7.3 Robustness of the conclusions

The possibility of using L2PSA results to build some classification of the individual risk taking into account both the frequency of the accident and its consequence is certainly one of the most useful potential applications of L2PSA results. If the conclusions are robust enough, it may provide a strong argument for recommending some precise directions to efficiently improve the plant safety.

The analyst should nevertheless provide some indication regarding the robustness of their conclusions:

- The uncertainties on both release category frequencies and consequences should be presented (the calculation mentioned above may be applied within each Monte-Carlo run, if Monte-Carlo method is applied) and/or commented; they should not be dominant in the final classification of individual risks.
- The definition of the release categories should not bias the final conclusions, especially regarding situations with multiple containment failure (e.g. one containment failure should not mask the other ones),
- The dominant Level 1 PSA sequences (if any) should not bias the conclusion (for example, if it can be demonstrated for a dominant L1PSA (e.g. 50 % of total core damage frequency) that the basemat penetration can be avoided, it may not be a global conclusion for the NPP).

4.8 SPECIFIC RESULTS

4.8.1 LERF OR LRF

Depending on the L2PSA application, it may be useful to calculate some specific global results like LERF (Large Early Release Frequency) or LRF (Large Release Frequency).

In that case, a definition of "Large" release and "Early" release has to be provided within the L2PSA. Such definitions can be precise (e.g. large release defined by 100 TBq of equivalent ^{137}Cs defined in the Finnish YVL rules) or only qualitative (e.g. for French PSAs, all release exceeding those calculated in case a late filtered containment venting are qualified of "large").

Some L2PSA may be developed to assess only the LERF for comparison with some probabilistic criteria depending on the national rule. If the limit for large release is high enough, it may allow high simplification of the L2PSA because many release paths may not be considered if they lead to "low" release.

One recommendation is to develop "LERF PSA" as a first model and then to progressively add complementary assessment of all lower release situations. Such an approach makes sense for a continuous plant safety improvement approach.

A detailed review of LERF/LRF notion has been developed in [72].

4.8.2 Containment efficiency (short term, long term …)

An important objective of a L2PSA in comparison with L1PSA is to assess the efficiency of the containment and all severe accident measures to mitigate a potential severe accident.

A Level 2 PSA provides quantitative information of the efficiency of mitigation measure. It is recommended that specific criteria regarding this efficiency are developed, for example:

- The conditional probability to have a containment failure in short term (short term = emergency preparedness not applicable),
- The conditional probability that accident consequences exceed a criteria in the short term (short term = emergency preparedness not applicable),
- The conditional probability to have a containment failure in long term (long term = emergency preparedness applicable),
- The conditional probability that accident consequences exceed a criteria in the long term (long term = emergency preparedness applicable).

For example, for some Gen II reactors, Level 2 PSA exhibits high conditional probability of late containment failure by basemat penetration after vessel failure. This may be considered as a major weakness regarding severe accident measure and containment efficiency although the emergency protection actions are applicable due to the large delay.

The analyst has to check that no dominant sequence of L1PSA drives the final conditional probability (e.g. a slow dominant sequence may lead to a false conclusion that the containment is efficient to avoid the earliest releases).

4.8.3 Atmospheric and liquid releases

Release Categories are generally associated with atmospheric release. Special care is needed for the case of liquid release especially in the case of basemat penetration. Most fission products may be retained in water in the reactor cavity (or containment bottom) and a leak through the basemat zone may lead to a contamination of the soils below the containment through liquid release.

This aspect should be clearly addressed in Level 2 PSA if relevant. In a process of risk ranking, the risk of ground contamination should be considered separately from the atmospheric release. This is due to the different nature of the consequences.

4.8.4 References

[72] A. Bareith, G. Lajtha, J. Dienstbier and E. Grindon, Stable or Final Reactor States and the definition of LERF, SARNET-PSA2-D99.

5 COMPLEMENTARY RISK MEASURES / SAFETY INDICATORS BASED ON EXTENDED L2PSA

5.1 INTRODUCTION

Level 2 PSA aims to calculate the possible sequences of release and their frequencies. The releases are supposed to be defined by their amplitude (expressed in Becquerel for each important isotope) and their kinetics. Any assessment of consequences is considered to be part of Level 3 PSA and is not state-of-the-art for Level 2 PSA.

In the practical application, the Level 2 PSA analysts need to make the link between the amplitude and kinetics of release and the consequences of the accident before deriving relevant conclusions. This may lead to the need for Level 3 PSA but for many organisations the development of a full-scope Level 3 PSA (including assessment of health and environmental impact, taking into account all the local conditions) would be a huge task regarding internal resources.

To overcome this difficulty, some organisations have developed some "extended Level 2 PSA" and have added some simplified assessments of the release consequences to help in the presentation of the conclusions. For example, the Level 2 PSA developed by IRSN for the French 900 MWe and 1300 MWe PWRs is a "Level 2+ PSA" and include, for each Release Category, a calculation of the atmospheric dispersion and dosimetric impact (with standard meteorological conditions and without any assumptions regarding counter-measures).

GRS has performed a Level 2 PSA for a German 900 MWe BWR. Parts of the final result consisted of a frequency distribution of "radiological relevance". For this purpose, the APET was linked to a simple and fast running source term assessment module. This module produced a source term for each individual sequence of the APET. The source term considered four different radioisotopes (J-131, Cs-137, Te-132, Kr-88). For each of these isotopes a relative radiological impact per Bq of release has been defined based on short term health effects. Finally, the total radiological relevance of the combined release of all four isotopes has been calculated for all source terms. Combined with the frequency of source terms, a frequency distribution of the radiological relevance could be produced.

The objective of this chapter is to describe some complementary risk measures / safety indicators that may be calculated by an extended L2PSA. This part should not be considered as state-of-the-art but it proposes some ideas for a multi-criteria analysis and some flexible views regarding the link between risk measures and quantitative safety goals. Such criteria should not be the same for existing and new reactors and they may depend on the NPP location. They can evolve during plant life management in relation with possible plant safety improvements and the requirements of the Safety Authorities.

5.2 FREQUENCY OF "AMPLITUDE BASED (LEVEL OF CONSEQUENCES)" RELEASE CATEGORIES CATEGORISATION BASED ON AN ACCIDENT ABSOLUTE SEVERITY METRICS

The main difficulty in assessing the severity of an accident is to take into account the different nature of the potential accident consequences:

- Early fatalities,
- Early injuries,
- Late cancer fatalities,
- Permanent or temporary loss of land,
- Number of persons relocated temporarily or permanently,
- The ground contamination (soil surface, groundwater, river ….),
- The loss of economical resources (industry, agriculture …),
- The negative image impact (locally, regionally, nationally depending on the amplitude of the consequence),
- The negative impact for nuclear industry (for the specific plant type but also the whole industry …),
- etc.

A precise assessment of all potential accident consequences for every release category would need the development of Level 3 PSA, and would highly depend on the plant location.

For the simplicity and the clarity of the presentation of L2PSA results, there is an interest in building an "accident absolute severity metrics" that would provide an indication of the severity of an accident without any considerations related to:

- The location of the plant (the local meteorological conditions, the population density, the economic activities, and the environment are taken into account to assess the "absolute" severity of the accident),
- The possibility and the efficiency of the emergency actions for the protection of the population.

Such "absolute severity metrics" would address only the NPP safety features without any consideration of offsite environment and the emergency response prepared by the local and national authorities. It could be named an "intrinsic reactor severity scale". It is particularly appropriate for the utility (or vendor) analysis when trying to improve the NPP safety features.

A solution may be to use an existing scale on the example of the INES scale developed by IAEA [73]. The INES scale has been developed *"to facilitate communication and understanding between the technical community, the media and the public on the safety significance of events. It is not the purpose of INES or the international communication system associated with it to define the practices or installations that have to be included within the scope of the regulatory control system, nor to establish requirements for events to be reported by the users to the regulatory authority or to the public."* . This solution has been proposed by Jirina Vitazkova and Erik Cazzoli representing the CCA Company within the project ASAMPSA2. Their main reasoning is presented in Chapter 6.

Using the INES scale as a harmonisation tool for the presentation of L2PSA results is not an application recommended by the IAEA. Nevertheless, it is presented here as something that can be easily done by a L2PSA analyst.

The INES scale is based on general criteria allowing the rating of the events as provided in Table 17.

Table 17 INES scale

TABLE 1. GENERAL CRITERIA FOR RATING EVENTS IN INES

Description and INES Level	People and the environment	Radiological barriers and controls at facilities	Defence in depth
Major accident Level 7	- Major release of radioactive material with widespread health and environmental effects requiring implementation of planned and extended countermeasures.		
Serious accident Level 6	- Significant release of radioactive material likely to require implementation of planned countermeasures.		
Accident with wider consequences Level 5	- Limited release of radioactive material likely to require implementation of some planned countermeasures. - Several deaths from radiation.	- Severe damage to reactor core. - Release of large quantities of radioactive material within an installation with a high probability of significant public exposure. This could arise from a major criticality accident or fire.	
Accident with local consequences Level 4	- Minor release of radioactive material unlikely to result in implementation of planned countermeasures other than local food controls. - At least one death from radiation.	- Fuel melt or damage to fuel resulting in more than 0.1% release of core inventory. - Release of significant quantities of radioactive material within an installation with a high probability of significant public exposure.	
Serious incident Level 3	- Exposure in excess of ten times the statutory annual limit for workers. - Non-lethal deterministic health effect (e.g. burns) from radiation.	- Exposure rates of more than 1 Sv/hr in an operating area. - Severe contamination in an area not expected by design, with a low probability of significant public exposure.	- Near accident at a nuclear power plant with no safety provisions remaining. - Lost or stolen highly radioactive sealed source. - Misdelivered highly radioactive sealed source without adequate radiation procedures in place to handle it.
Incident Level 2	- Exposure of a member of the public in excess of 10mSv. - Exposure of a worker in excess of the statutory annual limits.	- Radiation levels in an operating area of more than 50 mSv/h. - Significant contamination within the facility into an area not expected by design.	- Significant failures in safety provisions but with no actual consequences. - Found highly radioactive sealed orphan source, device or transport package with safety provisions intact. - Inadequate packaging of a highly radioactive sealed source.
Anomaly Level 1			- Overexposure of a member of the public in excess of statutory limits. - Minor problems with safety components with significant defence in depth remaining. - Low activity lost or stolen radioactive source, device or transport package.
	No safety significance (Below scale/Level 0)		

A Level 2 PSA is supposed to examine accident sequences leading to the level of consequences 4 to 7; *"For the accident levels of INES (4-7), criteria have been developed based on the quantity of radioactive material released (...). In order to allow for the wide range of radioactive material that could potentially be released, the scale uses the concept of "radiological equivalence". Thus, the quantity is defined in terms of terabecquerels of I131, and conversion factors are defined to identify the equivalent level for other isotopes that would result in the same level of effective dose."*

The release categories obtained in a L2PSA can be associated to an INES level of consequence in the following way:

- For each release category, the total release for each isotope is converted to an equivalent ^{131}I release, following the conversion table provided in the INES user guide,
- The release category can then be associated to an INES level by the following rule:

  INES - Level 7: "*An event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of more than several tens of thousands of terabecquerels of ^{131}I*",

  INES - Level 6: "*An event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of the order of thousands to tens of thousands of terabecquerels of ^{131}I*",

  INES - Level 5: "*An event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of the order of hundreds to thousands of terabecquerels of ^{131}I*".

The final result of this approach would be a simple list containing the INES levels and the associated frequencies for the plant under consideration.

Such an approach has been tested by IRSN and the following limitations have been identified:

- Some isotopes calculated in the release are not mentioned in the conversion table provided by the INES users guide,

- The limit between levels 5 and 6, and levels 6 and 7, is only indicative and would have to be precisely defined for the presentation of the L2PSA results,
- The dose conversion for ^{131}I mainly takes into account the long term dosimetric effect and the impact of noble gases may be underestimated,
- The INES scale only takes into consideration the atmospheric release: the liquid release and ground contamination are not taken into account.

These limitations are of course due to the fact that the INES scale was not developed for such an application. Nevertheless, the INES scale may be a starting point for the development of an international scale dedicated to Level 2 PSA presentation of results. Such an effort may be an interesting contribution for further harmonisation of L2PSA practices.

Note: the "monetisation" of the accident consequences is also a way to build some scale of the accident; this approach is not discussed here because it supposes a precise study of the local and regional consequences. The result would be very different from one country / region to the other.

5.2.1 Categorisation based on projected doses calculations

Each release category obtained from a Level 2 PSA is associated, for each considered isotope, to one set of kinetics and amplitude of atmospheric release. It may be useful in the final presentation of the results to calculate the radiation impact of the release for different distances and delays with some standard meteorological conditions.

Such a presentation of results may help considerably in the communication of L2PSA results. For example the following can be calculated:

- The projected dose (i.e. the dose likely to be received by an individual through all pathways when no protective actions are implemented) at different distances (e.g. 2, 10, 20, 50 km) and time scales (e.g. 15 days, one year, 50 years),
- The thyroid dose at the same distances and time scales.

When using one criteria (for example projected dose at 2km, 15 days), it becomes possible to classify the different accident scenarios in terms of risks (frequency x consequence) and to have a relatively clear indication of the severity of the accident regarding health effects. The uncertainties on release (source term) calculations can be taken into account especially if they will alter the conclusions.

5.2.2 Categorisation based on ground deposit of fission products

Long term ground contamination by aerosols like ^{137}Cs constitutes the larger impact of a NPP severe accident. It may be useful for the final presentation of the results to calculate the deposition of ^{137}Cs (or other radionuclides) on the ground, at different distances of the NPP (e.g 2, 5, 10, 20, 50 km). The results can be compared to the zoning criteria that may be use for the post-accidental management. Table 18 provides some criteria used for the Tchernobyl accident.

Table 18 Zoning criteria (used for the Tchernobyl accident)

Zoning	^{137}Cs activity
Closed area without permanent residence or economic activity	> 1480 kBq/m2 (40 Ci/km², 40 mSv/an)
Compulsory resettlement areas, where housing and industrial and agricultural production is prohibited	555 to 1480 kBq/m2 (15 to 40 mSv/an, 15 to 40 Ci/km²)
The zones of voluntary resettlement, where people	185 to 555 kBq/m2 (5 to 15 mSv/an, 5 to 15 Ci/km²)

can request a relocation, and no expansion of economic activity is permitted	
Radiological control areas, where no expansion of economic activity is allowed for companies whose activities may affect the environment or human health	37 to 185 kBq/m2 (1 to 5 mSv/an, 1 to 5 Ci/km²).

Each release category can be associated to an extension zoning criteria (taking into account some standard meteorological conditions). Such information can provide a relatively clear indication regarding the long term impact of the considered accidents.
The uncertainties (on the source term) can be presented. It will provide interesting information on the need for further characterisation.

5.2.3 References

[73] IAEA, INES: The International Nuclear and radiological Event Scale user's manual 2008 edition, IAEA-INES-2009.

5.3 SPECIFIC INFORMATION LINKED TO EMERGENCY PLANNING

Level 2 PSA results can be used to discriminate between the sequences that can be managed by the emergency offsite measures and those which are not. This compatibility depends mainly on both the kinetics of the accident and the spatial extension of the counter-measures.
If the Level 2 PSA is extended to some atmospheric dispersion calculations and projected doses, then it is recommended that the following should be provided for each release category:

- The time scale available before reaching some counter-measure criteria (projected dose for sheltering or evacuation, thyroid dose for iodine prophylaxis),
- The distance to which each short term countermeasure (sheltering, evacuation, iodine prophylaxis) should be applied.

Both distances and time scales can be compared to the provision of the emergency plans by the Level 2 PSA analysts. Each release category can be qualified as "compatible or not" to the emergency plans. Such information does not need to be a precise assessment; the main order of magnitude is sufficient to provide useful information to identify the possibility of improving plant safety.

5.4 DIAGRAMS FREQUENCIES-CONSEQUENCES

All measurements of accident consequences (absolute severity scale, projected doses (calculated at a defined distance), ground contamination (Activity of ^{137}Cs deposit, annual dose induced by deposit) versus frequency) can be presented as "cumulative probability for exceeding a certain consequence vs extent of Consequences" or "RC frequency x extent of Consequences diagram" (see section 4.7).

6 A PROPOSAL FOR A COMMON RISK TARGET

The following proposal of Common Risk Target was made by Jirina Vitazkova and Erik Cazzoli representing the CCA company within the project ASAMPSA2. The proposal attempts to reflect ASAMPSA2 contract requirements as well as the user's needs which are defined within the ASAMPSA2 questionnaire. It also reflects the requirements from IAEA, OECD, SARNET conclusions, and Council Directive of The European Union, therefore establishing a Community Framework for the nuclear safety of nuclear installations [74]. Nevertheless, it should be noted that it does not fully reflect the overall opinion of the majority of the community participating in the ASAMPSA2 project.

One of the objectives of L2PSA is the assessment of risk measures to be compared with requirements on safety goals or safety objectives. The demonstration of safety goals may be a requirement of the local authority whilst safety objectives may be defined by individual organisations. The two terms in fact are synonyms, and the distinction may be only formal, since both use essentially the same metrics of "risk".

Currently the local definitions are varied and still under investigation, and the situation could change so rapidly that the organisations performing a Level 2 PSA should carefully check the local requirements. Several panels have been, and are still, compiling and comparing the various practices; the situation is changing so quickly that the results should not be duplicated here. Therefore the users should refer to [75] and [76] for updated summaries.

Work performed for the EU Network of Excellence SARNET has identified the variety of practices as one of the major stumbling blocks in achieving harmonisation within the EU community [77]. More recently the European Council has issued a directive [74] that aims at establishing a Community framework for the safety of nuclear installations. These guidelines propose a common framework based on IAEA definitions and it will be shown that the proposal is compatible with the most stringent local requirements.

The safety targets defined here are not mandatory, but it would be advisable to follow the proposed instructions to attempt to achieve harmonisation. It would also show that IAEA principles have been met and that the community is trying to comply with recognised safety objectives.

6.1 CURRENT UNDERSTANDING OF SEVERE ACCIDENTS SAFETY CRITERIA

A summary accompanied by appropriate discussion on safety criteria for severe accidents (goals, targets, objectives) has been produced by the NEA (OECD) [75]. An up-to-date working report presenting results from the WGRISK task on PSA risk criteria has been published in 2009 [19]. Additional work is under way in the Nordic Countries PSA Group (NPSAG) on this subject ([76], [78], [79]). Related and ongoing work can be found in [80] through [82].

A variety of definitions (both of terminology and criteria) is used in the community, and there seems to be a certain reluctance to discuss the technical basis of the criteria. In general, one should distinguish between "limits" and "objectives" in that limits are numerical values that should not be exceeded, no matter what the circumstances, whilst objectives may be defined with a metric or with surrogates as a "level to which one should strive for but which may never be achieved". The limits are defined by safety authorities in what are commonly called "safety goals" and the objectives may be defined internally to an organisation or within the regulatory framework.

As [75] states, "*The most common metrics used are core damage frequency (CDF) and large release frequency (LRF) or large early release frequency (LERF). In some cases these criteria have been defined as surrogates for higher level metrics and [in] some cases they have been defined in their own right*".

There is no consensus on what LERF or LRF is, but for the most part the concept used by most parties involved is qualitative and complies with the USNRC definition as follows [80]: "*Large Early Release Frequency is defined as the* *frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and a loss of containment isolation".*

Some of the safety goals and objectives based on L(E)RF have added some quantification to this definition, setting limits both on frequency of large releases and on the magnitude of releases expressed in fractions of Iodine inventories. The concept suffers from two problems. The first is that the magnitude of release is plant specific (it depends on core inventories) and therefore a general definition of what constitutes a large release cannot be explicitly and numerically established. The second problem is that the timing of the release is defined relative to the site emergency plan for evacuation of the local population and final evacuation may not even be a strategy contemplated in some countries. For these reasons alone it is hard to think that common safety goals can be defined.

The problem of context was recognised in some countries and a more precise metric is defined there (e.g. UK, Japan, Canada, Holland, Finland). For the most part, the metrics have as a basis the wish to avoid individual and/or societal risks (specifically, one acute fatality in the immediate aftermath of an accident, or an excessive number of fatalities due to radiation-induced cancer, or avoidance of the need for relocation).

However the biggest shortfall of L(E)RF and the related release metrics is that the concept itself may only consider one possible consequence of severe accidents, namely early health effects to the population. In particular, when dealing with the Large Early metric it must be remembered that even the very large release that occurred at Tchernobyl did not result in any prompt fatality among the civilian population. Therefore these metrics, unless proven otherwise, do not provide sufficient sensitivity to measure consequences and do not comply with IAEA safety requirements, which are discussed in section 6.2.

It must be noted again that quantitative safety criteria, when they exist, seem not to have been justified with a technical discussion. When they target prompt health effects alone (with LERF to be exact), they address the least sensitive aspect of radiological releases due to the thresholds of consequences with respect to doses. Therefore LERF is in fact not well suited for the application of factors such as risk reduction or effectiveness of SAM measures. These points are covered in section 6.3.

6.2 SAFETY GOALS AND IAEA RECOMMENDATIONS

The IAEA recommendations and related material about the IAEA mandate on safety of nuclear installations, safety and risk targets, and recommendations can be found in Refs. [83] through [85] (amongst others). The definition, scope, and objectives of the INES scale are found in [86]. To demonstrate some of the shortcomings of the safety goals, definitions, and their practical uses, quotes from the documents in relevant references are given below.

In [83] (emphasis added) the following can be found which relates to responsibility on nuclear facilities, the need for PSAs, and quantitative safety targets:

"First and foremost, each Member State bears full responsibility for the safety of its nuclear facilities. States can be advised, but they cannot be relieved of this responsibility. Secondly, much can be gained by exchanging experience; lessons learned can prevent accidents. Finally, the image of nuclear safety is international; a serious accident anywhere affects the public's view of nuclear power everywhere.

The means for ensuring the safety of nuclear power plants have improved over the years, and it is believed that commonly shared principles for ensuring a very high level of safety can now be stated for all nuclear power plants.

The international consequences of the Tchernobyl accident in 1986 have underlined the need for common safety principles for all countries and all types of nuclear power plants.

The comparison of risks due to nuclear plants with other industrial risks to which people and the environment are exposed makes it necessary to use calculational models in risk analysis. To make full use of these techniques and to support implementation of this general nuclear safety objective, it is important that quantitative targets, 'safety goals', be formulated.

The following concerns general safety objectives and the need for common safety objectives [83]:

a) "General nuclear safety objective

- *To protect individuals, society, and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard.*
- *In the statement of the general nuclear safety objective, radiological hazard means adverse health effects of radiation on both plant workers and the public, and radioactive contamination of land, air, water or food products.*
- *The protection system is effective as stated in the objective if it prevents significant addition either to the risk to health or to the risk of other damage to which individuals, society and the environment are exposed as a consequence of industrial activity already accepted. In this application, the risk associated with an accident or an event is defined as the arithmetic product of the probability of that accident or event and the adverse effect it would produce. The overall risk would then be obtained by considering the entire set of potential events and summing the products of their respective probabilities and consequences.*

b) Radiation protection objective

To ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is as low as reasonably achievable [ALARA], economic and social factors being taken into account, and below prescribed limits, and to ensure mitigation of the extent of radiation exposure due to accidents.

c) Technical safety objective

To prevent with high confidence accidents in nuclear plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small."

One concept that should be emphasised is that the guiding principle should be ALARA. This implies that in the field of radiation protection no risk should be overlooked until it is proven impossible to avoid it. Most of the safety criteria in use on the other hand are based on the principle of ALARP (As Low As Reasonably Practical), which implies that some risks can be discounted if it proves too costly to reduce them.

The acceptance criteria for severe accidents are usually formulated in terms of **risk criteria (probabilistic safety criteria)** [84]:

- Large off-site release of radioactive material: A large release of radioactive material, which would have severe implications for society and would require the offsite emergency arrangements to be implemented, can be specified in a number of ways including the following:
  - As absolute quantities (in Bq) of the most significant nuclides released,
  - As a fraction of the inventory of the core,
  - As a specified dose to the most exposed person off the site,
  - As a release giving 'unacceptable consequences',
- The total Core Damage Frequency (CDF) should not exceed 10^{-4} per reactor year
- Probabilistic safety criteria have also been proposed by INSAG for a very large radioactive release with 'unacceptable consequences'. The following objectives are given:

  10^{-5} per reactor-year for existing plants

  10^{-6} per reactor-year for future plants

  There should be no excessive contribution of any sequence to the total risk of the plant [84].

There is an incontestable **need of international consensus on the risk criteria** [83], as presented already in 1992:

> "*A large off-site release of radionuclides can have severe societal consequences. There is at present [comment: in 1992 - i.e. 18 years ago] no international consensus on the most appropriate measure of what constitutes a large off-site release. Until such time as an international consensus has been reached, it is suggested that the target frequency for a large off-site release should be 10^{-6} / Ry. A large off-site release is defined as one that has severe social implication*".

The issue that is unresolved from these definitions is what exactly constitutes severe social implications. One answer is found in the grades of incidents and accidents provided by the INES scale [86]. Originally introduced in March 1990 jointly by IAEA and OECD/NEA, the aim of the International Nuclear Event Scale (INES) is to **consistently communicate the severity of reported nuclear and radiological incidents and accidents.** It was revised in 2009 to become a more versatile and informative tool. Although it is designed for communication purposes, the scale contains all principles related to nuclear safety, is founded on a sound technical basis (which will be discussed in the next section), and, if deemed complete, could be used to assess frequencies of events. Obviously, as any tool, its principles can be used for any purpose for which they can be applied, including the definition of safety targets.

Fig. 4, taken from [86], shows the levels or grades organisation of the INES scale. It must be remembered that the INES scale follows the ALARA principle, as explicitly stated in the IAEA quotes shown above. In addition, the revised scale considers that the impact on people and the environment may be localised, i.e. radiation doses to one or a few people close to the location of the event, or they can be widespread, as with the release of radioactive material from an installation.

Events are considered in terms of their impact on three different areas: impact on people and the environment; impact on radiological barriers and controls at facilities; and impact on defence in depth.

Fig. 4 INES scale

The exact definitions of the levels and grades are found in Table 19, taken from [86].

Table 19 INES scale: levels definition

LEVEL/ DESCRIPTOR	NATURE OF THE EVENTS	EXAMPLES
7 MAJOR ACCIDENT	• External release of a large fraction of the radioactive material in a large facility (e.g. the core of a power reactor). This would typically involve mixture of short and long lived radioactive fission products (in quantities radiologically equivalent to more than tens of thousands of terabecquerels of ^{131}I). Such a release would result in the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long term environmental consequences.	Chernobyl nuclear power plant, USSR (now in Ukraine), 1986
6 SERIOUS ACCIDENT	• External release of radioactive material (in quantities radiologically equivalent to the order of thousands to tens of thousands of terabecquerels of ^{131}I). Such a release would be likely. to result in full implementation of countermeasures covered by local emergency plans to limit serious health effects.	Kyshtym Reprocessing Plant, USSR (now in Russian Federation), 1957
5 ACCIDENT WITH OFF-SITE RISK	• External release of radioactive material (in quantities radiologically equivalent to the order of hundreds to thousands of terabecquerels of ^{131}I). Such a release would be likely to result in partial implementation of countermeasures covered by emergency plans to lessen the likelihood of health effects. • Severe damage to the installation. This may involve severe damage to a large fraction of the core of a power reactor, a major criticality accident or a major fire or explosion releasing large quantities of radioactivity within the installation.	Windscale Pile, UK, 1957 Three Mile Island nuclear power plant, USA, 1979
4 ACCIDENT WITHOUT SIGNIFICANT OFF-SITE RISK	• External release of radioactivity resulting in a dose to the critical group of the order of a few millisieverts.[a] With such a release the need for off-site protective actions would be generally unlikely except possibly for local food control. • Significant damage to the installation. Such an accident might include damage leading to major on-site recovery problems such as partial core melt in a power reactor and comparable events at non-reactor installations. • Irradiation of one or more workers resulting in an overexposure where a high probability of early death occurs.	Windscale Reprocessing Plant, UK, 1973 Saint Laurent nuclear power plant, France, 1980 Buenos Aires Critical Assembly, Argentina, 1983
3 SERIOUS INCIDENT	• External release of radioactivity resulting in a dose to the critical group of the order of tenths of millisieverts.[a] With such a release, off-site protective measures may not be needed. • On-site events resulting in doses to workers sufficient to cause acute health effects and/or an event resulting in a severe spread of contamination for example a few thousand terabecquerels of activity released in a secondary containment where the material can be returned to a satisfactory storage area. • Incidents in which a further failure of safety systems could lead to accident conditions, or a situation in which safety systems would be unable to prevent an accident if certain initiators were to occur.	Vandellos nuclear power plant, Spain, 1989
2 INCIDENT	• Incidents with significant failure in safety provisions but with sufficient defence in depth remaining to cope with additional failures. These include events where the actual failures would be rated at level 1, but which reveal significant additional organizational inadequacies or safety culture deficiencies. • An event resulting in a dose to a worker exceeding a statutory annual dose limit and/or an event which leads to the presence of significant quantities of radioactivity in the installation in areas not expected by design and which require corrective action.	
1 ANOMALY	• Anomaly beyond the authorized regime, but with significant defence in depth remaining. This may be due to equipment failure, human error or procedural inadequacies and may occur in any area covered by the scale, e.g. plant operation, transport of radioactive material, fuel handling, and waste storage. Examples include: breaches of technical specifications or transport regulations, incidents without direct safety consequences that reveal inadequacies in the organizational system or safety culture, minor defects in pipework beyond the expectations of the surveillance programme.	
DEVIATION 0	• Deviations where operational limits and conditions are not exceeded and which are properly managed in accordance with adequate procedures. Examples include: a single random failure in a redundant system discovered during periodic inspections or tests, a planned reactor trip proceeding normally, spurious initiation of protection systems without significant consequences, leakages within the operational limits, minor spreads of contamination within controlled areas without wider implications for safety culture..	

[a] The doses are expressed in terms of effective dose equivalent (whole dose body). Those criteria, where appropriate, can also be expressed in terms of corresponding annual effluent discharge limits authorized by national authorities.

As Table 19 shows, accidents with offsite risks (level 5) are what are referred to as minimum consequences of severe accidents, because they would result in "*the possibility of acute health effects; delayed health effects over a wide area, possibly involving more than one country; long term environmental consequences*". Also, the other quotes show how risks should be defined (at least those that can be defined as "objective" risks, i.e., that can be quantified), how to interpret risk measures, and what frequencies of consequences are "acceptable" because they can be compared with societal risks incurred by other human activities. The technical basis for the quantification of levels 5 through 7 in terms of Bq of Iodine released can be found in the following section.

6.3 TECHNICAL BASIS FOR LEVELS OF IODINE RELEASES: EXPECTED OFFSITE CONSEQUENCES, AND COMPARISON TO CURRENT SAFETY GOALS OR OBJECTIVES

The safety objectives and safety goals should be consistent with the IAEA documentation and should be comprehensive and consistent from the point of view of the PSA scope to assess the safety of the nuclear installations. Following the IAEA definition of the Technical Safety Objective, the following points are generally accepted:

- **Minor** (if any) consequences stem from DBAs,
- There is an **extremely small likelihood** of severe accidents with **serious radiological consequences**.

However, both the IAEA documents as well as PSA philosophy deal with terminology as "large release", "small likelihood", "severe", "serious", "minor", etc., without exact definitions of the terms. This is in contradiction with the IAEA requirements of quantitative targets and safety goals [83], which in turn would provide for credibility and wide acceptance of PSA.

As noted, the most used semi-quantitative target is L(E)RF (or at least, each practitioner seems to have such a concept in mind for the benchmark to measure the safety of a plant). It is understood as the frequency of "large early" radioactive release, but neither large nor early can be exactly defined. The consequences to which it points are also not clearly defined, because the concept involves plant-, site- and offsite countermeasure-dependent aspects. However, in general, and especially when a more precise metric is not used, some organisations seem to take only into consideration the releases of ^{131}I, thus the only risk to be avoided is the "early fatalities" component, i.e., the extreme consequences that would only be induced in humans through inhalation during the passage of a radioactive cloud.

To put into perspective the various definitions of limits and objectives in terms of offsite consequences, Table 20 shows the results of several MACCS2 [87] calculations. The calculations were performed for a plant located in Central Europe which has a relatively low population density around the plant (the first large settlement is located approximately 20 km away and the average population density is less than 150 persons per square km) with Central European weather data. The radioactive release has the characteristics of an early containment failure (initiation at approximately 6 hours after scram, short duration, relatively high energy, and occurring at 10m elevation. In addition, it has a radionuclide mix typical of severe accident calculations for an early containment rupture). The results shown in Table 20 are for the 95th percentile confidence level (i.e., consequences are not expected to exceed the values shown, no matter what the weather pattern will be). These assumptions, given the population density, can be said to be optimistic for an "average" European site.

Table 20 Consequences of an "early" release corresponding to some of the accepted safety objectives/limits*

Country	Metric	INES	Equivalent		Consequences					
			^{131}I [TBq]	^{137}Cs [TBq]	Early Fatalities (distance in km)	Early injuries	Late cancer fatalities	Permanent or temporary loss of Land (km^2)	Number of person relocated temporarily or permanently
US + others	LERF (Mimimise early cont. failure, cont. bypass, isolation failure, SGTR)	7	$20e^4$ to > 100 e^4	$2e^4$ to > 10 e^4	0 to > 2 (0.2 to > 5)	2 to > 300	8,700 to > 18,000	800 to > 20,000	57,000 to > 2,000,000
Canada	Limit 1% of ^{137}Cs core inventory								
UK**	Limit, 10,000 TBq ^{131}I	6	1 e^4	< 0.1 e^4	0 (0.1)	1	900	1,000	37,000
	Objective, 200 TBq ^{137}Cs	6	0.2 e^4	200	0 (0)	0	180	200	8,000
Sweden	0.1 % of core inventory	5-6	> 0.1 e^4	> 100	0 (0)	0	150	>100	> 5,000
Finland	Limit (new plant at a frequency of 5x10^{-7}/ry) 100 TBq ^{137}Cs	5-6	> 0.1 e^4	100	0 (0)	0	< 100	100	4,000
Canada	Objective (new plants) 100 TBq ^{137}Cs								
		5 lower limit	200	20	0	0	20	< 20	<< 800

* 1 - Consequences shown are for a site with low population density (< 150 person per km^2).
2 - Only long term countermeasures (relocation) are considered.
3 - The consequences will not exceed the values shown with a 95% confidence.
** The Large Release criterion is no longer in the UK legislation. In the UK Safety Assessment Principles the limit and objective are defined by frequency of release.

6.4 COMMON RISK TARGET

A risk target should be a parameter (or a set of parameters) defining the limits beyond which events are unacceptably dangerous with respect to all consequences. A safety goal itself such as L(E)RF is not sufficient from the PSA perspective because:

- Risk is an explicitly expressible value - i.e. multiplication of consequences and frequency, whereby consequences and frequency are concrete numbers (definition of technical safety objective, [79]), whilst the term "safety" does not provide a technically expressible metric,
- "Target" is something to strive for to all possible extents, and which should be achieved, else the endeavour should be abandoned.

The reason, why the term Target is preferred is that the "Goal" has been traditionally used together with "Safety" so it still represents vague content which should be avoided. In addition, "Goal" is traditionally reserved for regulatory authorities.

Thus, the IAEA requirement of quantitative targets and criteria [83] is fulfilled, as well as the requirement of the risk assessment ([83], [84], [85]).

The risk target, RT, can be defined as:

$$RT \leq \Sigma_i f_i \times c_i \qquad (1)$$

Where: i is the ith release mode (class, sequence, source term),

f_i is the maximum frequency per year of the ith release mode, and

c_i is the consequence in Bq of **^{131}I equivalent** for the ith release mode.

The choice of using the ^{131}I equivalent in terms of Bq to define the metric complies with the definition of breakdown in levels of the INES scale and may be related to other metrics by equivalence of effects.

The target for existing nuclear power plants, consistent with the technical safety objective, is a frequency of occurrence of severe core damage that is below approximately 10^{-4} events per plant operating year.

The lower border of the objective is releases from an event that may subsequently be categorised as INES Level 5 and is 200 TBq ^{131}I-equivalent. According to the INES scale this INES level 5 is likely to require implementation of some planned countermeasures and several deaths from radiation could occur. This objective may be justified for reactors where core melt accidents have not been part of the design and where backfitting is impractical. However, it should be considered to improve the objective and to require that INES level 4 be the characteristic outcome of a core melt accident with an accordingly lower acceptable source term.

Keeping INES 5 as the objective, the absolute value of RT is proposed as:

$$\mathbf{RT \leq 200\ Bq\ ^{131}I\text{-equivalent} \times 1\times10^{-4}\ /year} \qquad (2)$$

In order to comply with the numerical requirements set forth in [83] to [85] for frequency and in the INES scale [86] for the lower border consequences of Level 5, RT is a global risk value. In order to comply with the IAEA suggestion that risk should be balanced ("*there should not be excessive contribution to risk by any release mode*"), it is further assumed that the combined release modes included in Level 5 (i,5), Level 6 (i,6), and Level 7 (i,7) of the INES scale should give approximately equal contributions to the total risk, i.e.,

$$\Sigma_i f_{i,5} \times c_{i,5} \approx \Sigma_i f_{i,6} \times c_{i,6} \approx \Sigma_i f_{i,7} \times c_{i,7} \qquad (3)$$

Within these constraints, it follows that the sum of frequencies of all sequences belonging to INES class 5 (200TBq) should be of the order of approximately 3×10^{-5} /year, and those belonging to INES class 6 (2000TBq) should be of the order of approximately 3×10^{-6} / year, and those belonging to class 7 (20000 TBq) should be of the order of approximately 3×10^{-7} / year.

It should be stressed again that (1) through (3) are not prescriptions for safety goals, but they could form the basis of the common risk target to measure risks (and safety) of nuclear installations. These measures are strict enough that operators are not going to become complacent about their plant, especially the already operating installations.

Moreover, the proposed Risk Target methodology has an advantage; the performance of L3PSA is not necessary since the Risk Target itself is expressed in Becquerels related to consequences.

6.5 SPECIAL REMARKS AND CONSIDERATIONS

This section addresses special issues and possible criticisms for the adoption of a common safety target based on the IAEA recommendations. Objectively, all of the objections or comments to the presently proposed safety targets could equally be applied to the existing safety goals in use.

6.5.1 Releases through the ground

The consequences associated with (1) refer to radioactivity released to the atmosphere. The impact of ground contamination should be performed separately if the associated risk is considered to be significant. Some details on ground contamination are provided in Volume 2, chapter 7.

6.5.2 Design basis leak

Defence in depth should be checked by the use of PSA ([83], [84]). There are some reactor designs where the last barrier to the environment cannot be considered as a containment of all effects due to the fact that the design technical specifications allow for design leakage that significantly exceed 1 Volume % per day at a relatively low design pressure. For these designs one can speak of confinement instead of containment. A leak of this magnitude should be a priority concern in case of any severe accident. A proper justification of achievement of the risk target proposed here will be difficult, if not impossible, for these plants.

For existing and operating plants, some special provision may need to be devised to properly assess the risks, and perhaps specific risk targets, that address containment leaktightness, along with provisions to improve leaktightness. **The design leakage should then be such that the overall risk target is not exceeded** and the issue should be dealt with in the design phase for future plants.

6.5.3 Use of the safety target

The proposed definition is; from the point of view of offsite consequences, the threshold given by the IAEA (200 TBq ^{131}I equivalent) would ensure that consequences beyond the exclusion zone of the plants, do not warrant any form of long term intervention (excluding temporary evacuation or sheltering). The issue is whether the frequency given by the IAEA is "acceptable".

In view of future work that addresses risk perception, it could be useful to give consideration to historical evidence (including information that is available in the INES database) and include the data in the assessment of initiator frequencies.

However, **following the IAEA recommendation of using a conservative approach, the target must be especially focused on the consequence component of risk, hence the target should assure that offsite consequences of any accident should be avoided.** The final point about the frequency is that it follows from the INES scale (which is decreasing in frequency by a decade at every level).

6.6 REFERENCES

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7 LEVEL 2 PSA APPLICATIONS

This chapter tentatively discusses the possible applications of the L2PSA. It highlights in chapters 7.1 and 7.2 some requirements associated to L2PSA depending on the final applications.

7.1 LEVEL 2 PSA QUALITY AND CONTENT FOR VARIOUS END USER NEEDS

The ASAMPSA2 End-User survey [91] identified 6 areas of Level 2 PSA 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 larger 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 way of performing the L2PSA and presenting the results. The following paragraphs try to provide some explanations on the 6 areas.

7.1.1 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 in the Level1/Level 2 PSA interface (with definition of the plant damage states, the APET/CET, and the release category definitions) supports the release categories through the model, all the way back to the initiating events in the Level 1 PSA part. This relates to phenomena, technical functions and operator actions that impact on the release characteristics such as timing, amount, dynamics etc.

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

The quantification of frequencies for individual release sequences and for release category sequences must be able to track and take into account the identified important factors independent of the use of an integrated or separated PSA event tree modelling approach.

A precise source term analysis may be useful to provide information on the real efficiency of the containment systems.

7.1.2 To identify plant specific challenges and vulnerabilities of the containment to severe accidents

Identifying plant specific challenges and vulnerabilities of the containment to severe accidents is an objective very similar to the first objective and thus the requirements are similar.

However, the output from initial Level 1 PSA on safety systems related to core degradation prevention is less relevant to this issue. A focused study on severe accident conditions, including analysis of containment failure modes and leak paths and containment functions, is most important for this purpose. The possibility to track back information to the Level 1 PSA initiating events and functions is less important, unless the impact of the initiating and failure/success of the Level 1 functions have a dominating influence on the containment behaviour and containment system vulnerability. The complements in the L1PSA model (bridge event tree) for L2PSA containment analysis purposes can be crucial for this objective.

A precise source term assessment may not be needed if the study is limited to capturing only vulnerabilities that may lead to large release.

7.1.3 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

To provide good insights into whether quantitative safety criteria which typically relate to large release frequencies (LRF) and large early release frequencies (LERF), it is very important to review the Level 1 PSA regarding conservatisms. It is important that all contributors to the core damage frequency (and eventually to release categories) are taken into account. This relates to the scope in terms of the source of radioactivity, the operating reactor states covered by the analysis and the initiating event categories evaluated for each operating state. A similar degree of conservatism (or realism) is needed, e.g. regarding internal and external events frequencies and the conditional probabilities of failure of affected components. It might otherwise be difficult to prove that a criterion is met, or that there are no dominating contributors, and to be certain about the relative importance of different contributing factors.

The choice of (conservatism in) success criteria and data related to dominating sequences in the Level 2 PSA is further an important issue in getting realistic results.

The results have to be evaluated and assumptions and limitations/simplifications checked, and the modelling, data and assumptions, especially for dominating sequences, may require adjustment and fine tuning to enable use of the results for the purpose of showing compliance with quantitative safety criteria.
The exact definitions and effectively used definitions of core damage state in Level 1 and 2 PSA are an important parameter in determining the success criteria for the different Level 1 and Level 2 functions that eventually lead to the frequencies for plant damage states and releases.
Nevertheless, it can be mentioned that if plant design include very high safety margin regarding severe accident prevention and mitigation options, the demonstration that LRF or LERF criteria are met should be feasible with a simplified L2PSA model (including conservatisms).

A precise source term assessment may not be needed for this purpose if a consistent definition of large release is provided (for example, the failure of any component that would increase the normal leak rate of the containment building can be supposed to lead to larger release category).

7.1.4 To identify major containment failure modes and their frequencies, including bypass sequences; and to estimate the corresponding frequency and magnitude of radionuclide releases

The work to identify major containment failure modes and their frequencies, including bypass sequences; and to estimate the corresponding frequency and magnitude of radionuclide releases, requires a good knowledge of the containment performance including test procedures, experience from containment leak data, containment openings, and cable and pipe penetrations and other potential leak paths.

It is important to perform sensitivity assessments of changes in system and function data and phenomena, and how this affects the probabilities of the various containment failure modes in different scenarios, and this will in turn promote the understanding of these scenarios.

Further, it is important to understand the Level 1 PSA outputs and degree of realism/conservatism.

Some dedicated studies on the order of magnitude of the source term are needed to discriminate the different containment failure modes as a function of the severity of the consequences.

7.1.5 To provide an input to the development of plant specific accident management guidance and strategies

For a plant without any specific severe accident management guidance or dedicated systems, a L2PSA can be developed to obtain a ranking of the risk. The results can then be used to support a first version of severe accident management guidance and to be sure that the risk of large release is effectively reduced by application of the guidance.

When some specific severe accident guidance and measures have been developed on a plant, then the Level 2 PSA model should take into account all relevant systems and human actions, including possibility of failures. In that case, the L2PSA should model correctly the advantages and disadvantages (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 guideline (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.

It is important to address the sensitivity and uncertainty in the results related to severe accident management functions and operator actions that are part of the plant specific accident management strategies, to acquire the knowledge about causes and effects that is essential in assessing the applicability of existing or developing new accident management strategies and instructions.

It is necessary to consider all functions (systems, operator actions and phenomena) that influence the results concerning their impact on recovery potential.

The use of simulators including severe accident modelling is recommended to support the L2PSA development

7.1.6 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 specific issue, the Level 1 PSA or the Level 2 PSA are the most important parts for this application. It is important that the L1 and L2PSA scope and level of detail cover the risk reduction options being addressed.

It should be checked that specific L1PSA or L2PSA assumptions do not mask 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 concerning the containment that can be beneficial for internal and external event accident sequences.

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7.2 REQUIREMENTS FOR PRESENTATION OF RESULTS

The calculations of all risk measures can be performed either using point values representing best estimates for all parameters included in the analysis, or by propagating uncertainties with Monte Carlo models for a selected number of parameters characterised by distributions.

Some distributions in the APET are *typically* so wide (the 95th percentile being one or more orders of magnitude higher than the 50th) that, after propagation during APET quantification, the arithmetic means of the convolution of the distributions can be substantially higher than the product of the point or best estimate values. Distributions that typically are skewed towards high values are:

- Frequencies of some initiators (from Level 1 PSA, e.g. containment bypass, seismic events, due to sparse statistics),
- Certain specific phenomena (e.g. delay before vessel failure, mass of corium relocated, small radiological releases, delay before hydrogen ignition, steam explosion triggering, the delay before basemat penetration due to inadequacy of knowledge or simply to the stochastic characteristic of the physics).

It can also be noted that some local distribution can be extremely large without having any significant impact on the final results. In that case, of course, there is no interest to try to reduce the uncertainties.

The results of the analyses therefore can be presented as point value estimates or as the mean of distributions, depending on the objectives of the PSA and the consequent requirements for complexity, as defined in Table 21.

Mean values seem necessary when the requirements of the objectives include the assessment of risks of all offsite consequences (e.g. inclusive of the demonstration of safety goals that are linked to land contamination). The qualification of the results must be clearly identified in the presentation as “point value” or “mean”. If the mean is shown to be necessary to fulfil the objectives, point value estimates can also be provided. In general, it can be recommended to use the fractile 95 % result to check that the final conclusions of the study are robust enough.

Table 21 Requirement in terms of presentation of results

Objectives:	Requirement
To gain insights into the progression of severe accidents, To identify plant specific challenges and vulnerabilities, To identify major containment failure modes and assess their frequencies, To ensure that qualitative safety criteria are met, To develop plant specific accident management guidance.	Point value estimates or best estimates are sufficient. Fractile 95 % can be used to check that uncertainties do not contradict the conclusions.
To ensure that quantitative safety criteria or objectives are met, To evaluate risk reduction options.	Point value estimates or best estimates may be sufficient, but mean values may be necessary depending on the definition of the safety criteria and objectives. Fractile 95 % should be provided: if the result is largely above the criteria then some discussions on the conclusion of the study are needed.

* Demonstration of LERF based on release fractions of Iodine and related to prompt health effects alone may not require complete propagation of uncertainties in the case where LERF is defined a-priori by the major containment failure modes. Assessments that follow the safety objectives suggested in the present guidelines require propagation of uncertainties therefore mean values must be used in the presentation of results.

In addition, in many cases the results show that the mean values exceed the 95th percentile of the distributions. When mean results are required, it is therefore highly desirable to provide the specific quantiles (5th, 50th, and 95th) to show that the mean result may be exceeding the 95th percentile of confidence level (i.e. the mean value is unlikely to occur).

Presentation of the quantiles (at least the 95th percentile) is necessary when the analyses are performed for new reactors, where stricter safety requirements may be applied.

When calculation and presentation of quantiles of distributions are necessary, an uncertainty analysis must be performed through some Monte Carlo model that includes Level 1 PSA data and accident progression data including information needed for assessment of source terms.

7.3 CHECKING THE VALIDITY OF THE CONCLUSION REGARDING THE KNOWN WEAKNESS OF THE TOOLS, QUANTIFICATION THAT HAVE BEEN USED. PROVIDE SOME WARNING. (CONNECTION WITH ALL SUB-CHAPTERS)

This chapter will be completed later.

7.4 IDENTIFICATION OF CONTAINMENT FAILURE MODES, PLANT VULNERABILITIES, VALIDATION OF THE DESIGN

One of the most common applications of Level 2 PSA is the identification of "containment failure modes" and their frequencies. In a more general sense Level 2 PSA are sometimes applied to identify "plant vulnerabilities". A more general scope consists of the "validation of the design". The following sections will address those different – but interrelated – issues.

7.4.1 Identification of containment failure modes

Retention of radionuclides inside the plant is the ultimate safety goal. As long as the retention is assured, accidental consequences outside the plant are not significant. Therefore, containment failure modes and the associated probabilities are of utmost importance in almost every Level 2 PSA. A Level 2 PSA without this feature cannot be considered adequate or complete. However, it is not always obvious how "containment failure" is to be understood when performing a Level 2 PSA. Therefore the following sections recommend a scheme to define this issue.

Under nominal operating conditions the radioactive substances are contained by several barriers. Failure of one of these barriers normally does not lead to significant consequences outside the plant. Only the loss of all barriers between radionuclides inside the fuel and the environment should be called "containment failure".

The number and nature of barriers is not identical in different plants. An obvious example is the difference between PWR and BWR. The BWR does not have the secondary loop as an additional barrier (which can be considered for PWR), but to compensate for this, the BWR has isolation valves for the steam and feedwater lines.

Sometimes it is difficult to determine whether a system or a structure can be considered as a barrier. One example is the containment of some older VVER-reactors which has a significant leak rate by design. Another example in many reactors is the concrete building around a steel containment shell. This concrete building is not really leaktight, but it can mostly be isolated from the environment, and releases to the environment can be controlled and filtered.

These examples demonstrate that one of the first tasks in a PSA is to identify those buildings, structures and systems which are considered to contribute to the containment function. Under normal operating conditions the fuel matrix and the fuel pins are barriers with a retention function. But as per definition the core is damaged when Level 2 PSA begins. Therefore these two barriers have failed right from the beginning in Level 2 PSA and have not to be considered further.

Table 22 lists engineered safety features which contribute to the retention function after onset of core damage and whose failure shall be analysed in Level 2 PSA as a minimum.

Table 22 Issues which have to be analysed in order to identify containment failure

System contributing to retention	Issues which have to be analysed
Reactor coolant system	Pressuriser safety valve(s), critical parts of piping, steam generator tubes (PWR), isolation valves (BWR), systems bypassing the containment building (e.g. volume control system in PWRs)
Secondary heat removal system (PWR)	Steam system outside containment building (e.g. steam safety and relief valves leading to the environment)
Containment	Isolation valves (e.g. for ventilation systems), penetrations

	for tubes and cables, hatches, drains at bottom of building, containment venting system
Rooms and volumes around the containment	Isolation function for ventilation systems, emergency exhaust systems (if any), leaktightness of doors, exhaust route to stack

According to the different subcomponents or subsystems which contribute to retention, different failure modes also exist. Table 23, Table 24, Table 25 and Table 26 list failure modes which should – as a minimum – be considered in Level 2 PSA.

Table 23 Reactor coolant system

Components contributing to retention	Failure modes which have to be analysed
Pressuriser safety valve(s)	Stuck open (may fail due to frequent activation or due to beyond design loads [water, high temperature])
Critical parts of piping (hot leg, surge line)	Induced failure due to imposed loads (pressure, temperature …)
Steam generator tubes (PWR)	Induced failure due to imposed loads (pressure, temperature …)
Isolation valves (BWR)	Failure to close Failure to isolate properly combined with leak in piping outside containment building Failure due to beyond design loads (in particular water ingress into steam lines).

Table 24 Secondary heat removal system (PWR)

Components contributing to retention	Failure modes which have to be analysed
Steam safety and relief valves blowing into environment	Failure to isolate steam generator with tube rupture(s) Stuck open valves (may fail due to beyond design loads)
Feedwater system	Failure to provide feedwater in case of steam generator tube rupture (water covering the rupture location scrubs radionuclides)

Table 25 Containment

Components contributing to retention	Failure modes which have to be analysed
Isolation valves of ventilation system	Failure to close
Building including penetrations (hatches, piping, cables, drains)	Pre-existing beyond design leak Failure due to dynamic pressure (hydrogen combustion, direct containment heating) Failure due to mechanical impact (missiles due to high pressure scenarios, or due to hydrogen combustion) Local overheating of wall (e.g. due to standing hydrogen flames, exhaust from hydrogen recombiners, thermal radiation from or contact with core melt)

	Quasi-static overpressure (in particular due to long term core concrete interaction, or due to failure of pressure suppression systems [BWR]) Melt through of critical parts at bottom (drains, piping, doors) Failure due to sub-pressure (after noncondensible gas is lost and steam condenses)
Containment venting system	Failure to open when required (would lead to overpressure) Failure to close when required (may lead to intolerable releases or to sub-pressure in containment) Failure / overloading of filters in the venting system

Table 26 Rooms and volumes around the containment

Components contributing to containment	**Failure modes which have to be analysed**
Building structure including penetrations (doors, hatches)	Leaks (e.g. due to hydrogen combustion)
Isolation function for ventilation systems, operation of filtered exhaust systems (if any)	Failure to isolate Failure, overload or bypass of filters Failure due to beyond design loads (hydrogen burn, excess temperature)

Since the definition of barriers is not always straightforward, and since different barriers can have various degrees of leakage depending on their design or failure mode, the "containment failure" is not sufficient for estimating consequences of an accident. It could, however, be applied for determining the plant vulnerability which is addressed in the following section.

7.4.2 Identification of plant vulnerability

The expression "plant vulnerability" is sometimes used to characterise whether and how and to what degree the retention of radionuclides is threatened. This is a very general and imprecise term. It can be applied, for example, to express how well a plant is protected against external events. In the context of Level 2 PSA which deals with core melt accidents, the term should be understood as follows: plant vulnerability means the degree and dominant mode of the loss of radionuclide retention due to phenomena caused by core melt.

From this definition it is obvious that there is a very close relation to the identification of containment failure modes. Identification of plant vulnerability is a by-product or a summary of the identification of containment failure modes. The difference is that plant vulnerability is to be understood in a more qualitative and less quantitative way. There is also an implicit meaning that if a plant is vulnerable, something seems to be less than optimal. But altogether, the expression is so imprecise that its application in PSA is not recommended.

It may be used with a certain justification when describing very general characteristics, e.g.:

- Installing hydrogen recombiners will reduce the plant vulnerability with regard to hydrogen combustion,
- If the most significant containment failure mode is melt-through of basemat penetrations, the plant is vulnerable by melt attack on the basemat.

It has to be mentioned that in most cases the identification of plant vulnerability requires quantitative PSA analysis. The statement above with installation of recombiners can only be made with substantiation based on analysis of the complete hydrogen issue. Only very obvious statements may be made easily, e.g. if a containment is inerted, it is not vulnerable by hydrogen combustion.

7.4.3 Validation of the design

The expression "validation of the design" is very pretentious and general. In the context of Level 2 PSA it may be understood twofold:

1. Validation of the design means the demonstration that safety goals applicable to Level 2 issues (e.g. frequency limits for the release of certain quantities of radionuclides) are observed. However if this meaning is intended it would be better to refer to an expression like "compliance with safety goals" instead of "validation of the design",
2. Validation of the design means the demonstration that the plant has no particular vulnerability with regard to phenomena caused by core melt.

The first meaning implies that a complete Level 2 PSA must be performed and the metrics required by the safety goal must be determined. This is a challenging task, but the requirement as such is easily comprehensible.

The second meaning is less clear. Requirements exist in some rules that a particular containment failure mode or a particular phenomenon or a particular component failure must not be a dominant contributor to the consequences of core melt accidents. In addition it is often stipulated that a "cliff-edge" effect must not exist. This means that an expansion of the considered range of frequencies down to slightly lower values must not lead to a dramatic increase in consequences. If this requirement is met, the plant design is sometimes called "well balanced".

Since the expression "validation of the design" is badly defined, it is recommended not to use it in the context of Level 2 PSA.

7.4.4 Summary

Containment failure modes should be identified and their frequencies quantified by Level 2 PSA. However, the definition of barriers is not always straightforward for the different plant designs and therefore the PSA shall identify precisely which structures or components are considered as contributing to the containment function. In addition, different barriers can have various degrees of leakage depending on their design and on their failure mode. After clarification of the type of barriers and their failure modes the PSA shall determine the frequency of different containment failure modes.

Although being a common and recommended Level 2 PSA result, the frequency of containment failure modes is not suitable for estimating consequences of an accident. To characterise such consequences it is necessary to assign the quantity of released radionuclides to each containment failure mode.

The frequency of different containment failure modes might be further used to qualitatively characterise "plant vulnerability" or "validation of the design". However, since neither "plant vulnerability" nor "validation of the design" are expressions with a well defined and comprehensible meaning, it is not recommended to use them in the context of Level 2 PSA.

7.5 ASSESSMENT OF RELEASES

Assessment of releases to the environment and the associated frequencies is the final task in a Level 2 PSA. However, depending on the scope, it is not necessary for all Level 2 PSA to proceed so far. The PSA may, for example, terminate with the assessment of containment failure modes.

The following statements assume that releases to the environment have to be analysed in the Level 2 PSA. Such releases are commonly and throughout the following sections referred to as "source terms".

The assessment of releases provides information about the characteristics of the source term in terms of quantity, composition, timing and location. The source term is combined with release category frequencies in result presentations. 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 Level 3 PSA).

The source term assessment process includes the following steps:

- Choice of representative severe accident sequences within each release category
- Calculation of source terms for the representative severe accident sequences.

It should be mentioned that the uncertainty in the assessment of source terms is significant and could dominate the uncertainties involved in Level 2 PSA. Therefore, additional research in this field would be highly beneficial.

7.5.1 Strategies for different purposes / End Users needs

The end user survey identified 6 areas of Level 2 PSA 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 performance of a detailed assessment is not necessary in all cases. A more detailed assessment is needed especially for objective number 4. For objective number 3, it is necessary to estimate the large early release as the scope of the source term assessment for different release categories will vary depending on the more specific definitions that are used for large and large early release.

The other end user objectives 5 and 6 will also need source term assessment if the mitigation of releases to the environment is seen as the final goal of accident management and risk reduction.

7.5.2 Calculation of source terms for representative severe accident sequences.

The combined effect of the physical and chemical processes impacting on the source term is typically calculated using integrated accident analysis codes e.g. ASTEC, MELCOR, MAAP which model the release and transport of various fission product groups. 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 not explicitly modelled in the integral codes, additional analyses can be used to supplement the source term analysis. Recently, dedicated source term codes have been developed which model the source term phenomena more simply but have the flexibility to consider a much wider range of accident sequences.

Meaningful integral code calculations of source terms require large computing resources and manpower. Therefore minimisation of the number of analyses is desirable. To this end, the numerous APET sequences are grouped into release categories which, per definition, have comparable source terms. 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. Since the source terms are not identical for all sequences within a release category, it is not trivial to select the representative sequence. Furthermore, there are several uncertain parameters which have to be selected. Pessimistic (i.e. maximising release) or realistic assumptions are viable options for defining a source term analysis. Whatever the choice, this has to be clearly decided and documented, and commented in the result presentation.

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. Therefore, the analyst should be aware that source terms calculated by even the most advanced integral accident analysis codes will be subject to considerable uncertainty.

7.5.3 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 Volume 2, section 7) 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 (10 to 20) of groups that can be tracked in an integral code (i.e. to achieve reduction in memory requirements and run time). Grouping structures are based on similarities in the 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.

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. 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.

7.5.4 Source term assessment by integral codes

Two specific codes are widely used in the current generation of Level 2 PSA – MAAP (modular accident analysis program) and MELCOR. Both codes have undergone significant validation (based on both integral and separate effect experiments) and benchmarking exercises. To benefit from the most recent developments and to avoid known deficiencies, it is recommended to apply the latest versions. If this is not feasible or practical, one should at least discuss the potential drawbacks associated with earlier versions. The same remarks are also applicable for ASTEC.

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.

7.5.5 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.

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. 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. Many PSAs, 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 total influence of such factors may be up to several orders of magnitude for some fission product groups.

7.5.5.1 <u>Release in containment bypass sequences</u>

Containment bypass is often the dominating cause of large early releases in the results of Level 2 PSA studies. It is very difficult to find credible mitigative mechanisms for these sequences, since the containment function is lost immediately. However, it is 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.

7.5.5.2 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.

7.5.5.3 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 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. It is usually not considered in Level 2 PSA.
- 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.

Only 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.

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 floor 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.

7.5.5.4 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 [92] 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.

7.5.6 Source term assessment by dedicated (fast-running) source term models

This approach is only recommended if a detailed understanding of the source term issues and in particular of the uncertainty associated with source terms is to be addressed in the PSA. Considering the number of different release scenarios and the existing uncertainties, a large number of calculations may be useful and it is considered to develop fast running source term models. Such fast running models allow for calculating individual source terms for each sequence in an APET, and in addition may be applied for uncertainty and sensitivity analyses. The final result presentation of the PSA will not be able to document all the individual source terms, therefore grouping of source terms will have to be done. Due to the multitude of runs and explored parameters, it is possible to apply several grouping schemes, providing insight on the influence of various aspects (physical phenomena and parameters, accident management) on the source term.

Fast running models will of course be less sophisticated and therefore be less reliable than large models. 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. Therefore it is essential that the fast running models are properly qualified. Examples within the EU exist for successful development and application of such tools. They partly even extend into PSA level 3 issues, or into the field of supporting radiation experts and provide valuable insight into overall risk perspectives.

7.5.7 Presentation of source term assessment results

The source terms and frequencies of the individual Release Categories should be used 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 in terms of health effects 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.

7.5.8 Reference

[92] Project on 'Optimisation of Severe Accident Management strategies for the control of Radiological Releases (OPTSAM)' - CEC Project FIKS-CT1999-00013

7.6 DEVELOPMENT OR VALIDATION OF SEVERE ACCIDENT MEASURES

7.6.1 Introduction

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

When some specific severe accident guidance and measures have been developed on a plant, then the Level 2 PSA model should take into accounts all relevant systems and human actions, including possibility of failures. In that case, the L2PSA should model correctly the advantages and disadvantages (positive and negative impacts) of all actions performed during the severe accident progression. The Human Reliability Analysis has to be precise enough to capture the situations with an unfavourable context for the accident management. The conclusions of the PSA should contribute to the optimisation of the severe accident guideline (minimisation of the risks whatever the accident).

The sensitivity and uncertainty in the results that are related to severe accident management functions and operator actions that are part of the plant specific accident management strategies is important to address.

The use of simulators including severe accident modelling is recommended to support the L2PSA development. The development of simulators including some severe accident modules is identified as a need for complementary R&D activities in support of L2PSA.

7.6.2 Assessment of manual actions

For modern power plants, three stages of documentation cover manual actions:

- Normal Operating Procedures – these procedures are used during normal operation and have the goal to avoid an emergency.
- Emergency Operating Procedures (EOPs) – these guidelines are used during abnormal operation and have the goal to avoid a severe accident.
- Severe Accident Management Guidance (SAMG) – these guidelines are used during a severe accident and have the goal to mitigate the consequences of the accident.

Emergency Operating Procedures are directly coupled to Level 1 PSA. Their actions may be credited in Level 1 PSA. In addition, Level 1 PSA can be used to develop or validate the EOPs.

Similarly, Level 2 PSA can be used to develop and validate SAMG.

The development and validation of SAMG with the use of Level 2 PSA is a multi-step process. As a first step, a Level 2 PSA is performed taking into account only documented actions. Then, based on the results of the Level 2 PSA, measures which mitigate the consequences of relevant sequences can be derived. The following possibilities should be considered:

- Converting sequences to a more favourable release category, for example by avoiding containment overpressure.
- Reducing the source term for a specific release category, for example by taking measures that increase aerosol deposition.

- Increasing the time at which the release takes place, for example by delaying RPV failure with RPV ex-cooling if no RCS injection is possible.

To quantify the effect of a severe accident mitigation measure, variations of the deterministic calculations that illustrate the result of the measures should be performed. However, to be able to evaluate the effect of the measure to the Level 2 PSA, the failure probability of such measures must be evaluated. This may be a combination of system availability and human reliability analysis (see Volume 2, Chapter 3).

In many cases, the overall consequences of a severe accident mitigation measures are a priori unclear. In a typical situation, an earlier but smaller release is traded off against a later but larger release. To justify the use of the SAMG, a quantification of the L2PSA, taking into account the relevant measures, needs to be performed. The result of the L2PSA with and without SAMG should be compared with respect to the risk metrics that have been chosen based on the plant specific safety goals. Based on these risk metrics, the SAMG should provide sequence-dependent guidance on preferred actions.

7.6.3 Examples of PSA application for accident mitigation measures

Severe accident research combined with Level 2 PSA has provided suggestions for several accident management measures. Many of them have been implemented. If there are doubts whether such improvements also have negative consequences and which is the relative importance of advantages and potential drawbacks, Level 2 PSA are very well able to address such issues. Typical plant improvements which have been accomplished, together with their potential drawbacks and the resolution by Level 2 PSA are listed below.

- Installation of passive autocatalytic recombiners (PARs), to mitigate hydrogen threat. Such recombiners reduce the hydrogen content in the atmosphere by recombining it with oxygen. For high hydrogen concentrations PARS may become sources of ignition, which lead to concern whether they might at least partly increase risk. Level 2 PSA has been employed to demonstrate firstly that the probability of entering combustion regimes is significantly reduced. Beyond this, it has been shown that the potential ignition by PARs is even safety enhancing because it prevents later and potentially more critical combustion.
- Installation of containment venting systems to avoid containment overpressure failure: Level 2 PSA is a suitable tool to identify tradeoffs between (relatively early) release through the venting system and (later) containment failure. It is generally assumed that operation of the venting system is beneficial. However, venting systems need operator action and require some components with a finite reliability. Further, the filters/scrubbers and the venting / exhaust lines could fail. Such issues can be and have been addressed in Level 2 PSA to quantify or improve the benefit of these systems.
- Flooding the reactor cavity has been implemented in some reactors to support ex-vessel cooling of corium debris. The efficiency of this strategy has been evaluated by Level 2 PSA, together with the assessment of the potential drawback due to the possibility of steam explosion in the cavity.
- Containment sprays will be operated in several plants to reduce containment loads. There is concern that the condensation of steam might lead to a combustible atmosphere which otherwise might have remained inert. Level 2 PSA have are able to identify if and to what extent this concern is justified.
- Flooding a RPV during core degradations is an obvious means to mitigate the accident. However, additional hydrogen may be generated depending on the actual status of the core and the flooding flow rate. Level 2 PSA are applied to determine whether there are situations where flooding is not recommended.

7.7 PLANT MANAGEMENT (INSPECTION, RECLASSIFICATION OF SYSTEMS

This chapter will be completed later.

7.8 LINK BETWEEN L2PSA AND RESEARCH PROGRAMMES

7.8.1 General discussion

As already mentioned in the OECD [93] technical opinion paper, the integrated severe accident codes (supported by research), or simulation tools in general, play an important role in the quality and acceptance of Level 2 PSA. The progress made on these codes progressively diminishes the role of expert judgements or separate analysis in the quantification of the events.

The L2PSA will still encounter situations where simulation tools are not sufficient to obtain clear conclusions. Then, the L2PSA developers need to insert appropriate uncertainties in the quantification of the event or its consequences:

- it may be difficult (for some plant design) to predict the occurrence of a basemat penetration after vessel rupture ;
- it may be difficult to predict precisely the positive and negative impacts of the in-vessel water injection during core degradation for some sequences;
- the consequences of a reactivity accident may be difficult to address ;
- the behaviour of oxidized ruthenium is not precisely understood, although its dosimetric impact is identified as severe ;
- the degradation of containment tightness in case of ex-vessel steam explosion is a risk to be considered but uncertainties remain very high (advantages and disadvantages of water presence in the reactor cavity are difficult to clarify).

For all these types of issues, L2PSA results (if robust enough) could be used to provide arguments to support (or not) any further R&D efforts.

7.8.2 Examples of topics of interest for complementary research activity

The following table provides a list of topics where some additional research effort may be useful to improve the quality of L2PSA. This table has been established on the basis of the ASAMPSA2 guideline volume 2 and of practical experience from recent PSA.

Table 27 List of R&D topics of high interest for L2PSA development (PWR and BWR Gen II reactors)

Issue	Description
	This table will be completed before, during the external guideline review and next ASAMPSA2 workshop. This table will be an outcome of the workshop.
Introduction of recovery actions into PSA L2	Interface between L1 and L2 becomes complex when component failures, repair times, human actions are to be considered. Human performance in severe accidents could be better assessed if plant simulators for severe accidents exist.
Core degradation for shutdown states	PSA L1 identifies significant contribution to PDS from shutdown states.

with open RPV	Knowledge about core degradation with open RPV (air ingress) is limited.
Coolability of a partly degraded core in the original core region	Integral codes need improvement in the assessment of coolability of a partly degraded core and associated hydrogen generation.
Coolability of core debris in the lower plenum	Integral codes need improvement in the assessment of coolability of core debris in the lower plenum.
Induced failure of RCS components in high pressure sequences	The relative timing of potential induced RCS failures (in hot leg, surge line, safety valve, steam generator, pump seal, RPV bottom) is important for the accident progression. The consequences can vary from benign to catastrophic. Research is needed in two fields: a) structural mechanics taking into account real reactor situation (e.g. pre-existing SGT-faults) b) probabilistic models determining the relative failure contributions This issue is less significant for plants with low frequencies for high pressure sequences (e.g. due to efficient strategies for RPV pressure reduction)
Hydrogen combustion	Finite element (CFD) codes are most advanced with regard to containment atmosphere issues. However, due to resource needs their application for L2PSA is limited. Research is needed in improving the conventional lumped parameter models, and / or in rendering the CFD codes more applicable. This issue is less significant for plants with efficient hydrogen control (inerted containment, hydrogen combiners)
Ex-vessel coolability	A generally agreed map with necessary preconditions for successful ex-vessel cooling (maximum corium load and decay heat level, minimum water requirement, influence of CCI etc) should be established. Influence of real reactor conditions (e.g inhomogeneous corium deposition, steel debris from RPV bottom, small local sump cavities inside main cavity) should be discussed. This issue is less significant for high rated power plants where the chance for successful ex-vessel cooling seems to be small.
Fission product behaviour and source terms	An effort should be made to agree on the degree of uncertainty in source term predictions (which is high) by present-day integral codes. Accordingly, research for reducing these uncertainties seems important. Iodine, Ruthenium and Caesium Molybdate are of particular interest.

7.8.3 References

[93] NEA/CSNI/2007 Technical opinion Paper N°9 - Level-2 PSA for Nuclear Power Plants.

7.9 CAPITALISATION OF KNOWLEDGE – LIVING L2PSA -TRAINING

The development of a L2PSA for a NPP leads to examine many details of the reactor design and its operation. Many results coming from the R&D activities have to be applied to the specific features of the concerned plant.
As for L1PSA, a L2PSA should be considered as a living PSA to be updated during the plant operation life. With that perspective, and taking into account the complexity of many issues, it is crucial to organise the capitalisation of knowledge for a very long time:

- All versions of codes used should be kept including their documentation (particularly those related to the qualification),
- All expert judgements that may be used should be documented,
- The versions of L1PSA and L2PSA event trees should be strictly managed,
- All studies performed to support the L2PSA development have to be documented and required references kept available.

In relation to the effort mentioned above, knowledge coming from L2PSA can be an excellent basis of any training program on severe accident issues for a specific unit: L2PSA can help in formalising information on accident phenomenology, on the expected plant behaviour in degraded conditions and to provide information on risks. Development of such training based on L2PSA is highly recommended.

7.10 EMERGENCY PREPAREDNESS

7.10.1 Introduction

Emergency preparedness of nuclear power plants is handled in several IAEA documents and WENRA has set reference levels for on-site emergency preparedness. All nuclear power plants have to be prepared for different kind of emergency situations. Even though severe reactor accidents have a low probability of occurrence, the emergency planning has to take them into account.

In principle the Level 2 PSA can provide valuable input for emergency planning, but at the same time the emergency planning also influences some of the assumptions in the Level 2 PSA. Depending on the organisational structure and decisions made in SAMG development, the emergency organisation (typically a technical support organisation, including local and national teams, located outside of the main control room) might be the decision-maker in the application of SAMG. Even if the responsible applicant of the SAMG would be the operators in main control room, the emergency organisation still has an important role in influencing the severe accident progression through the actions taken to reach the severe accident safe stable state. Thus, information gained from severe accident progression modelling, and included in Level 2 PSA, can also serve the emergency planning.

In this chapter the main influences of Level 2 PSA on emergency planning and issues that should be addressed in Level 2 PSA for this purpose are highlighted.

7.10.2 Uses of Level 2 PSA to support emergency planning and emergency actions

The Emergency organisation will be in a position to make important decisions during a severe accident, some of which could influence the progression of the severe accident. All the work in the area of severe accident management will provide a background for a Level 2 PSA study and accident progression analysis, investigating a range of possible SAM actions as part of a Level 2 PSA, provides an important input for optimising plant emergency planning.

During a severe accident, information from a Level 2 PSA could be used to provide information key to implementing the most effective SAM actions. Such information includes:

- The most probable severe accident scenarios can be recognised based on available plant conditions and the expected accident progression as reflected in the Level 2 PSA.
- Information about the predicted environmental source terms from these probable scenarios can be gained from the Level 2 PSA (magnitude of releases, timing, release route, etc.).
- This source term information can be used in different ways to support the emergency response (evaluation of radiation conditions at and around the plant site, benefit to be gained from filtered emergency ventilation systems, radiation shielding of rooms where emergency organisation is working etc.).
- The critical plant components can be recognised and recovery actions can be anticipated during accident progression.

The educational aspects of Level 2 PSA are also very important. Persons involved in Level 2 PSA development are experts on severe accident phenomena and this expertise would be very useful in an accident situation. Information gained from the Level 2 PSA, on severe accident progression and physical phenomena, can be used to support emergency organisation training.

Also, some computational tools used in performing a Level 2 PSA might be usable on-line in real time (or faster than real time) during actual accident situations. In some power plants, different kinds of tools using PSA information have been developed for use. For example, in the area of source term calculations, some of the fast-running source term tools might be used in actual accident situations. Some severe accident simulators have been developed based on integral codes. In an emergency situation these could be used to support the prediction of the most probable severe accident scenarios and source terms.

7.10.3 SAM Issues to be addressed in Level 2 PSA

In an emergency situation the emergency organisation will be making important decisions about the possible recovery and mitigation actions and these actions, planned in advance and included in emergency procedures (or SAMG), should also be modelled in Level 2 PSA. When these actions are considered the consequences of severe accident at the plant site have to be taken into account. Especially important is evaluation of radiation conditions at key plant locations, since the recovery actions might not be possible in some accident scenarios with certain radiation conditions. Radiation conditions at key locations are likely to be different in different accident scenarios and modelling some recovery actions case by case might be required. Even in the cases where containment integrity is ensured by severe accident management, the pre-existing containment leakage (design leakage of containment) means that in certain locations at the plant the radiation conditions are more challenging than during normal operation. Level 2 PSA provides valuable input to this evaluation of radiation conditions at key plant locations.

If recovery actions credited in a Level 2 PSA need additional man-power (besides the operational staff always present at the plant) or other resources and materials, the time needed before they can be arranged should be taken into account. The time window during which the emergency organisation can be assumed to be present is defined in the emergency plan and this should also be taken into account in the Level 2 PSA.

If the actions taken by the emergency organisation are credited in Level 2 PSA, the human reliability analysis should take the procedures written for emergency organisation into account.

7.10.4 Examples

7.10.4.1 IRSN

As support of the French Safety Authority, IRSN includes a Crisis Centre that would be activated in case of accident to provide diagnosis / prediction for the situation and to formulate information and recommendations for the protection of the population.

Knowledge gained from the L2PSA development is made available for the Crisis Centre teams:

- A set of thermal-hydraulics studies on a large panel of accident sequences can help the experts to predict the delay before core degradation in complement of other tools.
- The development of a fast-running source term code for L2PSAs, updated with recent R&D results, provides a basis to define the assumptions to be made in the source term code included in the crisis centre SESAM system (SESAM is a set of software designed for diagnosis / prognosis of an accidental situation on a French PWR).
- Some short documents are drafted to summarise key aspects of a severe accident progression and help the crisis centre experts to make a prognosis of the accident, for example:
  - Delay before vessel rupture;
  - Hydrogen production, evolution of containment atmosphere composition, pressure and flammability, for core degradation and MCCI phases;
  - Delay before basemat penetration;
  - Assessment of DCH pressure peak as a consequence of vessel rupture and initial containment atmosphere composition;
  - Behaviour of reactor containment beyond design pressure.
- Fast-running software is also being developed on the basis of existing L2PSA modelling to predict the evolution of the containment atmosphere composition and its flammability, taking into account recombiners, spray system activation or in-vessel water injection.

7.10.5 References

[94] WENRA Reactor Safety Reference Levels, January 2008. Publication can be found through website www.wenra.org

[95] IAEA safety standards can be found through website http://www-ns.iaea.org/standards/documents/default.asp?sub=120

[96] Arrangements for Preparedness for a Nuclear or Radiological Emergency Safety Guide (IAEA Safety Standard Series No. GS-G-2.1)

[97] Preparedness and Response for a Nuclear or Radiological Emergency Safety Requirements (IAEA Safety Standard Series No. GS-R-2

8 SPECIFIC ISSUES RELATED TO SHUTDOWN STATES

8.1 INTRODUCTION

Level 2 PSA studies for low power and shutdown states have not been widely performed to date /1/. However, since the risk arising from shutdown states has been recognised from Level 1 PSA studies performed, more effort has been put into Level 2 PSA studies for low power and shutdown.

Shutdown states can be problematic for NPPs because the structural barriers normally used to ensure safety are challenged by the maintenance and refuelling. During shutdown states the containment might be open as well as the RPV lid during refuelling. As a result, in the event of a severe accident, recovery actions are needed to recover containment integrity. For BWRs in particular, shutdown states present difficulties as the RPV lid is also part of the containment barrier and the containment integrity cannot be recovered if an accident occurs. For BWRs the most important severe accident measure taken to ensure safety during shutdown is prevention of core damage. There are many other issues making the shutdown states different from power operation state - some of the systems normally available are unavailable due to maintenance, many personnel are working in the containment and in controlled zones, loose material is inside the containment etc.

Even though the decay power level in shutdown states is lower and the core inventory is very different, especially after fuel reloading, the severe accident progression is similar and the phenomena that are to be mitigated during a severe accident are the same as those important during power operation. This chapter of the guideline provides information about the specific issues related to shutdown states for PWR and explains how the Level 2 PSA will be affected by them. The severe accident phenomena are introduced later in the Volume 2 of the guideline.

It has to be noted that for some operating PWRs the fuel might be removed from the RPV to the fuel pool at the beginning of shutdown. If this is the case, the only major nuclear safety issue is to ensure fuel pool cooling. Depending on the fuel management scheme, the decay heat load in the fuel pool will be high. If the cooling fails, or if water is lost from the spent fuel pool, the event progression may lead to fuel degradation inside the fuel pool. This issue may be important; however it is not discussed further in this document.

There might also be administrative measures which ensure the containment integrity during most of the shutdown. However this is not the case for all operating reactors and the starting point for this chapter has been a case where fuel is in the RPV and the containment is not isolated for at least some periods during shutdown.

8.2 ISSUES TO BE ADDRESSED IN LEVEL 2 PSA

8.2.1 Open containment

Low power and shutdown PSA is typically divided into different parts according to plant operating mode, for example:

– Startup.
– Hot standby.
– Hot shutdown.

- Cold shutdown.
- Refuelling.

Operating modes differ from each other in respect of plant parameters. From a Level 2 PSA point of view startup, hot standby and hot shutdown can be considered relatively close to power operation. When a plant is approaching refuelling the decay power level remains high and the timing of phenomena is close to the power operation mode. After refuelling the decay power level is lower, core inventory is different and timing of events is also different. Plant operating modes have to be divided appropriately for Level 2 PSA purposes but it is not necessary to handle all the modes separately. For example, modes close to power operating states (startup, hot standby and hot shutdown) can be grouped together with power operation mode and cold plant states can be divided according to primary circuit integrity (for example cold shutdown with open primary circuit / cold shutdown with closed primary circuit). In source term calculations the initial core inventory, which is different after refuelling, has to be taken into account.

Maintenance is mainly performed during cold shutdown and refuelling stages, including different kinds of maintenance actions (periodic maintenance, repair actions, inspections, periodical testing etc.). In these stages the containment might not be leaktight due to maintenance actions and in cold shutdown state the primary circuit pressure is decreased to atmospheric pressure as the primary circuit is open.

Containment integrity might be lost due to maintenance work:

- Access hatch might be permanently open or only one door of double air-lock is used.
- Material hatch might be open.
- Cavity access door might be opened for inspections.
- Systems are opened for maintenance and components might be removed:
    - steam generator access hatch or collector hatches might be open creating connection(s) to secondary side;
    - emergency core cooling systems might be under maintenance;
    - valves in process lines penetrating the containment might be under maintenance.
- Penetrations normally closed and sealed might be opened for example for cabling.
- ...

In case of a severe accident the containment integrity has to be recovered to provide a barrier against fission product releases to the environment. The number of required actions depends on the containment state at the time of an initiating event. Some recovery actions can be performed in a short time, but other actions may require longer. The accident sequence might, in some cases, make the recovery actions impossible. In addition the safety of personnel has to be taken into account since during shutdown there are many people inside the containment.

Plant specific study of containment state during shutdown and recognition of recovery actions is needed for Level 2 PSA purposes. Procedures for performing the recovery actions might also be needed. In some cases the time available to perform the recovery actions may be inadequate and in these cases administrative changes to maintenance practise might be suggested based on Level 2 PSA.

Shutdown sequences typically progress slower than those during power operation due to lower decay power levels (for example in case of loss of residual heat removal). An important question for severe accident management of accidents arising from shutdown states is the timing of recovery actions. Since the actions might require a long time in some cases the actions have to be started well in advance, during the period when the main goal is prevention of core damage. The success of recovery actions for containment integrity has to be evaluated for sequences taking the timing of sequences, human reliability analysis and containment conditions into account.

8.2.2 Open RPV

When the RPV is open, some specific issues have to be taken into account:

- There is easy access to the RPV for additional accident management measures to keep the water level sufficiently high (e.g. use of fire fighting equipment). However, when vaporisation from the RPV begins, access by rescue teams to the RPV (or to the containment in general) may no longer be possible.
- Depending on the outage management, it may be difficult or time consuming to close the containment.
- Core degradation analysis in an open RPV will have to consider the influence of air (less hydrogen production, generation of potentially volatile oxides, chemical reactions with nitrogen)

Convection and thermal radiation from core melt in an open RPV may generate significant thermal loads to structures above the RPV, in particular the containment itself.

8.2.3 System availability

System availability during different modes of shutdown has to be evaluated. The availability of safety systems has already been evaluated for Level 1 PSA purposes, but if additional (systems not included in Level 1 PSA) or dedicated systems are used for severe accident management, their availability has to also be assessed. If the systems are unavailable due to maintenance, the recovery actions are to be recognised and modelled in Level 2 PSA. In addition for system recoveries, the issues considering sequence timing and conditions inside the containment have to be taken into account and additional procedures may be required.

8.2.4 Success criteria for phenomena mitigation

As stated earlier, the lower decay power level influences sequence timing in severe accidents during shutdown states. Even though the decay power which eventually has to be transferred from the containment is lower, this does not generally mean that success criteria can be relaxed.

Different severe accident phenomena during shutdown have to be studied. Separate integral code calculations are needed and they can be supported with calculations using separate tools concentrating on specific issues.

Most effort can be put into assessment of significant plant-specific issues identified by the Level 2 PSA performed for power operating states. However, any justification used to exclude phenomena must be re-evaluated for shutdown states. For example the hydrogen issue should be separately studied starting from core degradation and hydrogen generation scenarios. Containment atmosphere mixing might differ from that expected in power operating states, due to potential additional flow routes between containment compartments. This will have an influence on hydrogen concentrations in different compartments. Also the amount of water available in the containment might be very different and this might influence several issues. For example, in power plants relying on in-vessel corium retention by cavity flooding, the availability of water during shutdown has to be carefully evaluated as the measures normally used for cavity flooding might be unavailable. This issue also links to the containment integrity - if the lower compartment is not watertight the water might not be retained in the containment.

8.3 SHUTDOWN MODELLING IN APET/CET

Accident progression event trees and containment event trees have to be modified according to system availability and possible recovery actions have to be included in the models. Particular issues related to an open RPV (see section 8.2.2) have to be considered.

8.4 SOURCE TERM EVALUATION FOR SHUTDOWN SEQUENCES

Releases of radioactive substances to the environment in shutdown scenarios might be even higher than the expected releases from a typical scenario during power operation, e.g. when the RPV is open and core damage occurs. Typically during power operation the fission product release from the primary circuit to the containment depends on many issues. Leak location, flow velocities, possible water pool above the leak location and physical phenomena (deposition, resuspension, revaporisation etc.) affect the fission product release to the containment. In general more than 50 % of fission products might be deposited in primary circuit piping. During shutdown when the release from the open RPV flows directly to the containment, the amount of fission products in the containment is considerably higher and hence the potential for environmental releases is higher.

In shutdown scenarios, the initiating event may occur several days after reactor shutdown (scram) which will affect the core fission product inventory. The most significant change in fission product inventory will happen during refuelling when old fuel is removed from the reactor and new fuel loaded. Initial core inventory in Level 2 PSA source term calculations has to be chosen according the plant operating mode. Also the capabilities of the source term model used for shutdown state source term calculations have to be evaluated and further modelling development may be necessary.

8.5 REFERENCES

[98] Improving low power and shutdown PSA methods and data to permit better risk comparison and trade-off decision making, Volume 1: Summary of COOPRA and WGRISK surveys. Joint Report Produced by the Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment and the Cooperative Probabilistic Risk Assessment (COOPRA) program. NEA/CSNI/R(2005)11/VOL!, 21-sep-2005

9 APPENDIX

9.1 SEVERE ACCIDENTS CODES

9.1.1 ASTEC

9.1.1.1 Introduction

ASTEC (Accident Source Term Evaluation Code), jointly developed by the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN) and by the German Gesellschaft für Anlagen und Reaktorsicherheit mbH (GRS), aims at describing the behaviour of a whole nuclear power plant in severe accident (SA) conditions including engineered safety systems and procedures used in SA management, from the initiating accidental event until the possible radiological release of radionuclides from the containment building.

The main ASTEC applications are therefore source term determination studies, Level 2 Probabilistic Safety Assessment (L2PSA) studies including the determination of uncertainties, accident management studies and physical analyses of experiments to improve the understanding of the phenomenology.

Its development was based in a first stage (1995-1998) on former codes, respectively at IRSN the ESCADRE system of codes and at GRS the containment codes RALOC and FIPLOC. From that time, ASTEC has progressively reached a larger European dimension, notably within the 5th Framework Programme (FP) with the EVITA project devoted to code validation by independent users [100].

Since 2004, ASTEC is progressively becoming the reference European severe accident integral code for Water-Cooled Reactors through the capitalisation in terms of models of the knowledge produced in the SARNET European Network of Excellence and the assessment by 30 network partners [101],[102].

The first version of the new V2 series, whose development started in parallel at IRSN and GRS in 2007, was released in July 2009 to SARNET partners, and recently improved through the release mid-2010 of its first V2.0-rev1 update. Therefore, while the series of ASTEC V1 versions were the reference one in the SARNET phase-1 project as well as for the realisation at IRSN of the L2PSA on French PWR 1300 MWe and at GRS of the L2PSA consolidation study on German KONVOI 1300 MWe PWR, the V2 series are now the new reference SA analysis tool in the SARNET2 project as well as for the L2PSA on French EPR which is starting at IRSN [99].

A summary description of various ASTEC models is reported below [104],[103],[99].

9.1.1.2 Description of ASTEC V2.0 code

ASTEC V2.0 models most of the physical phenomena involved in SA (except steam explosion and mechanical response of the containment). The ASTEC code structure is modular, each of its modules simulating a reactor zone or a set of physical phenomena (Fig. 5):

• CESAR module simulates the thermal-hydraulics in the primary circuit, secondary circuit and in the reactor vessel (with a simplified core modelling) up to the beginning of the core degradation phase, i.e. roughly up to the start of core uncovery, and in any case before the start of Zr cladding oxidation by steam. After the onset of the core degradation phase, the CESAR module computes only the thermal-hydraulics in primary and secondary circuit as well as in the vessel upper plenum. The CESAR thermal-hydraulics modelling is based on a 1-D 2-fluid 5-equation approach, accounting for both thermal non-equilibrium and momentum non-equilibrium between liquid and gas phases. Up to 5 non-condensable gases (hydrogen, helium, nitrogen, argon, oxygen) are available. The numerical approach is based on differential balance equations (mass, energy and momentum) and algebraic equation which models the interfacial drag between the liquid phase and the gas phase.

• ICARE module simulates the in-vessel degradation phenomena (both early and late degradation phases), including the thermal-hydraulics in the core and vessel lower plenum. This module, which is issued from the ICARE2 IRSN mechanistic code, allows to simulate the early-phase of core degradation with fuel rod heat-up, ballooning and burst, clad oxidation, fuel rod embrittlement or melting, molten mixture candling and relocation, etc. and then the late-phase of core degradation with corium accumulation within the core channels and formation of blockages, corium slump into the lower head and corium behaviour in the lower head until vessel failure.

• ELSA module calculates the release of fission products (FP), actinides and structural materials (Ag, In, Cd, Sn, Fe, Ni, Cr) from the degraded core. The ELSA modelling allows description of the release from fuel rods and control rods, followed by the release from debris beds (if any) and, then, the release from the in-core molten pool. The modelling is based on a semi-empirical approach and the physical phenomena considered are the main limiting phenomena which govern the release. For intact fuel rods and debris beds, the release of fission products is described according to the degree of fission product volatility (volatile, semi-volatile and non-volatile). Regarding the molten pool configuration, given the high-temperature conditions, chemical equilibrium can be assumed in the magma so that release is governed by mass-transfer and evaporation processes from the free surface of the molten pool.

• SOPHAEROS module computes the aerosol and vapour transport through the Reactor Cooling System (RCS) via gas flow to the containment. Using twelve families of species (elements, compounds, gas, volatile, non-volatile…) and five states (suspended aerosols, suspended vapours, vapour condensed on walls, deposited aerosols, sorbed vapours), the mechanistic or semi-empirical approaches model the main vapour-phase and aerosol phenomena. With regards to the vapour phase, the main phenomena taken into account are: equilibrium chemistry, chemisorption of vapours on walls, homogeneous and heterogeneous nucleation, condensation/revaporisation on/from aerosols and walls; moreover, a preliminary model for kinetics of gaseous phase chemistry is available too. The main phenomena considered for the aerosols are: agglomeration, turbulent diffusion, thermophoresis, diffusiophoresis, impaction in bends and constrictions, remobilisation of deposits and pool scrubbing.

• RUPUICUV module aims at evaluating Direct Containment Heating (DCH) i.e. ex-vessel discharge of hot corium into the cavity after lower head failure (involving vessel blow-down and cavity pressurisation) and potential corium oxidation and entrainment from the cavity to the containment. Two kinds of cavity are accounted for: one with an annular space around the vessel as in European PWRs and one with several intermediate compartments between cavity and containment as in USA PWRs.

• MEDICIS module simulates the Molten-Core-Concrete Interaction (MCCI) with a lumped-parameter 0-D approach with averaged melt/crust layers. This module assumes either a well-mixed oxide/metal pool configuration or possible pool stratification into separate oxide and metal layers. It describes concrete ablation, corium oxidation and release of incondensable gases (H_2, CO, CO_2) into the containment. The module is interfaced with the general physico-chemistry package for element speciation in a mixture, thermodynamic data (i.e. liquidus and solidus temperatures, mass and volumetric solid fractions) and thermo-physical properties (i.e. density, viscosity). Moreover, dedicated models are now available in the V2.0 version to account for the specifics of the EPR ex-vessel geometry (treatment of sequential MCCIs, first in the cavity and then in the spreading chamber, modelling of the corium pouring from the cavity into the core catcher, corium spreading …).

• CPA module is used for the simulation of containment thermal-hydraulics and aerosol behaviour. The module is based on a “lumped-parameter” approach. Most models are derived from former GRS codes RALOC and FIPLOC. The containment can be nodalised as several 0-D zones (connected by junctions and surrounded by walls) simulating simple or multi-compartment containments (tunnels, pit, dome…) with possible leakages to the environment or to normal buildings, with more or less large openings to the environment. The containment atmosphere heats up under the effect of sources of steam, FP gases and aerosols, and pressure increases. CPA describes phenomena such as gas distribution, pressure build up, hydrogen combustion and the behaviour of engineered safety systems such as passive autocatalytic re-combiners, sprays or other pressure suppression systems. With regards to the aerosol behaviour, the code describes phenomena such as volume condensation and growth of insoluble and soluble aerosol particles, behaviour of chemically different aerosol components, and agglomeration and deposition processes. Two main models are available in ASTEC V2.0 to simulate hydrogen combustion, namely the FLAME-FRONT models which account for the flame front propagation in a multi-compartment geometry (part of the CPA module) and the COVI model, based on AICC approach (which is managed as a separate module). In ASTEC, combustion occurs according to different criteria: user-input or crossover of flammability limits in the Shapiro diagram. For the latter, 4 different flammability limits determined at atmospheric pressure and room temperatureare defined on the ternary Shapiro diagram hydrogen-air–steam.

• <u>IODE module</u> deals with iodine and ruthenium behaviour in the containment. For iodine, the IODE module is composed of around 40 phenomenological models that focus on the predominant chemical reactions in sump, gas phase and at contact with surfaces and the effect of spray on molecular iodide. More precisely, it describes in a kinetic way (i.e. non-equilibrium) the chemical transformations of iodine in the containment reactor. As concerns ruthenium, the IODE module is focusing on the three predominant chemical reactions in gas phase.

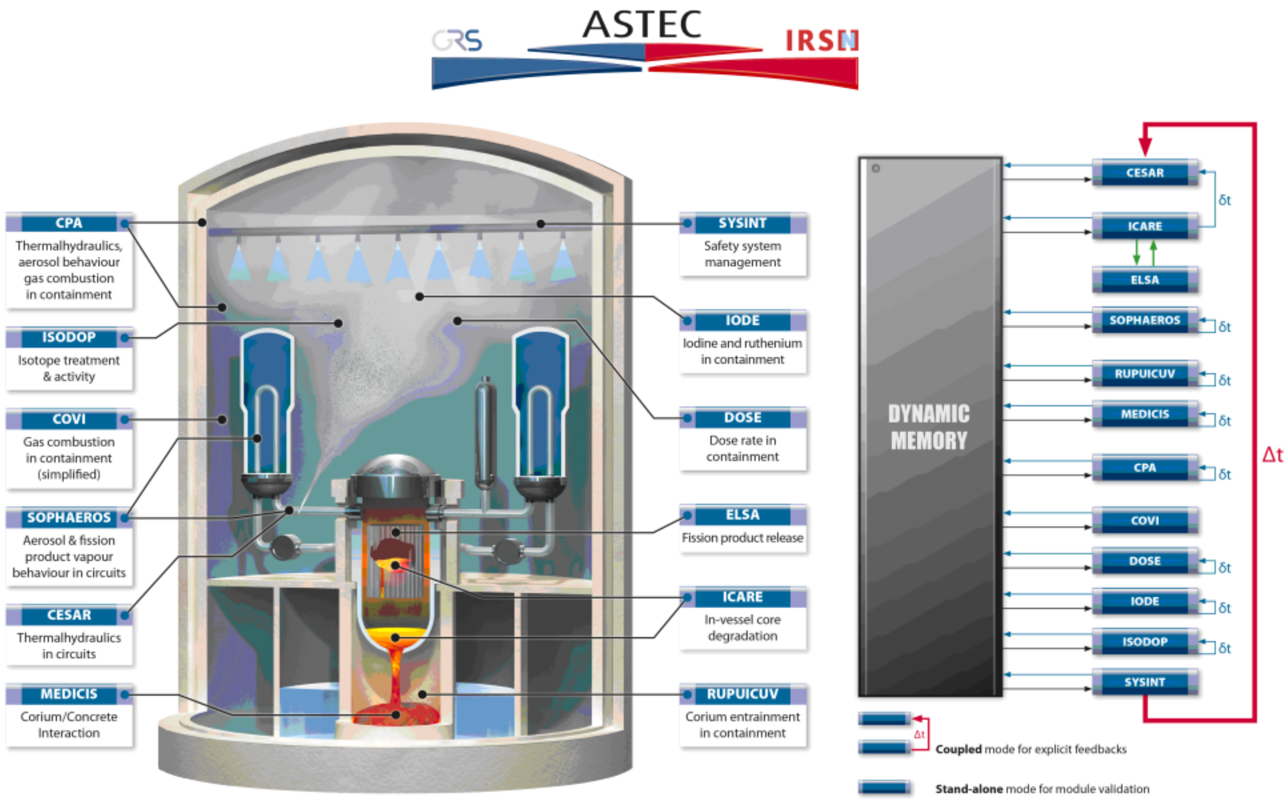

Fig. 5 Scheme of the ASTEC V2.0 modules, code structure and running model [99]

9.1.2 MELCOR

The MELCOR code [Gauntt et al. 2005] developed by Sandia National Laboratories (SNL) under the sponsorship of the United States Nuclear Regulatory Commission (NRC), is a fully integrated, full plant severe accident simulation code for the prediction of the progression of accidents in light water nuclear power reactors and other nuclear facilities. Originally designed to be a fast running PSA severe accident code using simplified parametric models, today, owing to significant advances in computing power, MELCOR now also serves the role of a best estimate code for predicting plant response to severe accident.

The code is intended to predict accident progression from the initiating event, to the point of core uncovery, through vessel failure and the expulsion of core debris into the containment, to the point of containment failure and the prolonged escape of radioactive materials into the nuclear power plant environment. The MELCOR code provides input to a companion code, MACCS, for the analysis of radioactive material dispersion in the environment and the consequences of this dispersion. The MELCOR code has a substantial, world-wide community of users. The code has a rather flexible architecture so that it can be used to predict accident progression in many different types of nuclear reactors. MELCOR is also applied to the prediction of accident progression in facilities for processing of nuclear materials especially for accidents involving fires.

The code is based on specially developed models for thermal hydraulics, core melt, fission product release and transport processes. A number of existing codes have been directly integrated into MELCOR architecture, these include CORSOR/CORSOR-M/CORSOR-BOOTH, VANESA, CORCON/MOD3, MAEROS, TRAP-MELT2, and SPARC-90 physics [NEA/OECD 2007].

With the consolidation of modelling capabilities from other NRC codes, MELCOR today stands as the repository of knowledge concerning severe accident and fission product release phenomena, benefiting significantly from important international research programs, including PHEBUS, CORA, QUENCH, RASPLAV, MASCA, ARTIST, HEVA, and VERCORS.

MELCOR is intended to be applied by the NRC for PSA studies for existing and advanced LWRs, best-estimate accident sequence studies to develop insights into physical phenomena and hardware performance, audit reviews of PSAs and accident management studies that analyse the progression of accidents and evaluate the detrimental and beneficial effects of various strategies.

MELCOR is used to assist the NRC in the design certification process for a number of new plant designs, including AP1000, ESBWR and the US-EPR, and to assist in the evaluation of numerous license amendment requests in the context of regulatory processes. Additionally, MELCOR is used as a code based means of conducting uncertainty analysis in Level 2 PSA applications.

The more important packages are listed in Table 28 with a description of various MELCOR models [NEA/OECD, 1997, 2007, 2009].

Table 28 Packages in the MELCOR computer code for reactor accident analysis [NEA/OECD 2009].

Symbol	Package Name	Description
EXEC	Executive	Responsible for overall execution control of the calculations
BUR	Burn	Models the combustion of gases in control volumes
CAV	Cavity	Models the attack on the basemat concrete by hot or even molten core materials
CND	Condenser	Models the effects of Isolation Condenser Systems and Passive Containment Cooling Systems found in some boiling water reactors
CF	Control Function	Allows users to modify the modelling in MELCOR by defining functions of variables in the MELCOR database and make the values of these functions available to other MELCOR packages
COR	Core	Calculates the thermal response of the reactor core, the lower plenum internal structures, core internal support structures and the reactor vessel lower head
CVH	Control Volume Hydrodynamics	Modelling of the thermal-hydraulic behaviour of liquid water, water vapour and gases in control volumes
DCH	Decay Heat	Models the decay heat power from fission products
FCL	Fan Cooler	Models the heat and mass transfer associated with operation of fan coolers in the reactor containment
FDP	Fuel Dispersal	Models fuel expulsion from the reactor vessel to the reactor cavity. This includes modelling high pressure melt ejection and the dispersal of core debris over several volumes
FL	Flow Path	Description of interconnection of volumes and the condensation or evaporation of water along flow paths
HS	Heat Structures	Models energy transfer to and within structures
MP	Material Properties	Models thermophysical properties of materials needed in the modelling done in

		other packages
NCG	Noncondensible Gas	Models noncondensible gases as ideal gases
PAR	Passive Autocatalytic Hydrogen Recombiner	Calculates the removal of hydrogen from the containment atmosphere caused by the operation of passive hydrogen recombiners
EN	Radionuclide	Models release, transport and behaviour of radionuclides
SPR	Containment Spray	Models heat and mass transport between spray droplets and the containment atmosphere

9.1.2.1 Thermal-hydraulic modelling

In MELCOR, the thermal-hydraulic processes are modelled by the Control Volume Hydrodynamics (CVH) and Flow Path (FL), while the thermodynamic calculations are performed within the Control Volume Thermodynamics (CVT) package. The CVH/FL packages are based on general control volume hydrodynamic network concept, which provide thermal-hydraulic boundary conditions to other MELCOR phenomenological packages.

Control volumes are interconnected via "flow paths" through which hydrodynamic material may pass without any residence time (assumption of negligible volume). The material and energy contents of both coolant and non-condensible gases are assumed to reside within control volumes. Mass and energy sources and sinks are treated as boundary conditions to CVH/FL.

In CVH/FL, hydrodynamic materials are assumed to be separated by gravity into a lower pool region (which may contain steam bubbles, but not non-condensible gases), and an overlying atmosphere (which may contain liquid droplets, gases, vapour). The mass exchange models include options for thermal and mechanical equilibrium model which assumes the same pressure and temperature for both pool and atmosphere, and thermal non-equilibrium model which assumes the same pressure, but different temperatures for pool and atmosphere (vapour superheat and liquid subcooling).

9.1.2.2 Core geometry and core melt modelling

The core and the lower plenum in MELCOR are divided into a number of user specified concentric radial rings and axial segments. A number of component types and materials are modelled.

A simple, candling model treats the downward flow and refreezing of molten core materials, thereby forming layers of solidified debris on lower cell components, which may lead to flow blockages and molten pools.

The code contains model for initial Zr melt formation and release, and subsequent fuel rod collapse and debris bed formation. Furthermore specific models for the release of Ag-In-Cd aerosol from damaged control rods and for the oxidation behaviour, particular to PWR-type boron-carbide control rods, are included.

Modelling for late phases of core damage provide for prediction of molten pools either in the core regions or in the lower plenum, accounting for molten fuel pool natural convection, perimeter pool crust formation, and separation of pool components in metallic and ceramic molten phases.

Failure of the core structures such as the core plate, as well as lower head heat up and failure followed by debris ejection, are treated by stress-based failure models accounting for creep failure modes as well as temperature criterion.

9.1.2.3 Other physical processes

Besides the processes already mentioned, MELCOR includes models for: the forming of non-condensible gases, combustion of gases, the thermal-hydraulic part of core-concrete interactions, and direct containment heating.

With regards to the interaction of the debris released from the vessel with the concrete basemat in the cavity, the code calculates the rate of erosion in the concrete basemat, the temperature and composition of the molten layers, the temperature, flow rate and composition of gases such as CO_2, CO, H_2, and water vapour evolving from the concrete.

Heat is exchanged between the melt and the concrete, layers of the melt, the top surface of the melt and the atmosphere, water (if any) and the structures above it. The melt concrete heat transfer includes options for a gas film model and an intermittent film model. The concrete ablation products (i.e. steam and CO_2) are modelled to react with the un-oxidised metals present in the melt.

9.1.2.4 Radionuclide behaviour

The aerosol model includes the release of aerosols and vapours from the core materials and from core - concrete interactions. During the heat up phase of the accident, additional fission products are released by vaporisation or other thermally activated process. In addition, materials from structural cladding and control rods heat up, vaporise and leave the core.

Transport of aerosols and vapours between control volumes occurs with the bulk fluids, gases or water, with zero slip, and aerosols can be removed as they pass through water suppression pools. User-specified chemical reactions can be treated, which should be based on the results of more detailed codes or on experiments.

Aerosol transport calculations are performed to determine: the suspended mass concentration as a function of time, the size distribution of airborne particles as a function of time (mass concentration of water and particles in each size class), the cumulative settled out quantity, the cumulative plated out quantity and the cumulative leaked out masses.

The phenomena treated include: agglomeration (random movement, gravity and turbulence), removal (random movement, gravity, movement in a condensing steam, thermophoresis and sprays), steam condensation onto aerosols, and homogenous nucleation of water droplets.

Models for chemical behaviour of Iodine exist, but they have been applied in Level 2 PSA only to limited extent until now.

9.1.3 MAAP4

The Modular Accident Analysis Program (MAAP) Version 4 is a computer code that simulates the response of light water and heavy water moderated nuclear power plants, during severe accident sequences.

MAAP4 is an integrated code with capabilities to calculate the thermal-hydraulic response of the core, the RCS, the containment and the auxiliary buildings, as well as the fission product release, transport and deposition during postulated severe accident conditions.

MAAP was developed and maintained by Fauske & Associates Incorporated (FAI), since the beginning of the code in 1981, under the sponsorship of the Electric Power Research Institute (EPRI) and the MAAP Users Group (MUG). The code continues to be developed and maintained by FAI. The new version of MAAP4 (MAAP4.0.7) was released by FAI in January 2008.

Validation of MAAP4 was performed against HDR experiments, CORA tests, TMI-2 accident, CSTF tests, PHEBUS FPT0 test, ORNL VI test series, SFD tests at INEL, AP600 OSU tests, and LOFT experiments [NEA/OECD 2009].

There are parallel versions of MAAP4 that support BWRs and PWRs and unique versions VVER, CANDU, and ATR designs. Models for ALWR plant designs, including their passive features were also implemented, benchmarked, and accepted for design certification.

The code was subjected to independent design review and it was also reviewed by the US NRC. MAAP was compared with other codes on: pertinent aspects of severe accident phenomena (i.e., core melt progression, source term estimates for plant applications using MELCOR), containment response (GOTHIC), and mass and energy releases for small and intermediate LOCA break sizes (RELAP). MAAP was also benchmarked against a variety of integral and separate effects experiments.

Accidents was analysed for a variety of transients, including Loss of Offsite Power (LOOP), Loss of Coolant Accidents (LOCAs), Main Steam Line Breaks (MSLBs), bypass, mid-loop operation, and shutdown sequences.

The code is used for many PSAs, especially for most of the U. S. Individual Plant Examinations (IPEs) and for studies supporting the development and implementation of Severe Accident Management Guidelines (SAMGs) [NEA/OECD 1997, 2007].

MAAP4 also includes a graphical interface, MAAP4-GRAAPH, enabling the user to interactively interface with the code during execution, to modify the status of on-site power, pumps, valves, etc., as well as, to analyse the results.

The principal characteristics of the MAAP4 code are illustrated in the following sections [NEA/OECD 2007, 2009].

9.1.3.1 Thermal-hydraulic modelling

MAAP uses a control volume and flow path approach in which the geometry of the control volumes (called regions) is pre-specified and different for a PWR and a BWR. The BWR version has 8 control volumes for primary system gas flow and the PWR version has 14 plus the pressuriser and the quench tank. The reactor containment building has an arbitrary user-defined nodalisation. The BWR and PWR primary systems are divided into regions: upper and lower plenum, reactor core, downcomer, and for PWRs, (un-)broken cold and hot legs, and (un-)broken steam generator loops. Separate mass and energy conservation equations are solved for each of the regions.

For the containment analyses, the containment model provides a generalised description of the containment, such that the nodalisation can be specified by the user. In addition, the containment model considers counter-current flows and plume behaviour, which are influential in containment stratification and mixing, as well as fission product transport. The containment models for the advanced plants represent those features typical of the ALWR designs, including passive systems and passive hydrogen recombiners.

Flows consist of steam, water, hydrogen, other non-condensible gases, aerosol and corium. Flow paths can describe pipes, surge lines, penetrations, relief valves, and general openings. Flow rates are determined from quasi-steady momentum balances. Separate mass and energy conservation equations are solved for each ordinary differential equation.

9.1.3.2 Core geometry and core melt modelling

The core is divided into radial rings (up to 7) and axial rows (up to 50). Once the core is uncovered, it can overheat sufficiently to result in rapid oxidation of the Zircaloy or stainless steel cladding. In MAAP4, control rod material can relocate downward away from the fuel prior to fuel relocation. In addition, the MAAP4 models include the process of dissolving the uranium dioxide fuel with molten zirconium and the relocation of eutectic material.

If the accident sequence being considered results in reflooding of the reactor core once core degradation has occurred, the MAAP4 models address this reflooding process and the potential for quenching the core debris, both within the original core boundaries and in the reactor pressure vessel lower plenum. If water is available on the exterior of the Reactor Pressure Vessel (RPV), the influence of external cooling in removing energy from the vessel wall and in preventing the potential creep rupture of the vessel due to core debris thermal attack on the vessel lower head, is modelled.

9.1.3.3 Other physical processes

MAAP has a model for flammability which depends upon the gas mixture composition and temperature, a model for combustion completeness in case of incomplete combustion, and a model for burn time. Flame propagation between compartments is also treated. MAAP also considers "jet-burning" (i.e., ignition of a hot jet containing flammable gases that enter a compartment with oxygen available) and auto-ignition of gases at high temperature.

The additional models include the RPV and penetration failure models, the molten debris heat transfer model, a jet entrainment model for the debris fragmentation in the RPV lower plenum, an optional debris dispersal model, a two-dimensional core-concrete interaction model, the RPV external cooling model, direct containment heating and the in-vessel debris cooling model.

9.1.3.4 Radionuclide behaviour

MAAP models the transport and retention of fission products in the RCS and generalised containment. The materials released from the core are divided into 12 fission product groups, divided according to chemical characteristics. The fission product can exist in the solid, liquid and vapour form. Furthermore three chemical compounds which affect the pH value in the water pool are tracked by the code.

The aerosol model considers the combined effects of agglomeration and removal mechanisms, including gravitational sedimentation, condensation removal, inter-compartmental transport, thermophoresis, diffusiophoresis, and impaction. Revaporisation is included as transfer between the states.

9.1.4 THALES-2

The THALES-2 code is an integrated severe accident analysis code developed at the Japan Atomic Energy Agency (JAEA), formerly JAERI (Japan Atomic Energy Research Institute) to simulate the accident progression and transport of radioactive materials for the PSA of nuclear power plants. In 1982, JAERI developed, as a first step, the computer code system THALES (Thermal-Hydraulic Analysis of Loss of Coolant Emergency Core Cooling and Severe Core Damage) for the analysis of accident progression. In 1988, the code was combined with the ART (Analysis of Radionuclide Transport) code developed also by JAEA and the THALES/ART code system started. After that, the code system was improved by coupling the radionuclide transports models with the thermal hydraulic ones and a prototype of single code, namely, the THALES-2 code was completed in 1991. Then, the abbreviation THALES was changed to the Thermal Hydraulics and radionuclide behaviour Analysis of Light water reactor to Estimate Source terms under severe accident conditions [NEA/OECD 1997, 2007].

The code was also validated through analyses of experiments and comparison with other computer codes. The THALES-2 code currently consists of BWR and PWR versions [NEA/OECD 2009].

A summary description of various THALES-2 models is below reported [NEA/OECD 1997, 2007, 2009].

9.1.4.1 Thermal-hydraulic modelling

The thermal hydraulic model of THALES-2 is based on control volume and flow path approach. Each volume is further divided into a gas region and a liquid region by a mixture level. For junctions a counter-current flow model can be applied.

In the thermal hydraulic calculation of the volume, the conservation equations of mass and energy are solved but the momentum calculation is not performed to reduce the computation time. The basic assumptions adopted in the calculation are uniform pressure and thermal equilibrium in a volume. The system pressure is determined to keep the total system volume constant, and the temperature in each volume is determined from the mass and energy conservation law.

9.1.4.2 Core geometry and core melt modelling

The core is represented by groups of fuel assembly (maximum 5) and vertical nodes (maximum 25). Fuel rods begin heat up when they are exposed to the steam over the mixture level. The model allows the simulation of the core heat up, modelling the Zr-water reaction, cladding oxidation occurs and hydrogen generation, core meltdown, fuel rods fragmentation, corium slump into the lower head and corium behaviour in the lower head until vessel failure.

9.1.4.3 Other physical processes

Models are provided for metal/water reaction, molten fuel relocation, debris relocation to selected containment volumes at the reactor vessel failure, hydrogen burning, core /concrete interaction at each location to which debris dispersed. Actuation logics of various plant systems and operator actions can be simulated. Containment pressure and temperature rise with blow down of the primary coolant, gases generated by concrete decomposition and hydrogen burning, can be also taken into account.

9.1.4.4 Radionuclide behaviour

In the code, 20 radionuclides are classified into several groups (maximum 10) in terms of their chemical characteristics. Typical elements or compounds of each group are Xe, CsI, CsOH, Te, Sr, Ru, La and other particulates. An aerosol form is assumed in the code for Sr, Ru and La because their vapour pressures are very low in severe accident conditions.

For radionuclide release from fuel before the reactor vessel failure (in-vessel release), the CORSOR model and the new model with pressure effect proposed by the VEGA program, are applied to calculate release rates of radionuclides. After the vessel failure (ex-vessel release), an empirical model is used to calculate generation rates of aerosols of concrete components during core/concrete interaction. In addition, the CORSOR model is also applied to calculate release rates of radionuclides during the ex-vessel release.

In this code, radioactive materials can take the form of gas, aerosol, deposit on structure walls and floors, and solution in water. The code solves the governing equations for multi-component aerosol, taking into account the size growth by agglomeration and condensation/evaporation of steam and volatile materials on the aerosol. Models are provided for various transport processes, including the condensation/evaporation and chemical absorption of the gas species at structure surfaces, deposition of aerosol to walls and floors, removal by sprays and filters, scrubbing by water pools, and convection by liquid as well as gas flow.

9.1.5 ECART

ECART (ENEL Code for the Analysis of Radionuclide Transport) is an integrated primary circuit and containment code, for nuclear power plant severe accident analysis, but it can be also applied to fusion reactors, industrial plants etc [Parozzi et al. 2006].

The work on ECART started in 1989, and utilities ENEL and EdF contributed to its initial development. ECART is presently developed by ERSE.

ECART architecture consists of three main sections, coupled in an explicit way and also able to be activated als as stand-alone modules: a thermal-hydraulic section providing the boundary conditions; an aerosol-vapour section calculating the transport of radioactive or toxic substances; a section evaluating the chemical equilibrium among airborne compounds and some plant specific reaction kinetics between gases and solid materials.

For accidents with fires within closed environments, specific models can simulate both thermal and chemical processes, accounting for combustion of gases and solids, as well as pool fires. The radiative heat transfer and the action of water sprays on atmosphere cooling and aerosol removal are properly taken into account, as verified by comparing the code predictions to full-scale experiments and to the consequences of actual fire accidents.

ECART belongs to the category of "eulerian" and "mechanistic" analysers. Eulerian because it traces the transport of radiotoxic species taking the plant as the reference system, to give, as a function of time, concentrations and physical forms along the followed pathways. Mechanistic because it follows, whenever possible, physical and chemical laws, avoiding the use of assumptions of limited applicability.

An interesting feature of the code useful in experimental analysis is its ability to accept incomplete thermal-hydraulic data e.g. data specified only at certain junctions or boundaries, and then to use its internal calculation capabilities to complete the thermal-hydraulic conditions required by the aerosol and chemistry sections of the code [NEA/OECD 2009].

Validation studies have used data from the ATT-Marviken V, LACE, DEMONA, VANAM, STORM and PHÉBUS programmes. Recent applications of ECART have been to aerosol resuspension and chemical reactions in PWR circuits, and transient analyses of the fusion experiment ITER [Parozzi et al. 2006].

The principal models used and the phenomena considered are listed in Table 29 Parozzi et al. 2006, NEA/OECD 2009].

9.1.5.1 Thermal-hydraulic modelling

ECART was set up to treat the pure transport phenomenology through generic flow systems with Eulerian approach. It requires dividing the analysed pathway into a series of control volumes connected by flow junctions. Within each volume, the code can simulate two-phase flow under stratified regime, with possible formation/fallout of suspended water drops. The gas phase is treated as perfectly mixed. The interaction between the fluid and the walls, as well as the thermal conduction within the wall materials, is also calculated.

This thermal-hydraulic section provides the solution of mass, energy and momentum balance equations to give a realistic representation of the fluid flow, allowing for counter-current flow conditions at junctions, gas pressure and temperature, heat transfer to the circuit and containment structures, as well as water pool levels.

The steam condensation is modelled splitting bulk and wall condensation, which influence, respectively, the aerosol growth and the aerosol deposition by the mechanism of diffusiophoresis.

9.1.5.2 Radionuclide behaviour

The main phenomena that can influence the retention or re-entrainment of radioactive or toxic substances can be taken into account, firstly detecting the chemical speciation and then the interaction among vapours, aerosols and the wall surfaces. Aerosol transport mechanisms accounted are: growth, agglomeration, deposition, scrubbing and resuspension.

Irreversible sorption of I, I_2, HI, CsOH, Te and Te_2 vapours onto structure surfaces and airborne particles are also modelled by adopting experimental correlations.

Moreover, the decay heat of most powerful elements undergoing transport processes (Kr, Rb, Sr, Mo, Ru, Ag, Sn, Sb, Te, I, Xe, Cs and Ba) can be taken into account through time-dependent correlations giving the β and γ specific power for typical LWR shutdown cases, and distinguishing between the absorption behaviour of the gas and liquid phases, and the structures.

Table 29 Principal models used and the phenomena considered in ECART code [103]

	Mechanism	Literature source and /or brief description
Thermal - Hydraulics	Transport of carrier gas/liquid mass, momentum, Energy	Control volumes, each with a liquid and a gas volume in equilibrium; 1-d and 2-d connection of volumes; Bulk and wall condensation split;
	Exchanges with structures	Included
	Vapour/gas transport	Secondary gases e.g. nobles, accounted for
	Sprays and sprinklers	Ad-hoc Lagrangian model accounting for droplet size distribution and injection speed
	Pool scrubbing	Included within control volume
	Gas combustion (hydrogen and others)	Accounted for through equilibrium chemistry
	Fires and explosive aerosol clouds	Models of pyrolysing solid surfaces, pool fires and detection of explosive aerosol clouds
	Radiative heat transfer from flames	View factors among flames and structures calculated with Monte Carlo method; Aerosol cloud absorption accounted for.
	Decay heat	13 most powerful elements accounted for
Vapour phenomena	Vapour-phase chemistry	Equilibrium with 126 reacting species (including carrier gases)
	Homogeneous nucleation	Not modelled (source seed required)
	Heterogeneous nucleation	Not modelled (source seed required)
	Sorption on surfaces (one-way)	Selected species/surface combinations e.g. irreversible sorption of I, I_2, HI, CsOH, Te and Te_2 vapours on steel
	Condensation/Evaporation onto/from surfaces and aerosol particles	Calculated by diffusion equations

Aerosol Phenomena	Transport	Well-mixed within each volume. Corrections for components with concentration gradients e.g. long pipes; Discretised size distribution; simplified multi-component description (composition accounted for each size bin in each volume, for both airborne and deposited particles)
	Aerosol shape	Aerodynamic and collision shape factors
	Particle growth	Includes hygroscopic behaviour, Kelvin effect
	Settling	Stokes and non-Stokes regimes
	Turbulent impaction	Liu-Agarwal data
	Diffusion	Davies, Gormley-Kennedy
	Thermophoresis	Brock correlation with Talbot coefficients
	Diffusiophoresis	Schmidt-Waldmann
	Bend impaction	Stokes and non-Stokes regimes; size-dependent trapping in narrow bends
	Agglomeration	Brownian (Smoluchowski); Gravitational; Turbulent (Saffman and Turner);
	Mechanical re-suspension	Modelled through experimental correlation
	Aerosol fall-back	Accounted for
	Scrubbing in water sumps	Lagrangian model accounting aerosol depletion within rising bubble

9.1.6 Reference

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9.2 EVENT TREES CODES

9.2.1 EVNTRE

9.2.1.1 Introduction

Several approaches exist to perform Level 2 Probabilistic Safety Assessments (PSAs) for Nuclear Power Plants (NPPs). One widely accepted methodology is based on an Accident Progression Event Tree (APET), which was first introduced in NUREG-1150 [108]. The APET consists of a large number of questions determining the containment performance and/or the fission product release categories with the corresponding probabilities of numerous possible severe accident sequences. Once the initial frequencies and the split fractions of all the questions are determined, the EVNTRE code [109] can be applied to calculate the frequencies of the Plant End States (PES) and Source Term (ST). This FORTRAN-77 based program is also used to perform additional sensitivity studies and uncertainty studies on these results. In this appendix a short description of the code is given, together with the advantages and disadvantages, and the application of EVNTRE in the Belgian Level 2 PSA update.

9.2.1.2 Description

To run the EVNTRE executable several input files in an unformatted text form are required. The main parameter influencing both the accuracy of the calculation as the duration is the cut-off frequency, defining the minimum probability linked to a path through the tree. Another input file, the Tree Definition File, contains the whole APET structure with quantified split fractions for each question and the initial frequencies of the Plant Damage States (PDS), resulting from the Level 1 PSA work. In the Binning and Sorting Definition File, the necessary output must be requested before the actual run. In [109], it is explained that, given that a certain path through the tree can be represented by a state vector with an element for each question, a 'bin' can be considered a transformation of those state vectors into a smaller vectors corresponding to only one or a few questions. By 'sorting' it is referred to the order of the bins in the output tables to check for possible correlations between those bins.
The structure of the APET, defined in the Tree Definition File, consists of a number of questions, each containing one or more branches. The split fractions or the probabilities corresponding to the branches of each question can be defined in several ways: they can be fixed or independent of any other question; they can be dependent of one or more questions or, in other words, a certain path through the tree; or they can be determined by a user function. Notice that in the first two cases, only some values are altered in the Tree Definition File, while in case user functions are modified, changes in the FORTRAN source code of the EVNTRE executable are necessary.
The main output file of an EVNTRE run is the Binning and Sorting Report File [109], starting with a table of all the bins that are defined and the corresponding probabilities followed by the total frequency of all the bins and the total frequency of the lost sequences due to the cut-off frequency. If the fraction of the lost sequences is too high, one can decide to decrease the cut-off frequency; while if, on the other hand, the calculation time is too long, one can decide to increase the cut-off frequency. As such, the preferred value of the cut-off frequency is found by an iterative process. The headers of the subsequent tables are in accordance with the sorting order, requested in the Binning and Sorting Definition File. This tool makes it possible for the analyst to find correlations between different bins and questions, e.g. one can investigate for which initiating events the most hazardous ST are more probable.

Finally, it is mentioned that the input and output files of the EVNTRE run of the Peach Bottom APET of NUREG-1150 [108] are provided in NUREG/CR-5174 [109]. This APET consists of 107 questions and serves as a good example for other large APET approaches with the use of the EVNTRE code.

9.2.1.3 Advantages

EVNTRE can be described as a simple and flexible event progression analysis code, mainly useful for Level 2 PSA application. Due to the many possibilities in defining the split fractions of each question, complex tree-like or even network-like structures can be realised. In particular, the implementation of user functions in the FORTRAN source code offers the opportunity to build small models in the tree. For instance, different contributors to a pressure increase in the containment, such as core-concrete interactions and hydrogen burns, can be added and compared to a simplified containment fragility curve to determine the outcome of a question determining the containment structural failure.

In addition, sensitivity and uncertainty studies can be easily performed by a rerun of the executable with different values for certain split fractions or by a number of runs with values determined by Monte Carlo or Latin Hypercube sampling.

9.2.1.4 Disadvantages

Nevertheless, the simplicity and the flexibility of the program results in a less user-friendly environment. Furthermore, the output files are data sheets in an unformatted text with no graphical representations. Consequently, the output must be interpreted with care and the debugging of the input files requires a meticulous checking. As mentioned, not all the input is written in the input files, but the application of user functions require programming in FORTRAN.

Finally, it is noted that the calculation time increases significantly with decreasing cut-off frequencies, but this is an intrinsic drawback of all event progression analysis codes if a high accuracy level is necessary.

9.2.1.5 EVNTRE in the Belgian Level 2 PSA update

In the framework of the Belgian Level 2 PSA update, a generic APET has been developed for all Belgian units to evaluate the Containment Performance (CP-APET) and the Fission Product (FP-APET) release categories for a representative range of severe accidents. As a generic and large APET was constructed, it was decided to apply the EVNTRE code for calculating the PES and ST frequencies.

Due to this generic approach, the Tree Definition File is written in Excel which makes a simple implementation of the adaptations to the initial frequencies and split fractions possible for each specific NPP or for the purpose of sensitivity or uncertainty studies. This is possible because the structure of the APET remains unchanged, only the values coming from the quantification process or the expressions of certain user functions can change.

In addition, the flexibility of the code in defining the split fractions, particularly in making the split fractions of one question dependent on a range of other questions, makes the code very useful in case the quantification of the APET is performed by diverse means, e.g. in case of expert judgement the considered issue can easily be made dependent on other phenomena or, in the APET terminology, questions, if considered necessary by the expert panel.

9.2.1.6 Conclusions

In case a Level 2 PSA is based on a large APET approach with a complex tree structure, the EVNTRE code can be a very useful tool. Furthermore, it is possible to implement small models of overall phenomena by user functions. The main drawback is that the program is not very user-friendly.

9.2.1.7 References

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9.2.2 SPSA Level 2 PSA code

Radiation and Nuclear Safety Authority of Finland (STUK) started the development of living PSA computer code (SPSA) in 1988. Level 1 part of this code was taken into trial use in 1991. The initial development of an integrated Level 1 and Level 2 methodology was completed in early 1993. After further validation, two level 2 pilot models (one BWR[1], one PWR[2]) were constructed in 1995. They demonstrated the feasibility of the level 2 approach. In 1997, TVO power company completed the first version of Level 2 PSA using SPSA code. The Olkiluoto 3 Level 2 PSA is also performed with the SPSA code.

9.2.2.1 Containment event trees

The Plant Damage State (PDS) cutsets or PDS frequency distribution form the starting point of the Containment Event Trees (CET). A CET can also start from any other event, represented by a numeric value or distribution, but the PDS cutsets provide a tight and two-directional link between Levels 1 and 2, since they automatically import Level 1 accident sequence data to CETs. This allows results of Level 2 analysis to also be stated in terms of Level 1. E.g. the importance of level 1 accident sequences based on weighted Iodine release are available, which can in turn be expressed in terms of cutset importance for Iodine release or basic event importance for Iodine release.

The CETs contain a new modelling methodology, which links together parametric dynamic models describing plant physical behaviour and the probabilistic computations. A CET model consists of a graphical event tree and associated functions that are described in specialised CET programming Language (CETL). The CETL compiler and run-time system are integrated in the code.

SPSA allows high flexibility in modelling. A CET can contain fixed values, distributions, minimal cutsets, dynamic parametric models of the process behaviour, and computation of probability values. This allows flexible sensitivity analysis, since the user can modify any parameter and see the effects of the modification in the CET. SPSA can also load external user-written routines and libraries (as a DLL). The use of functions makes it possible to model non-coherent or one-directional dependencies or/and time-dependency. Thus, the CETs or individual functions can be either dynamic or static, while the run-time system takes care of positive time (no backward dependencies).

For each PDS, a CET is developed as shown in figure 1 (for Olkiluoto 1 High Pressure Transient). The elements of a CET are initial conditions, sections, branch points and functions. The number of sections is not the same as the number of questions in the CET, since each section can contain a large number of conditional statements, thus hiding the complexity of the CET.

Under initial conditions, the plant damage state is described with probability values, process parameters and any other desired values. It contains also global variables and functions that are available in all remaining sections.

Under sections, the related functions for each branch are defined. In figure 1, section RECO contains functions OK_RECO (successful recovery), NO_RECO (no recovery) and RECRIT (re-criticality during recovery), which return the conditional probability for the branch, based on process parameters or any other parameters. One function returns 1.0 minus the sum of other results (called NIL in CETL). A number of variable and function types are available (e.g. real, integer, Boolean, distribution, string, table, cutset list). For variables, special properties can be defined (e.g. source term variable, collect during simulation, include in correlations, use in classification of release categories).

After computation, the user can view the process parameters and all other global variables at any point in the CET. If a syntax error (e.g. type conflict, undeclared identifier) is found during compilation, the editor is activated at the error position with the relevant error message. During CET computation, a number of run-time checks are performed. If a run-time error occurs (e.g. index of vector out of range, division by zero), the editor is activated at the error position with the relevant error message. A special <u>common section</u> contains variables and functions that are available to all CETs. It can contain process variables, source term definitions, parametric models, classification and binning rules, etc.

Numeric variables in a CET can be point values or distributions. The uncertainty analysis can be done with DPD, Monte Carlo simulation or any combination of these. The result analysis part contains a features from chaos theory, to identify dependencies and gain insight into correlations (during the development of SPSA, they revealed weaknesses in the random number generator, which was then replaced!).

SPSA offers also a general tool to PSA analyst: he can use only the CETL with a large number of reliability and other functions as a general "reliability problem solver".

9.2.2.2 <u>Risk integrator</u>

The risk integrator manages the uncertainty analysis and combination of the results from individual CETs. In the risk integrator, CETL can be used to introduce new variables and functions, and new binning rules can be defined. Complete uncertainty analysis and risk integration can be performed in one run. The risk integrator automatically detects modified CETs. The number of simulations for each CET is also controlled in the risk integrator. If the number of simulations is not the same in all CETs, the results are automatically retaken. After running an uncertainty analysis, SPSA automatically performs the binning followed by statistical and correlation analyses. The results are written to an output file (optional, since the output can become very large, containing thousand of tables and graphs) and stored for viewing and printing with the hierarchical viewer.

The risk integrator takes the whole data set produced by uncertainty analysis and classifies the data according to definitions of release categories. The grand total result of the Level 2 PSA is not affected by release category definitions, since the release categories are extracted from the grand total, and not the other way around. It is possible to create several sets of classification rules, one of which may active at a time.

9.2.2.3 Olkiluoto 1 Level 2 PSA model

Accident progression

SPSA does not contain any built-in level 2 physical models or source term models. Instead, it provides a high-level programming language and event-handling environment for building the models. Thus, any modelling detail below is more a feature of the Olkiluoto 1 model than a feature of the SPSA code. The available building blocks in SPSA are 'normal' programming functions, mathematical functions, a large number of probabilistic functions (analytic, Monte Carlo and DPD) and special functions like user-definable release class binner. In the Olkiluoto 1 model, no external libraries (DLLs) were used.

The level 2 model of Olkiluoto 1 consists of 11 CETs for full-power operation and 1 for refuelling. The CETs contain altogether approximately 10 000 lines of CETL models. The CET in figure 1 represents High Pressure Transient of Olkiluoto 1. Although the CET contains only 13 headers, there are tens of IF-THEN-ELSE statements that form additional questions.

Nearly all values in the Olkiluoto 1 CETS are distributions. Very few point values are used, and most of the comparisons in IF-THEN-ELSE statements are based on distributions. In SPSA, Monte Carlo simulation is the 'standard' quantification, but point value quantification can also be made.

Each CET contains variables, tables and functions, describing the related events, phenomena and associated probabilities. The purpose of the functions is not to model the accident progression in detail; i.e. a CET is not a thermal-hydraulic model. Instead, most functions model the results of thermal-hydraulic analyses, including uncertainties and dependencies. For this purpose, a large number of thermal-hydraulic analyses have been made. However, there are also functions that model the behaviour of parameters as a function of time or events (e.g. water level in lower drywell is a function of flooding event and time). SPSA can be considered as an expert tool, where the results of various analyses can be expressed in several forms and integrated into an overall deterministic-probabilistic model.

Each CET contains relevant initial values for process parameters, timings and reliability parameters. During the computation, the process parameters and timings are adjusted or calculated according to the events. An example of a function for successful lower drywell flooding is shown in listing 1. The function determines the start time for flooding *TiFlSt* based on the distribution of Accident Management (AM) timing distributions. These are in turn based on analysis of the AM tasks and on the results of thermal-hydraulic analyses. Once the gravity-driven flooding is started, it is assumed to occur at constant rate, being completed in 1300 seconds after initiation, and the flooding is finished at time *TiFlEnd*. Successful flooding is indicated by setting *LDFW* to true (see listing 1).

The functions thus describe the results of thermal-hydraulic analyses and physical dependencies with associated success/failure probabilities and uncertainties. As an example, listing 2 describes the lower drywell basic pressure as a function of time in case of reactor overpressurisation with pedestal flooding at 60 min. and containment water filling. The pressure development is approximated from MAAP calculations. Dynamic pressure loads and other variations (for example steam explosions, DCH etc.) are handled separately and added to the basic pressure when they occur. The dynamic pressure is then compared to the temperature-dependent pressure tolerances of containment weak points to determine containment failure probability.

Table 1 shows the issues related to containment failure that is treated in the model. Due to modelling containment weak points and their temperatures, it is possible to compute the distribution for containment failure location, as shown in table 2. The data columns of the table are, in order:

- % of all accidents: Code Damage (CD) plus Fuel Cladding Failure (FCF);
- % of CD accidents;
- % of FCF accidents;
- % of containment failures cases in CD accidents;
- % of containment failures in CD on power operation only.

When computing the probability of failure, the temperatures of weak points of containment are computed. Their failure probability distribution is a function of temperature and pressure. The pressure load at any moment is the sum of basic pressure and other loads.

Source term calculation

The source term model of Olkiluoto 1 is somewhat like the XSOR model (time-step methodology). It has 4 control volumes with dynamically varying volumes (e.g. the gas volume of lower drywell changes according to the amount of water in it; e.g. during flooding). Figure 2 shows flow paths during different time intervals before containment failure in transient initiated sequences. The dashed arrows that go to reactor building and environment represent containment leakage.

Since the computation of the source term is based on the parameters describing the accident progression, the source term is computed for each endpoint of a CET for each simulation run. Performing 1000 simulations for a CET with 80 endpoints gives 80000 source term samples.

A variable becomes a source term variable by being declared as SOURCE. Each source term sample is classified with user-defined CLASS variables and binning rules. In addition to the source term and CLASS variables, all variables defined for collection and/or correlation-pair computation are stored as a part of the source term and analysed statistically. Olkiluoto 1 SOURCE and CLASS variables are listed in table 3. The binner can either combine or split CET end points to release categories.

The source term computation routine has a large number of process/containment parameters and timings available. Some examples of used parameters are the following:

- Release, revapourisation and deposition rates for different nuclides (as a function of pressure, water levels and atmospheric conditions etc.).
- Decontamination factors of filter and pools (as a function of pool depths and temperatures).
- Containment initially bypassed.
- Containment failure time and location.
- Containment filling start time, end time and flow rate.
- Lower drywell flooding start time, end time and flow rate.
- Start time of diaphragm floor leakage.
- Condensation pool saturation time.
- Start and end time of containment sprays.
- Time of venting.
- Running time.

As the source term model is a time-step model, it is also possible to plot the development of the source term or any variable as a function of time. All simulation data, including points and graphs, can be exported to other tools for further analysis.

D. Risk integration

In the Olkiluoto 1 model, no additional computations have been programmed in the risk integrator. Thus, the integration is just a combination of the results from individual CETs. The results are computed from many points of view to gain insight into contribution from different PDSs, individual CET sequences and correlations. An example of a density function is shown in output 1.

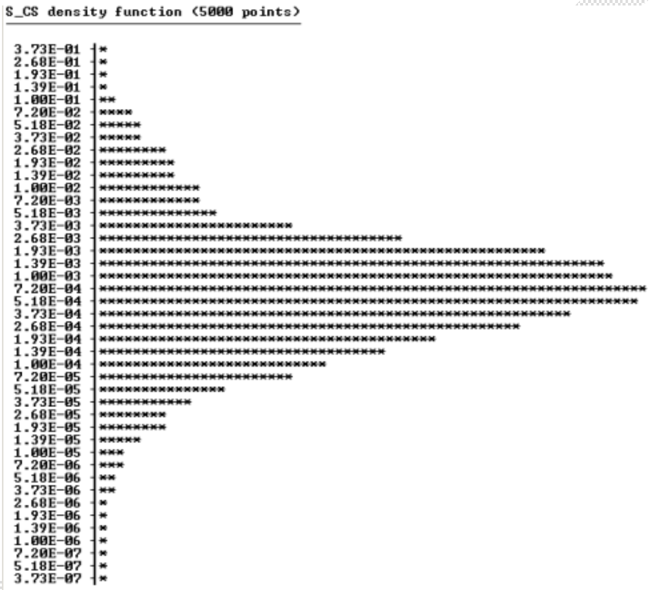

Fig. 6 Output 1. An example of a density function.

The corresponding percentiles and individual accident sequence contributions are shown in output 2. The contributions of individual sequences or whole CETs can further be examined with the interactive result viewer.

```
Percentiles: 5000 points

  0.1   8.64E-07  Min value:  2.70E-07
  1.0   6.08E-06
  2.5   1.46E-05
  5.0   2.96E-05  P50/P5   :  1.88E+01
 10.0   6.39E-05
 20.0   1.44E-04
 30.0   2.45E-04
 40.0   3.82E-04  Mean     :  3.38E-03
 50.0   5.56E-04  StDev    :  1.45E-02
 60.0   8.09E-04  P95/P5   :  4.58E+02
 70.0   1.21E-03
 80.0   1.87E-03
 90.0   4.72E-03
 95.0   1.35E-02  P95/P50  :  2.43E+01
 97.5   2.81E-02
 99.0   5.82E-02
 99.9   2.13E-01  Max value:  3.31E-01

CET sequence fractions    Raw%     Weighted%

Seq 7                     49.03     82.26
Seq 9                      0.76      0.16
Seq 33                    49.44     17.54
Seq 35                     0.77      0.03
```

Fig. 7 Output 2. Percentiles and PDS contribution.

Another sample, a scatter plot, is shown in output 3. The release seems to be due to two different phenomena.

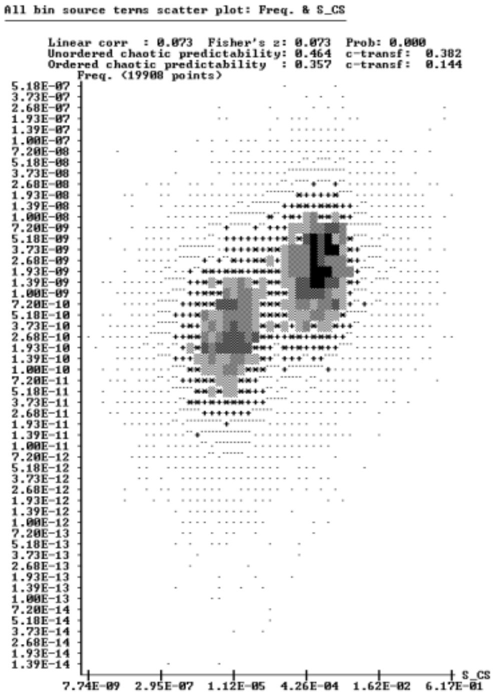

Fig. 8 Output 3. A Scatter plot.

9.2.2.4 Conclusions

Experience gained through the pilot studies and Olkiluoto 1 Level 2 PSA has shown that an integrated Level 1 and 2 PSA model can be implemented with high level of detail. Levels 1 and 2 are tightly integrated, while still preserving freedom for both models to change. Use of programming language allows modelling of dynamic, time-dependent and non-coherent events, which is imperative for severe accident modelling.

9.2.2.5 References

[110] T. Okkonen, *Development of a Parametric Containment Event Tree Model for a Severe BWR Accident*, STUK-YTO-TR 81, Radiation and Nuclear Safety Authority, Helsinki, Finland (1995)

[111] T. Okkonen, *Development of a Parametric Containment Event Tree Model of a Severe PWR Accident*, STUK-YTO-TR 104, Radiation and Nuclear Safety Authority, Helsinki, Finland (1996)

[112] E. Cazzoli, I.Niemelä, "Concepts for a Level 2 Living PSA Approach", *4th TÜV Workshop on Living PSA Application*, Hamburg, May 2-3, 1994, chapter 15 (1994)

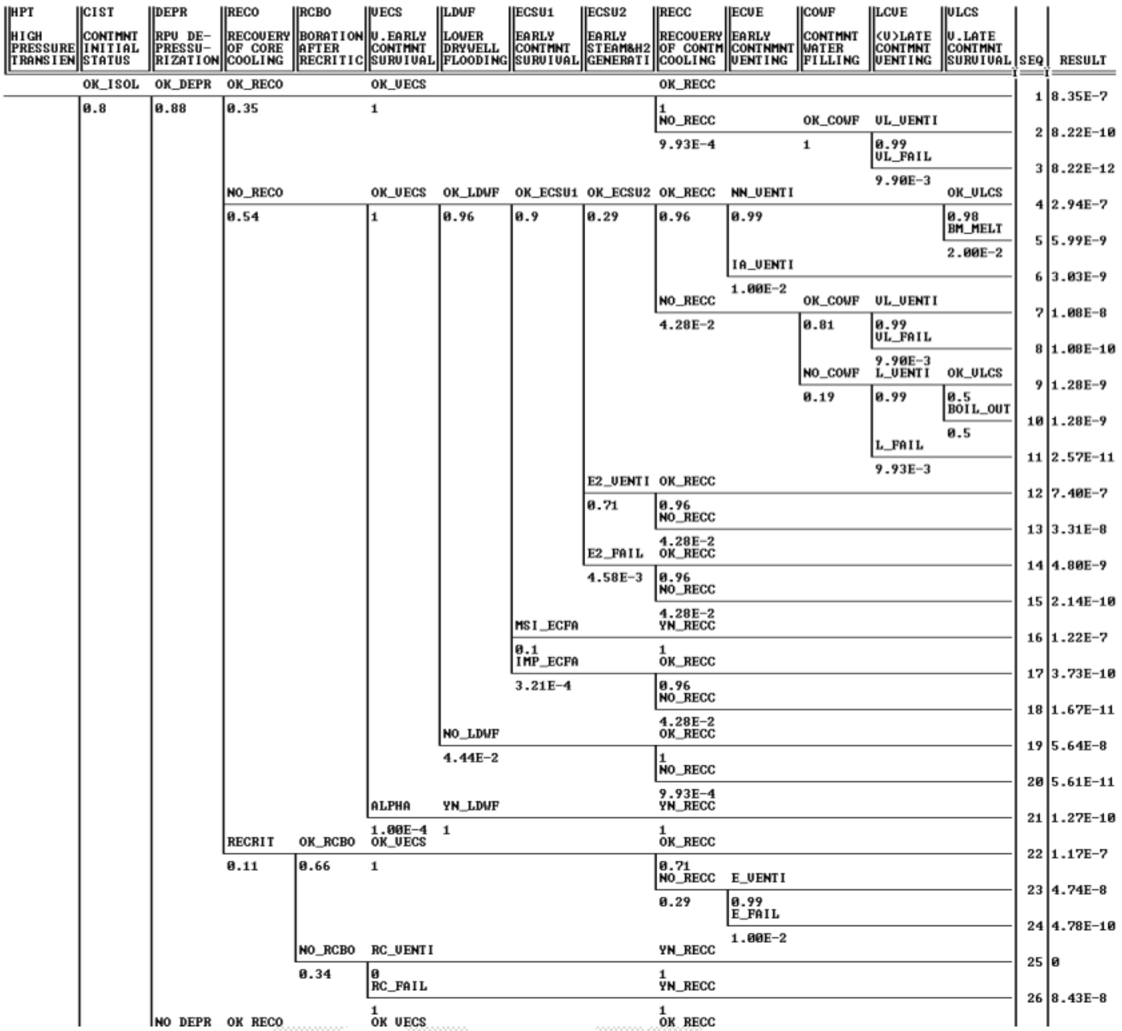

Fig. 9 An extract of a CET for Olkiluoto 1 High Pressure Transient (with mean values)

```
function nil OK_LDWF
  $----------------------------------------------------------
  $ Pedestal flooded before vessel failure
  PrCalc
  Prob    = cumul(Cu322Fl, TiAvStFl)
  Pr1     = Pr1*Prob
  TiFlSt  = icumul(Cu322Fl, Pr1)
  TiFlEnd = TiFlSt + 1300.        $ 780m3 / 0.6 m3/s
  LDWF    = true
return nil
```

Fig. 10 Listing 1. Function for successful lower drywell flooding

```
table (10) P3F = (   0., 1.0e5,
          6626., 2.3e5,    $ Pressure just before RPV failure
          6627., 7.8e5,    $ Max Pressure (MAAP prediction)
          6628., 3.1e5,
         15000., 3.1e5,
         30000., 5.3e5,
         85000., 7.1e5,
         95000., 6.3e5,
        120000., 9.5e5,
        200000., 9.5e5)    $ SPSA end point, equal to prev value
```

Fig. 11 Listing 2. Development of LDW basic pressure in case of reactor overpressurisation

Table 30 Issues related to containment failure in Olkiluoto 1 Level 2 PSA model

Name	Contents
Common section, COMM:1	static capacity of containment
COMM:2	break pressure of 361 and 362 rupture discs
COMM:3	impulse tolerance of lower drywell
COMM:4	displacement of outer wetwell walls
COMM:5	leaktightness of suppression pool
COMM:6	temperatures of weak points for containment
Initial conditions, INCO:5	breach of containment before rupture discs
INCO:6	probability for filtered venting valve being closed
INCO:7	static capacity of containment
Containment initial status, CON_BYPA	steam line or feedwater line unsuccessful isolation
CON_LEAK	containment leaks in the beginning
LEAK_361	leak through direct venting line
HIGH_O2	oxygen more than 3%
CIST:1	steam line or feedwater line unsuccessful isolation
CIST:2	containment leaks into reactor building
CIST:3	containment isolated in the beginning of the accident
CIST:4	containment not inerted
Recovery of core cooling, RECO:6	start time for leak
Very early containment survival, VECS	early containment failure
ALPHA	steam explosion breaks the containment
RC_VENTI	filtered venting rupture disc opens before containment break
RC_FAIL	containment breaks before filtered venting

Name	Contents
	rupture disc
DB_FAIL	containment breaks even if filtered venting rupture disc opens
H2DEF	containment breaks before filtered venting rupture disc - not inerted
H2DET	hydrogen detonation breaks containment
INI_FAIL	containment initially failed
VECS:3	break of containment due to re-criticality
VECS:4	probability of hydrogen detonation
VECS:5	break of containment when not inerted
VECS:6	hydrogen burn breaks containment
VECS:7	probability of steam explosion
VECS:8	in-vessel steam explosion breaks the containment
VECS:9	filtered venting rupture disc opens before containment failure
Early containment survival, MSI_ECFA	core-concrete interaction breaks containment
IMP_ECFA	containment fails due to impulse load
E2_VENTI	filtered venting, no containment failure
E2_FAIL	containment failure due to pressure rise
ECSU:1	steam explosion in lower drywell
ECSU:2	core-structure interaction breaks containment
ECSU:3	non-condensable gases break the containment
ECSU:4	rapid steam development breaks containment
ECSU:5	ex-vessel rapid steam development and NG
ECSU:6	ex-vessel steam explosion
ECSU:7	corium flow to water
ECSU:8	hydrogen development in-vessel
ECSU:9	core-structure interaction
Early containment venting/failure, IA_VENTI	inadvertent filtered venting
E_VENTI	early filtered venting
E_FAIL	early containment failure
ECVE:1, ECVE:2	inadvertent manual depressurisation
ECVE:3	automatic depressurisation
ECVE:4	time point of containment failure
ECVE:5	containment failure before rupture disc
Late and very late venting or cont. failure, L_VENTI	successful filtered venting (WW/DW)
L_FAIL	above fails

Name	Contents
VL_VENTI	successful depressurisation from DW
VL_FAIL	above fails
LCVE:1	time point of depressurisation
LCVE:2	timewindow available for depressurisation
LCVE:3	probability of success for manual depressurisation
LCVE:4	containment fails before rupture disc
Basemat melt through, BM_VLCF	bottom of containment fails
BOIL_OUT	containment dries and fails
VLCS:1	corium not coolable
VLCS:2	probability of containment failure
VLCS:3	uncoolable corium fails containment
VLCS:4	containment dries

Table 31 Containment failure mode distributions in Olkiluoto 1 Level 2 PSA.

Failure mode	% all	% CD	% FCF	% fails in CD	% fails in CD on power
Already open (in refuelling)	60	6	83	22	n/a
Upper drywell	2.3	7		26	33
WW or LDW, flooded	0.6	2.0		7	9.0
Isolation valve failure	0.02	0.01	0.02	0.02	0.03
Overpres. prot. lines do not close	0.08	0.03	0.1	0.1	0.1
Bottom (CCI)	0	0.01		0.05	0.07
Hood	0.07	0.2		0.8	1.0
Filtered venting from DW	1.0	3.0		11	14
Filtered venting from WW	0	0		0	0
WW or LDW, not flooded	3.0	10		34	43
No failure	34	72	17	n/a	n/a
Leak	0.01	0	0.01	0.01	0.02

Table 32 Olkiluoto 1 SOURCE and CLASS variables.

SOURCE term variables	Main source term
S_Xe	release fraction for Xe
S_I	release fraction for I
S_Cs	release fraction for Cs
S_Te	release fraction for Te
S_Sr	release fraction for Sr
S_Ru	release fraction for Ru
S_La	release fraction for La
S_Ce	release fraction for Ce
S_wX	Weighted severity factor based on early effect of nuclides
TiConSt	Start time of release to environment
CLASS variables	**Used in release class binning, supplementary source term**
FLoc	String, Containment failure location, 8 locations
FTim	Real, Containment failure time
VB	Boolean, Vessel breach occurred
LDWF	Boolean, lower drywell flooded
RECC	Boolean, recovery of containment cooling

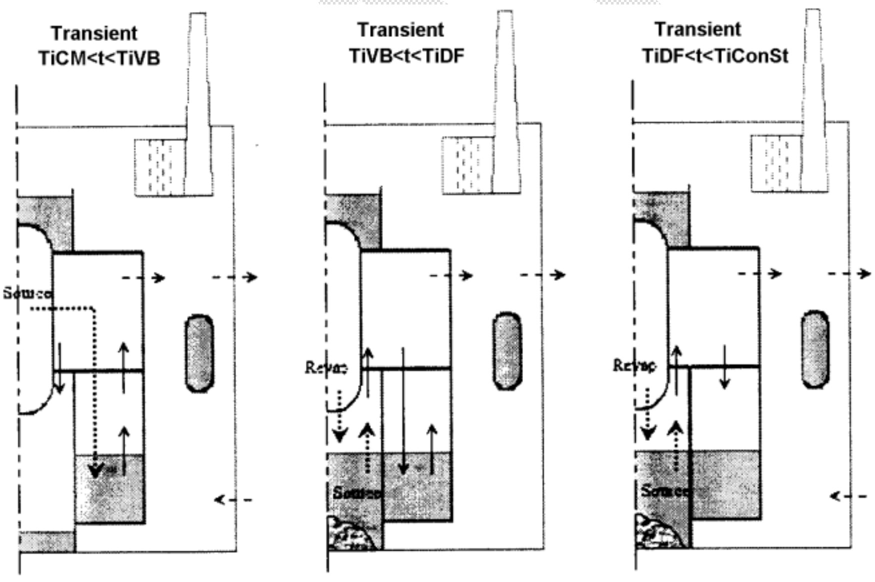

Fig. 12 Gas flow paths in transient initiated sequences before containment failure.

9.2.3 KANT

KANT has been developed by IRSN after examination of existing codes, to fulfil specifics needs of the IRSN Level 2 PSA project. The development started in 1997, and, since this date, the software is continually improved based on users feedback. The first L2PSA developed with KANT was for the French 900 MWe PWR, and KANT is now used for the development of the L2PSA for the French 1300 MWe PWR.

The IRSN L2PSAs integrate a large number of events, associated with different types of function: logical function for system availability or human reliability analysis, and simplified physical models for physical phenomenon. Some physical global variables describing the plant state must be propagated through the successive events. These variables have an impact on the probabilities of events and on the classification of sequences in release categories. The APET is unique for all the PDS. KANT has been designed to deal with all these specificities. Moreover, KANT has been designed as a user-friendly tool to facilitate the treatment of the results. Considering all these objectives, KANT particularly allows the user to:

- represent the L1-L2 interface;
- represent the APET;
- quantify the frequency of accidental sequences, and perform grouping into release categories;
- calculate the release level for each accidental sequence;
- perform uncertainty analysis by Monte-Carlo simulation;
- perform the post-treatments of the results.

A more precise description is provided in the following chapters.

9.2.3.1 Description of KANT

KANT has been programmed in C++ language, and uses an ACCESS database to stock results and data. KANT is currently a set of 4 different modules:

- the "study development module" enables the creation or modification of an APET, and all the elements related to the APET;
- the "quantification" module enables performance of quantification of a study (parameters definition, APET choice, type of calculation choice, cut-off frequency definition,…);
- the "post-treatment" module enables visualisation and utilisation of results;
- the "administrator" module enables management of the KANT users rights.

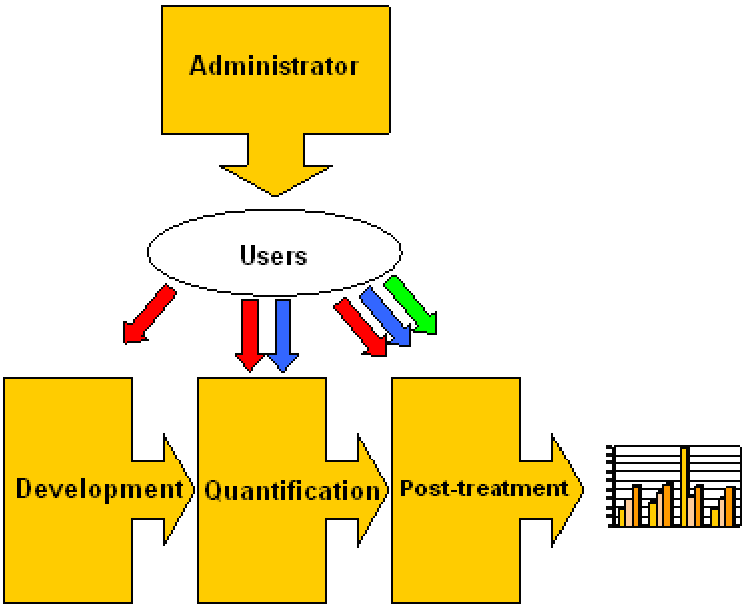

Fig. 13 KANT architecture

9.2.3.2 Study development module

This KANT module is used for the creation and modification of all the elements composing an L2PSA event tree: PDS, events, release categories, functions, etc. The module integrates a programming coherence verification tool that notably checks the existence of parameters and variables used in the events and functions.

The APET starts from the PDS. The PDS are defined from the L1PSA results, and are defined by the combination of interface variables values (that describe the state of the plant at the beginning of core melt). A frequency is attached to each PDS. KANT allows performance of some operations on the PDS (grouping, selection).

The event-tree is represented as a list of nodes called "questions" or "events", linked by branches that represent the different answers to questions. Events can be grouped for better presentation.

Each event is associated with a model that enables the calculation of plant state modification just after the event and the probabilities of the different paths if any (depending on the values of the local variables describing the plant state, some global parameters of the study or random parameters). Some external function can be also used for the calculation of physical phenomenon.

A specific language has been developed for the coding of the event models. It integrates logical and arithmetical operators, and enables manipulation of tables and scalars.

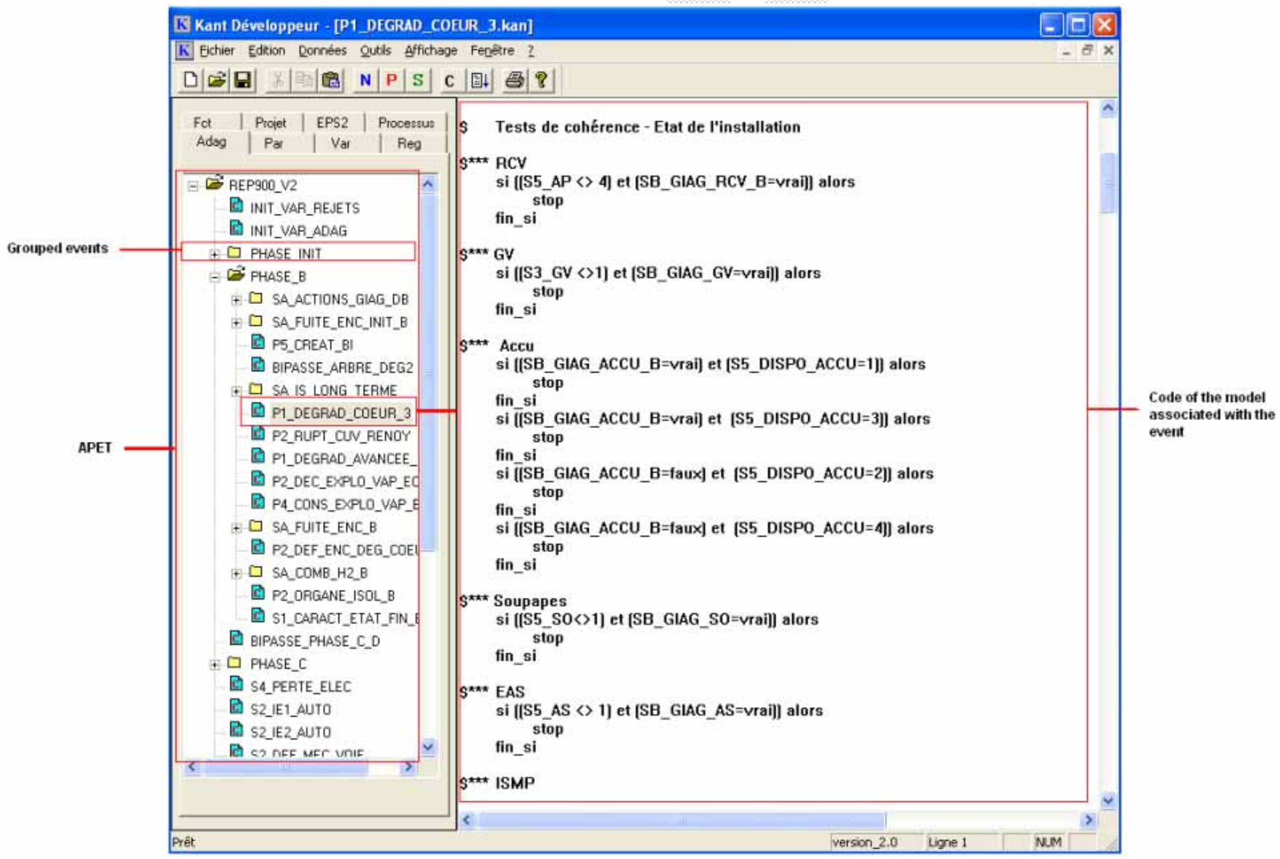

Fig. 14 View of the « development » module

The release categories (RCs) are not pre-defined by the user who defined a set of variables used for the grouping of sequences into release categories (variables that enables to define the level of release of an accidental sequence), and the RCs are dynamically built by KANT during the APET quantification.

9.2.3.3 Study quantification module

This module enables the probabilistic quantification of a Level 2 PSA study. A study is defined by:

- the APET selected;
- the definition of parameters values: the parameters are data used for the quantification of the study (for example: type of concrete composing the basemat, probability of event, etc). These numeric parameters can reflect epistemic uncertainties; in that case they are distributions (8 kinds of probabilistic laws are used by KANT). The value of the parameters can be changed to perform sensibility analysis;
- the type of calculation:
  - point values analysis: the quantification is done with no respect to uncertainties, using the fractile 50% value of the uncertain parameters;
  - uncertainty analysis: the uncertainty analysis is performed by Monte Carlo simulation;
- the calculation (or not) of the release associated with each release categories: this calculation is performed by an external function linked with KANT. It uses the values of the variables defining release categories and gives characteristic values of the release (Becquerel by radionuclide species). This function is plant dependent and has to be provided by the user;
- the cutoff frequency.

The following figure (Fig. 15) explains how the quantification is performed for an uncertainty analysis:

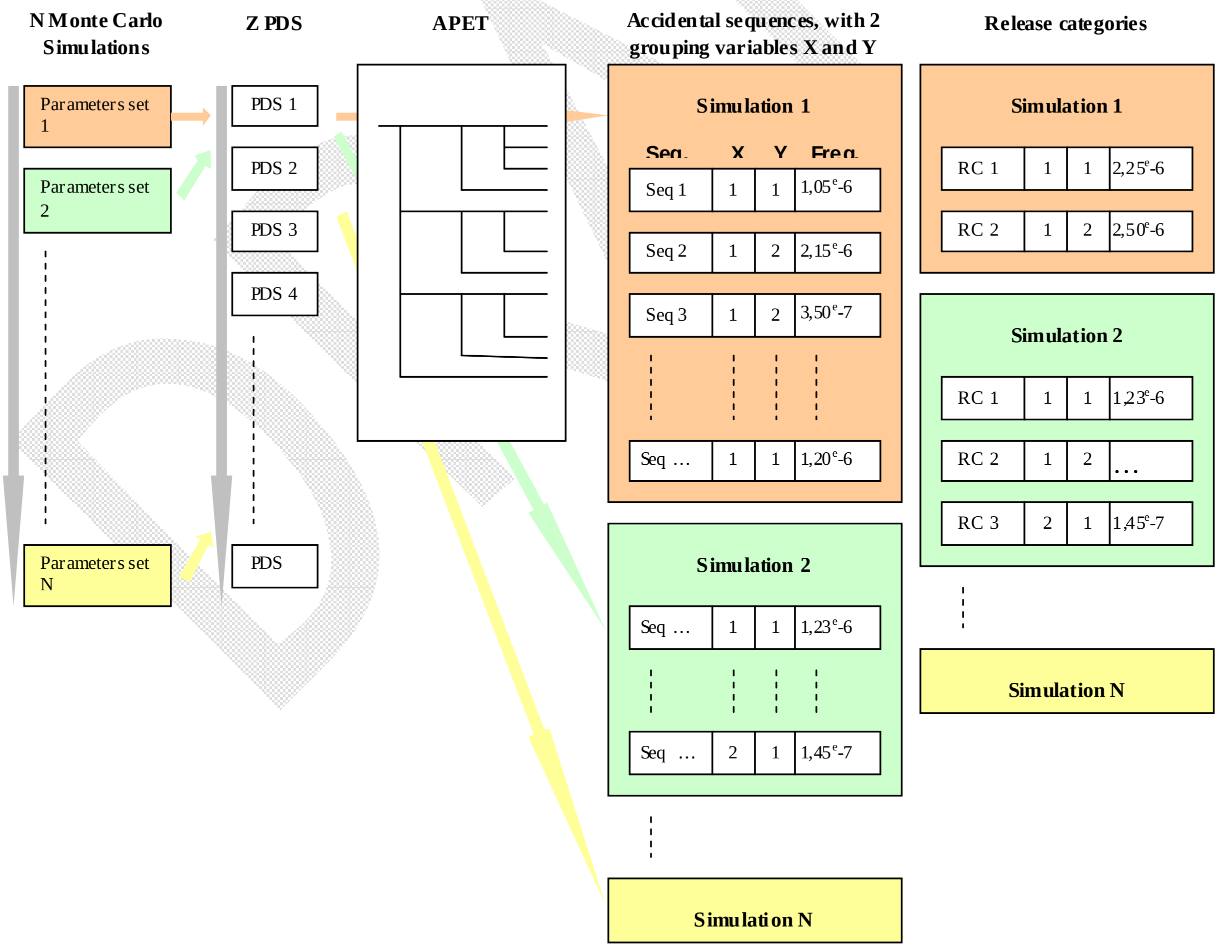

Fig. 15 Uncertainties analysis quantification

The accidental sequences generated by the APET are grouped in RCs for each Monte Carlo simulation, so a RC is not associated with on frequency but with a distribution of frequencies, as shown on the following figure (Fig. 16):

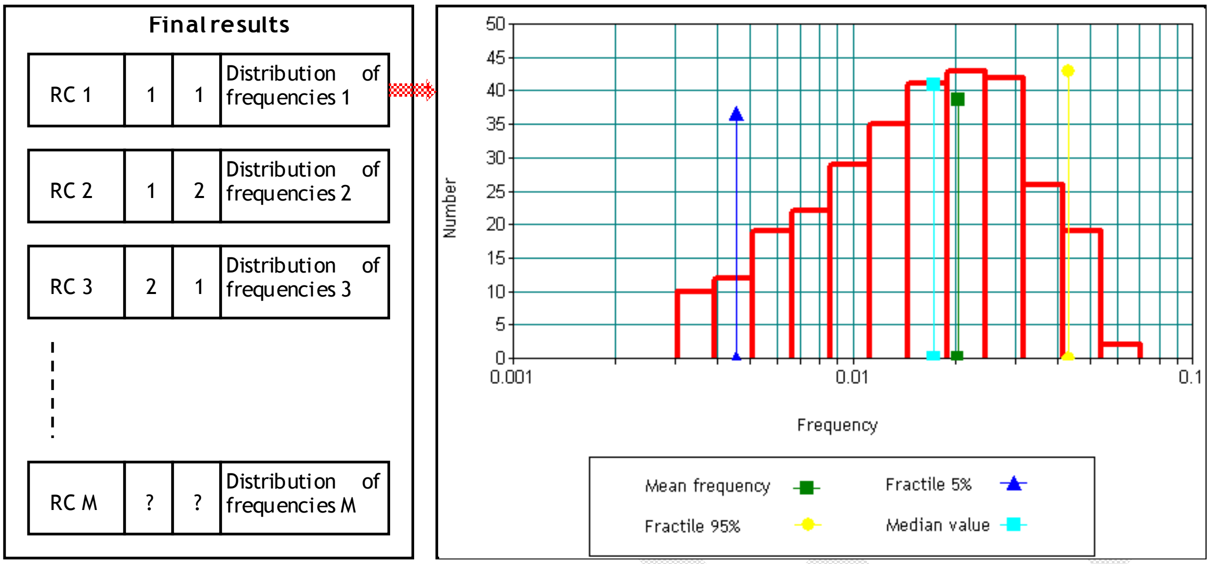

Fig. 16 Presentation of RC's distribution of frequencies in KANT

9.2.3.4 Results post-treatment module

This module is dedicated to exploitation of results. It enables viewing of the results as in Fig. 16, and to group release categories in more global groups. This functionality enables definition of more global release categories (for example based on containment failure modes, or delay before containment failure …) from the more detailed release categories definition.

For example the IRSN study on French 900 MWe PWR generates more than 5000 RCs after the quantification (based on more than 30 variables). These RCs are defined to contain all the information needed for the source terms assessment (cf. Volume 2, section 7.4.4.2). But for the global presentation of results (synthesis report), these 5000 RCs are grouped in 25 global RCs based on the containment failure modes.

9.2.3.5 Further improvements

KANT has been developed as a complete and flexible software for L2PSA development and application. For the development of the APET, it enables user-friendly GUI and the use of external function developed in C++ for the physical phenomenon modelling. For the quantification, it enables uncertainties analysis, and easy sensitivity analysis. Finally, KANT contains a results post-treatment and visualisation tool.

The KANT software will be maintained by IRSN to support the long term L2PSA activities (IRSN updates the L2PSA models in relation with plant modifications). Next developments will be linked to L2PSA application for risk monitoring and will concern mainly the coupling between L1 and L2PSA models.

9.2.4 Risk Spectrum PSA for Level 2 PSA

9.2.4.1 History

Risk Spectrum is a PSA tool developed by Scandpower.

The first RiskSpectrum version was limited to fault tree analysis and was introduced in 1989. The first complete RiskSpectrum PSA tool with integrated fault trees and event trees was introduced in 1991.

The code has since undergone several large development phases. The first windows version named RiskSpectrum PSA Professional was released 1998 and a major upgrade with completely rewritten code, RiskSpectrum PSA, was released by the end of 2007.

RiskSpectrum is a platform consisting of several tools facilitating PSA work. The tools available are:

- RiskSpectrum PSA – a fault tree and event tree tool.
- RiskSpectrum Doc – a documentation tool.
- RiskSpectrum RiskWatcher – a risk monitor application.
- RiskSpectrum FMEA – a tool for identification and creation of basic events and related data parameters providing a link between a failure mode effect analysis and the RiskSpectrum PSA database.
- RiskSpectrum HRA – a tool for human reliability analysis.
- R-DAT – a tool for Bayesian update of reliability data.

RiskSpectrum PSA is built on a well structured relational data base. All records are only stored once. This means that all data only need to be edited once, and that these events can be reused in different places in the PSA model without need for entering similar data again.

The user interface is designed to support efficient modelling by providing both graphical fault tree and event tree editing and tabular record editing capabilities.

RiskSpectrum PSA is licensed to more than 1600 users in some 450 organisations. RiskSpectrum is used both for developing and maintaining baseline Level 1 and Level 2 models and for various PSA applications such as evaluation of design changes and risk follow-up activities. 50% of the nuclear power plants in the world use RiskSpectrum in their daily PSA work.

9.2.4.2 Integrated level 1 and level 2 models

A Level 2 PSA analysis includes development of the model that describes the scenarios from initiating events through core damage states to release categories. The Level 1 core damage states are split into plant damage states (PDS) that are the initiating events for the further accident progression with modelling of phenomena and other threats to the containment and also consequence mitigating functions. The end states in the Level 2 part of the model are the different release categories.

The integrated model approach has the PDSs as the link between the Level 1 and Level 2 parts of the PSA model. The transfer from Level 1 PSA to Level 2 is explicitly taken care of by the PDSs. This means that all failures occurring in Level 1 PSA can be fully taken into consideration when analysing a specific release category (if equipment has failed in one cutset in the Level 1 PSA, it will also be treated as failed in the Level 2 PSA part when that cutset is evaluated). The whole model (without intermediate cutoff) is considered when the minimal cutsets are identified and the frequency is calculated for a specific release category.

The integrated approach means that the amount of explicitly documented PDSs can be decreased significantly compared to a non-integrated approach.

As has been stated in the section on EVNTRE, it is usual in Level 2 PSA to consider in the order of 100 branching points in the event tree, some of them with more than two branches, and many of them quantified by user defined functions representing physical phenomena. Such a multitude is impossible to model in RiskSpectrum, therefore a condensation of the number of branches into a manageable number is needed. To retain the respective background information and correlations of the analysis, it is necessary to model a large set of information and dependencies within single branching points in RiskSpectrum. Since RiskSpectrum is designed to deal with system dependencies, but not with interrelated physical phenomena evolving along a time scale, it is not a trivial task to create a Level 2 PSA model which would be equivalent to the features of a large event tree. Therefore, the advantage of having an integrated Level 1 - Level 2 model is partly offset by the loss of some favourable features of level 2-specific event tree codes.

9.2.4.3 Use of RiskSpectrum for Level 2 PSA analyses

RiskSpectrum has a feature for linking event trees using the end state in a sequence as an "initiating event" to another event tree. This linking is applied many times when a Level 1 event tree model is split on several event tree pages.

This linking feature is also useful when performing a Level 2 PSA when developing the interface between the Level 1 and Level 2 parts of the PSA. The plant damage states (PDS) are set as the end states in the level 1 event trees and then used as initiators in the level 2 accident progression event trees representing the accident scenario after core damage and until release (sometimes called containment event trees). This linking provides an explicit logic connection between level 1 and level 2 event trees.

The events in the CETs are defined so that they represent the important factors for the accident progression and support the definition and assignment of release categories to the CET sequences. The events are represented by basic events (direct definition of the branch probability) or by fault trees. RiskSpectrum also allows more than two branches in one branch point.

The events in the CET are normally a mix of probabilities for phenomena and system functions. These system functions can share dependencies with the system functions in Level 1 PSA, for example regarding power supply, cooling etc. These dependencies are taken into full consideration by RiskSpectrum during the quantification. Fig. 17 gives an overview of dependencies between event tree functions and between event trees.

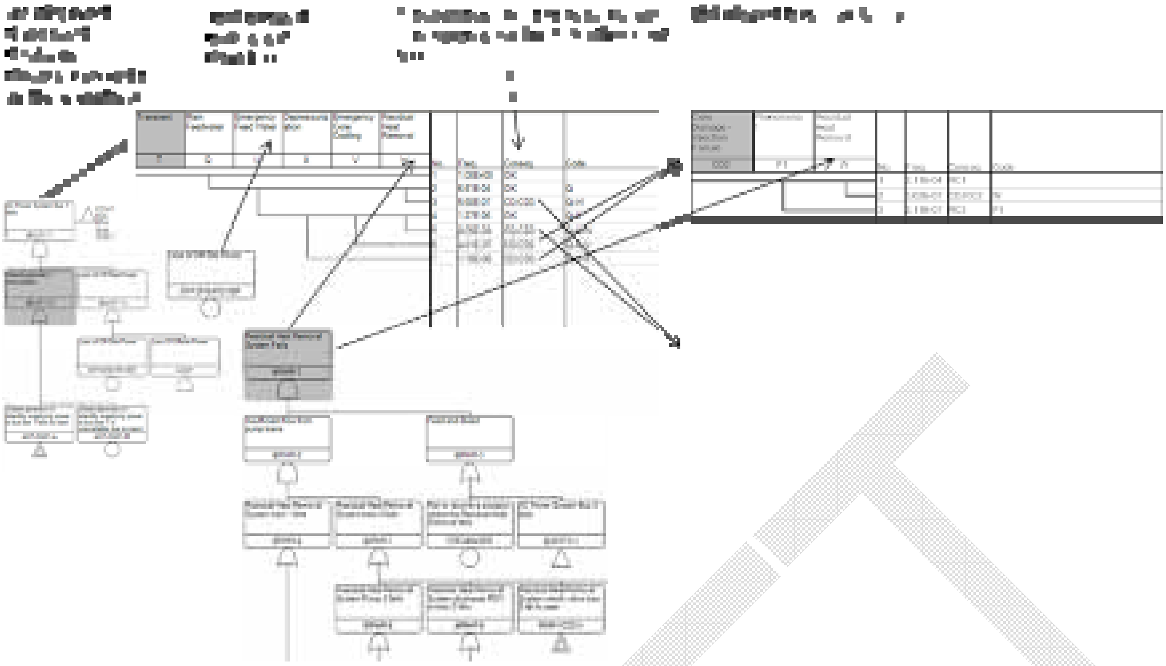

Fig. 17 Dependencies between event tree functions and between event trees.

RiskSpectrum considers success events from a logical point of view, i.e. a success event in the same cutset as the failure of the event will result in the removal of that cutset.

Usually, the failure (event) probabilities are low, and hence, the success branches have a probability close to 1.0. However, RiskSpectrum offers several ways of treating events with high probabilities:

- Use of simple quantification of success branch
- Definition of the success branch with a specific basic event

RiskSpectrum also has specific functions to treat mutual exclusivity between basic events, the MUX editor, (if that would be necessary) and to specify that the probability of one event is a function of another, the BE-BE editor. These two functions may be relevant in some specific situations. The functions are not limited to Level 2 PSA, but may be of specific interest due to the presence of large probabilities.

The functions are depicted by the figure below (Fig. 18).

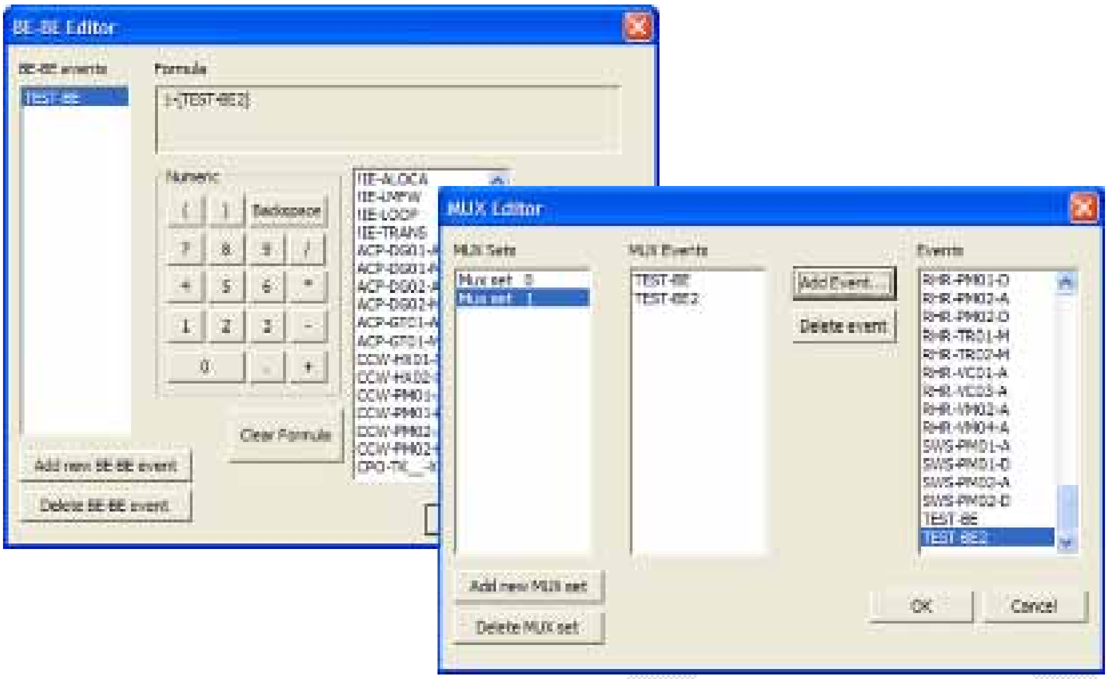

Fig. 18 Example of the functions basic event to basic event relation and mutual exclusivity treatment between events

Inheriting boundary condition sets

A boundary condition set in RiskSpectrum is a set of rules (house event, basic events or gates set to TRUE or FALSE). By applying a boundary condition set a given fault tree structure is automatically modified to represent a different situation. This means that the same fault tree structure can be used for several situations, though it should be modified to consider special success characteristics, e.g. number of pump trains needed and operating time of pumps. All modifications are built into the original model together with the applicable set of house events (boundary conditions). The use of this approach facilitates model maintenance and understanding of the model and the review process may be significantly more efficient.

A function in RiskSpectrum of great importance for creating a compact Level 2 PSA that is easy to review is called *"inherit boundary condition sets between event trees"*. The function means that a boundary condition specified in one event tree is inherited through all sequences starting from the actual location (also including child event trees).

The *inherit* functionality means that the CETs may be defined so that there is only one CET per PDS – and the differences for the different function events (top events) in the CETs are changed via the use of BC-sets defined already in Level 1 PSA (the father event tree).

It is also possible to design Level 2 PSA without the use of an *inherit* functionality, but that typically increases the complexity in the modelling.

9.2.4.4 Results

The results generated by RiskSpectrum are based on minimal cutset lists (MCS). The purpose of generating MCS is

- to be able to verify that the results generated are reasonable;
- to quantify the top frequency/probability.

Based on the MCS list there are several results that may be quantified, to support decision making. These results are typically:

- Importance analyses.
- Uncertainty analyses.
- Sensitivity analyses.

An example of an importance analysis is the contribution from a specific phenomenon to the top result. The integrated approach between Levels 1 and 2 makes it possible to correctly and directly evaluate the importance of equipment that are used both in Level 1 and 2 PSA. Thereby it is possible to do an integrated risk evaluation and to discuss which equipment is most important for the plant (from initiating event to release).

RiskSpectrum also offers the possibility to perform MCS post processing. MCS post processing is cutset manipulation based on rules. Cutset manipulation may be used for example for dependency treatment for events representing operator action errors.

RiskSpectrum is very flexible with regard to the definition of results to be calculated (Analysis cases). Normally the following are calculated in a Level 2 PSA project:

- Minimal cutsets and frequency for plant damage state per initiating event.
- Minimal cutsets and frequency for release category per initiating event.
- Minimal cutsets and frequency per release category.

The results are typically used to generate uncertainty distributions for each release category, as exemplified by the figure below (Fig. 19).

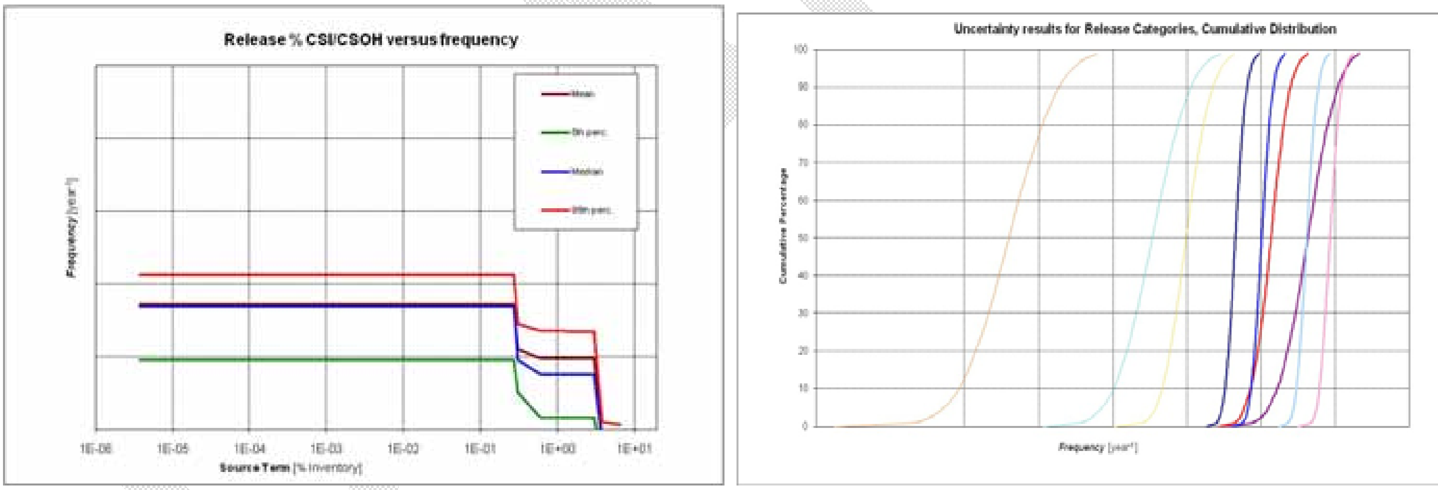

Fig. 19 Two examples of normal results from Level 2 PSA using RiskSpectrum

A function that may be helpful in RiskSpectrum is the *consequence matrix* function that provides a tool for arithmetic operation based on the results generated by the probabilistic quantifications.

9.2.4.5 Conclusions

RiskSpectrum is a tool for integrated analysis of L1PSA and L2PSA. It is an excellent and user friendly tool used by many organisations.

The functions within the tool make it possible to fully take into consideration dependencies between the Level 1 and 2 parts of the PSA model. However, the advantage of having an integrated Level 1 - Level 2 model is partly offset by the loss of some favourable features of Level 2-specific event tree codes.

9.3 SOME VIEWS ON INTEGRATION OF SOURCE TERM CALCULATIONS IN THE APET

The purpose of risk integration is to condense the vast amount of information in Level 2 PSA and to prepare the results for interpretation and presentation. This includes collecting the results of individual APETS to form the results of release categories and the overall results. Depending on the contents and structure of the PSA, this task can be performed in different ways.

One of the most important presentations of the risk is the "complementary cumulative distribution", the preparation of which is discussed below.

A Level 2 PSA includes a huge amount of information, much of which may be in the form of distributions. Complete risk integration is the process of combining results of frequency calculations and source term calculations. The amount of information may be tremendous, especially if the Level 2 PSA contains uncertainty analysis.

Risk integration is a demanding process not only due to amount of data involved. If not done correctly, risk integration may distort results, miss correlations between events, and give erroneous interpretations from otherwise correctly executed Level 2 PSA.

The following paragraphs try to provide information on both detailed and simplified approaches.

9.3.1 If APET includes source term calculation…

In the following section, the process of generating integrated results from individual APETs and Release Categories is described from the point of view of uncertainty analysis. The starting point is uncertainty analysis, in which frequency and release are calculated for each APET sequence in each Monte Carlo run - i.e. situation where there is no binning and no condensation of the results. It is examined under which conditions the information can credibly be binned and condensed. After this detailed approach, some evaluation of simplified approach is presented.

In a Level 2 PSA, different phenomena are modelled. Some phenomena affect only the frequency of a release category, some affect only the source term, and some affect both. It is often assumed that a source term can be divided into two independent dimensions: frequency and release fraction. Other dimensions may also be included, like timing, energy and height of release. In practice, frequencies are often calculated using one model (APET) and release fractions are calculated independently for representative sequences. This approach contains an assumption that frequency and release fraction are not correlated. An interesting question is: how independent are the frequency and source term of an accident sequence?

The results of a Level 2 PSA can be described as a scatter plot, where the points are placed on coordinates of release fraction and frequency, such as in Fig. 20, where source term calculation and frequency calculation has been performed for each end point of APET in each simulation run. Since the APET is based on mutually exclusive conditional probabilities, the end points of the APET are mutually exclusive.

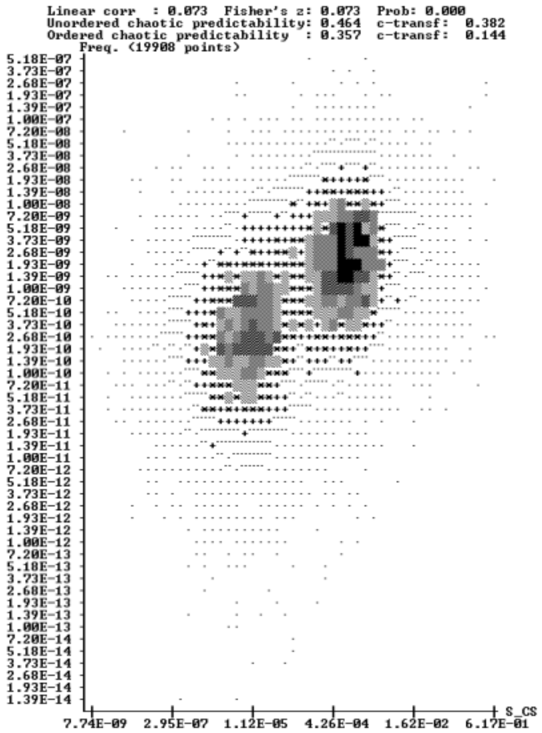

Fig. 20 Scatter plot of Cs release fraction vs. frequency for end points of release category L_VENT_U in one APET

Fig. 20 shows a Cs scatter plot of Monte Carlo uncertainty analysis for release category L_VENT_U in one APET in Olkiluoto 1 Level 2 PSA. There are 19908 source term samples, which are presented on coordinates Frequency and Cs release fraction. These source terms come from 4 different sequences. In this figure, no loss of information occurs, since each [release fraction, source term] pair is individually computed and plotted on the graph. Each point describes one physically credible scenario. For each simulation run, the scatter plot describes a set of 4 alternatives, of which only one can occur, since the end points of an event tree are mutually exclusive. Thus, for each simulation run, the frequencies of sequences are additive, and one can make complementary cumulative distribution, as shown in Fig. 21.

One simulation run produces 4 source terms:

end point	Cs	Freq
#7	6.60E-04	1.00E-08
#9	1.10E-05	1.30E-09
#33	9.00E-04	2.00E-09
#35	1.50E-05	2.40E-10

Cumulative graph is formed by sorting according to release and summing frequencies:

Cs	Freq	Cumul Fr
9.00E-04	2.00E-09	2.00E-09
6.60E-04	1.00E-08	1.20E-08
1.50E-05	2.40E-10	1.22E-08
1.10E-05	1.30E-09	1.35E-08

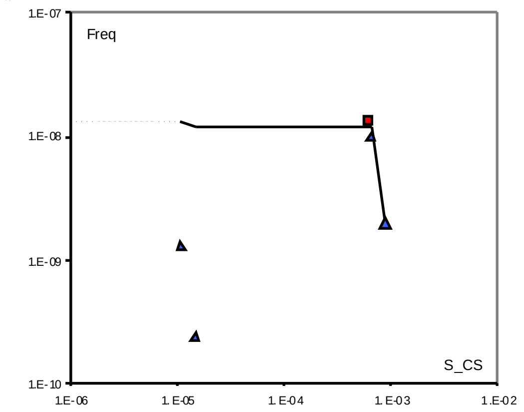

Fig. 21 One Monte Carlo sample of complementary cumulative distribution for release category L_VENT_U made from four points in one simulation run, including weighted sum (□)

As Fig. 21 shows, in further analysis of the scatter plot it is necessary to know which points belong to the same simulation run, since each set of 4 points produces one complementary cumulative distribution. Thus, Fig. 20 contains 19908 source term samples, which form 4977 samples of complementary cumulative distribution, one sample of which is shown in Fig. 21 .

It is also possible to combine frequencies and releases within one simulation run, as represented by the weighted sum in Fig. 21. This can be done by adding the frequencies of the points and by computing the weighted average of the release fractions. This will produce a scatter plot, which is shown in Fig. 22.

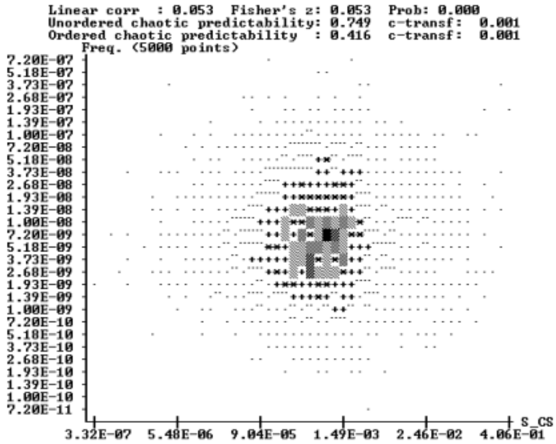

Fig. 22 Scatter plot of Cs release fraction vs. frequency for release category L_VENT_U in one APET

Fig. 22 represents a scatter plot of one release category. From Fig.22, one can create uncertainty distributions for frequency and release fraction by projecting the scatter plot on frequency and release fraction axes. This projection is not reversible, i.e. the scatter plot can not be constructed from the distributions of release fraction and frequency. When looking at the projections of release fraction and frequency, it is not possible to deduce whether the two variables are correlated or not. On the contrary, starting from the projections, distributions can be constructed whose correlations range from -1 to 0 to 1, as shown in Fig. 23.

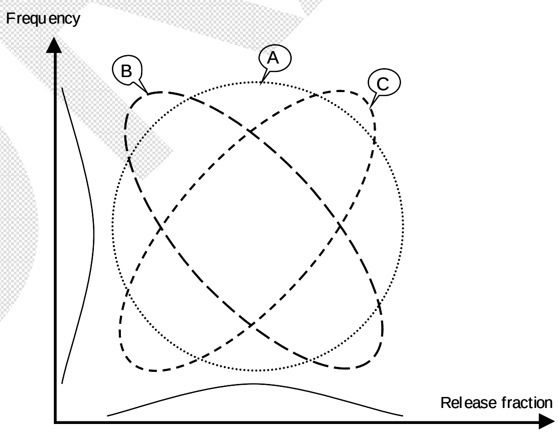

Fig. 23 Projections on frequency and release fraction axes of distributions A, B and C are similar

The shapes of distributions in Fig. 23 can be summarised as follows:

A. Release fraction and frequency are not correlated. The probability of large release is the same as the probability of small release.

B. The probability of release decreases as the magnitude of release increases. This might be a desirable state concerning risk.

C. The probability of release increases as the magnitude of release increases. This is not a desirable state concerning risk.

Alternatives A, B and C have very different risk profiles. The complementary cumulative distributions for A, B and C are sketched in Fig. 24.

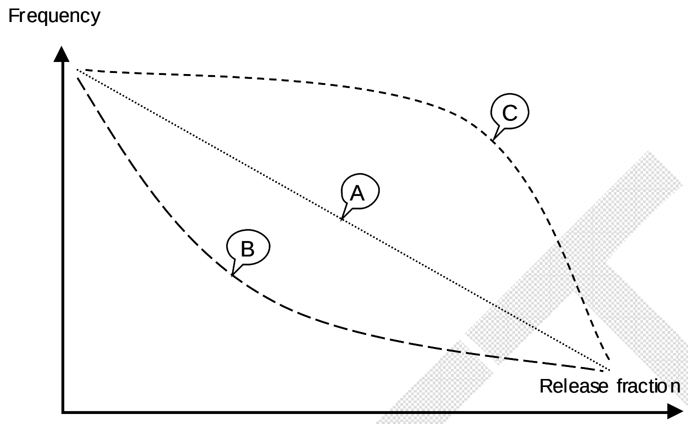

Fig. 24 Sketches of complementary cumulative distribution for cases A, B and C

In reality, release fraction and frequency are not independent variables. There are several phenomena that affect both the probability of release and magnitude of release. For example, high hydrogen concentration may increase both the probability of containment failure and size of the containment failure. Timing of operator action may affect both the probability of high pressure vessel failure and amount of release. Time of containment spray failure may affect the probability of containment failure and the amount of release.

To preserve these correlations in the final results, projections must not be made, which means that [frequency, source term] pair must not be separated. The separation can be caused, for example, by simple summing. Each point in Fig. 24 contains a complementary cumulative distribution, as shown in Fig. 21 and Fig. 25

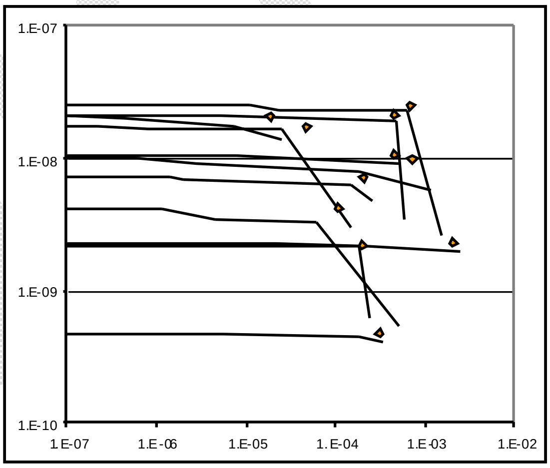

Fig. 25 Each point of Fig. 22 actually a complementary cumulative distribution (11 points shown)

Each release category consisting of more than one accident sequence contains complementary cumulative distributions in itself. This means that the uncertainty of such a release category is more complicated than hinted by the scatter plot in Fig. 22. Fig 26 displays the complementary cumulative function with uncertainty limits, as calculated from the simulated 4977 distributions.

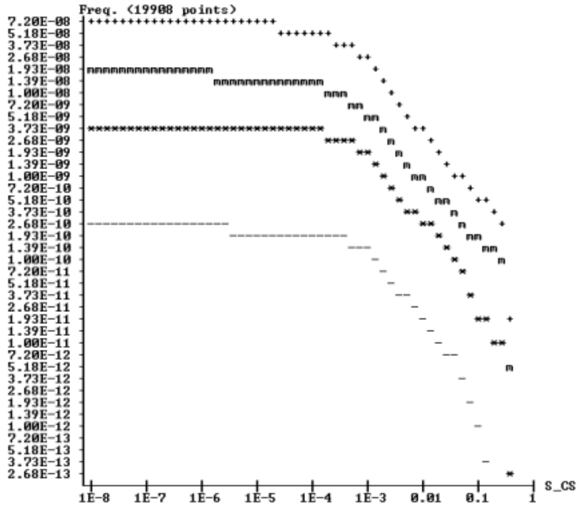

Fig. 26 Complementary cumulative distribution

The uncertainties of frequency and Cs release fraction, as derived from the total result in Fig. 22 are shown in Fig. 27.

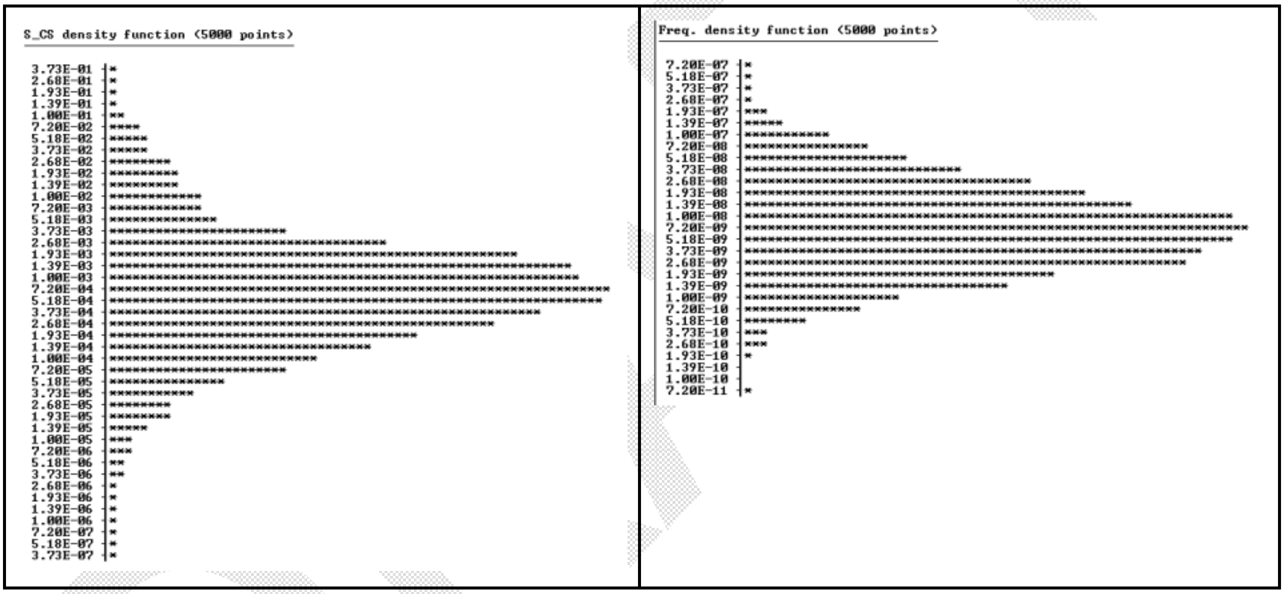

Fig. 27 Uncertainties of release fraction and frequency as derived from the total result of release category L_VENT_U

When Fig. 26 and Fig 27. are compared, it is evident that treating the results of a release category consisting of multiple accident sequences as points of [release fraction, frequency] instead of complementary cumulative functions flattens the uncertainty in risk. In the example, each point in Fig. 22 is a graph consisting of 4 points in Fig.20. The actual deviation is hidden and the correlation is gone. In addition, since each point presents a mean value, the plot may contain unphysical points. Thus, addition even within one Monte Carlo run may distort the results and the shape of the complementary cumulative distribution, i.e. the risk profile. This occurs due to the complementary nature of containment event tree branch points: when the probability of on branch decreases, the probability of other branches must increase to preserve the sum as exactly 1. This tends to stretch the distributions, since the movement of one point towards high value moves another point towards low value. The difference between Fig. 20 and Fig 22 is clear.

In practice, the simplest way to avoid pitfalls and to preserve all correlations is to make no intermediate combinations of the results, but to keep all points to the final level and perform all analysis there. Thus all [frequency, source term] pairs and individual complementary cumulative distributions are preserved to the final level of risk integration. Then statistical analyses are performed with (subsets of) the original data, and there is no possibility for loss of information. However, the amount of data may become quite large.

The sum values are useful, for example, in calculation of importance of accident sequences belonging in a release category. For example, in the analysis of data in Fig. 22 the following table of importance is generated:

Table 33 Example of table of importance

sequence	Raw%	Weighted%
7	49.03	82.26
9	0.76	0.16
33	49.44	17.54
35	0.77	0.03

The table means that when looking only at the Cs release, sequences 7 and 33 are responsible for nearly whole release. When the release is weighted by frequency (=risk), it can be seen that 82% of the risk of Cs release comes from sequence 7.

Olkiluoto 1 example

When performing 5000 simulations of Monte Carlo uncertainty analysis for Olkiluoto 1 Level 2 PSA, 1755000 source term samples are generated. Each source term sample contains the following variables:

1. Release category,
2. Frequency,
3. Containment inerted,
4. Vessel failure,
5. Lower drywell flooded,
6. Containment failure location,
7. Containment failure time,
8. Time of start of core melt,
9. Time of vessel failure,
10. Time of start of release to environment,
11. Release fractions for 9 species.

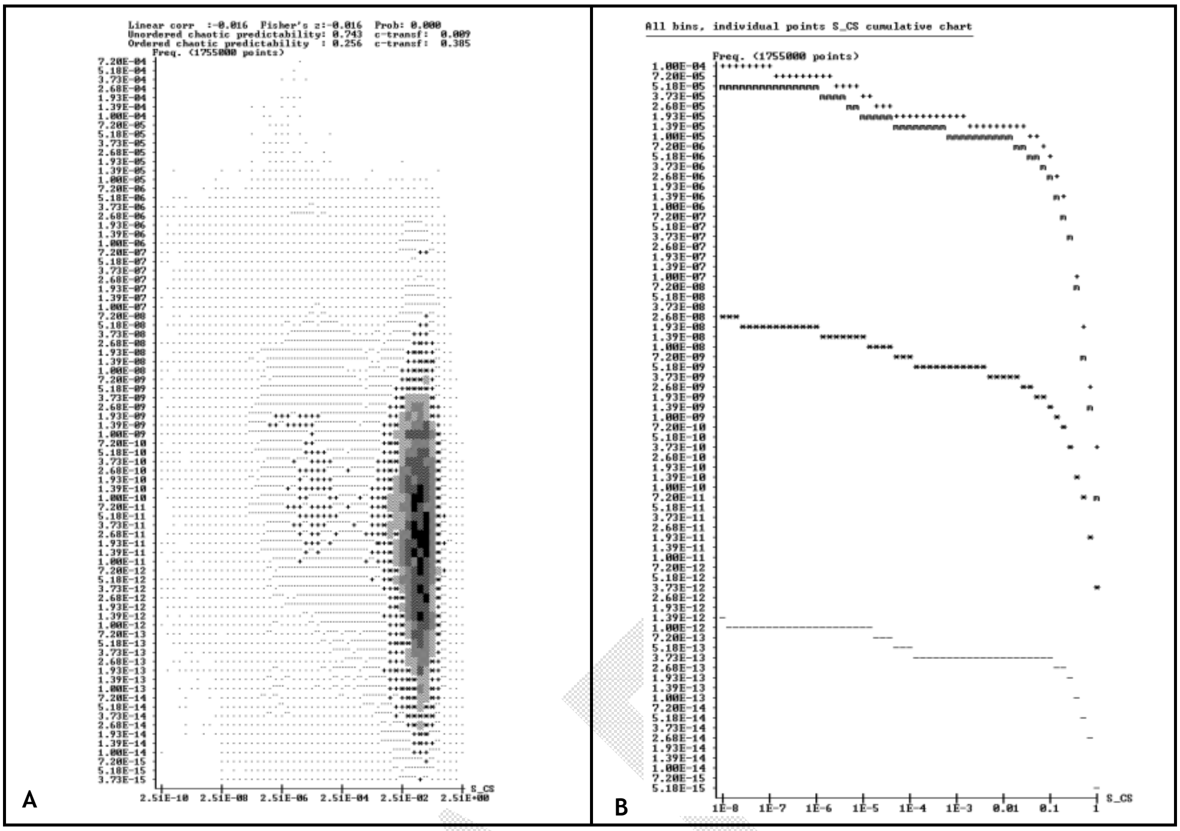

Fig. 28 A) Caesium release fraction - frequency pairs of Olkiluoto 1 Level 2 PSA - B) Complementary Cumulative release of Caesium +=95^{th} percentile, m=mean, *=median, -=5^{th} percentile

Thus, a huge amount of data is generated, preserved and analysed by the risk integrator. Fig. 28 shows the total release of Cs over all release categories.

Fig 28 also shows some high frequencies, which are due to Fuel Cladding Failure, where a small amount of the fuel is affected. This is a specific plant damage state.

As can be seen, the variation in frequency is large, and the complementary cumulative distribution shows large variation as well. In Fig 28 B, the 95^{th} percentile graph is a decreasing function. This is because it represents the complementary cumulative distribution of 95^{th} percentiles, which is decreasing by definition. It is different from the 95^{th} percentiles of complementary cumulative distribution, which usually are not decreasing and may fluctuate up and down.

If the results of the Cs release are summed over the whole PSA model for each simulation run, Fig. 29 appears.

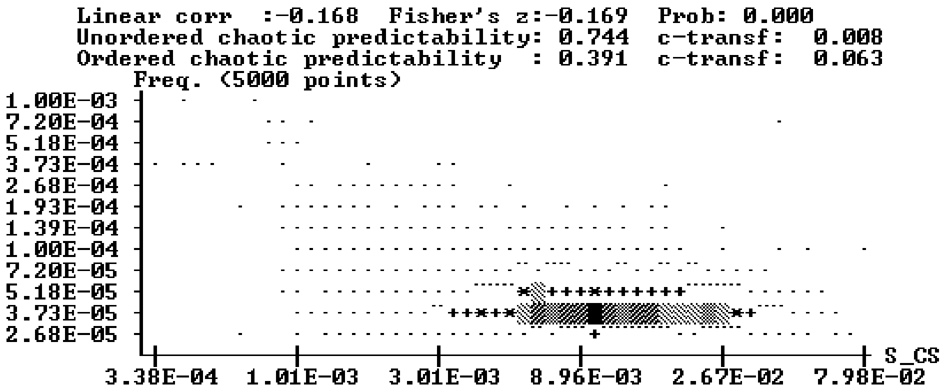

Fig. 29 Cs release fraction - frequency pairs over the whole Level 2 PSA for Olkiluoto 1 when summed in each Monte Carlo run

Each point in Fig. 29 is a mean value of complementary cumulative function consisting of approximately 350 points. Fig. 29 shows very small variation in frequency, which can be explained by one property of APET: Sum of conditional probabilities over each branch point equals 1. Thus, when the frequency is summed over all release categories, one ends up with the uncertainty distribution of Level 1. This is an effective test of consistency for the uncertainty analysis of a Level 2 PSA. If this condition is not fulfilled, the uncertainty analysis is not statistically correct.

The data in Fig. 29, summed over the whole Level 2 PSA, is again useful for calculating important contributors. Table 34 displays the fractional contributions of each APET in Cs release.

Table 34 Fractional contribution of each APET in Cs release

APET	Raw%	Weighted%
01CBP-12	3.16	0.28
02RCO-12	4.12	1.05
03ROP-12	13.27	0.37
04COP-12	4.97	0
05HPL-12	10.9	0.08
06HPT-12	15.61	11
07LPL-12	10.9	1.41
08LPT-12	14.04	80.51
09RHL-12	8.45	0.36
10RHT-12	12.62	4.94
11VLL-12	1.95	0

From Table 34 one see that the largest contribution to risk of Cs release comes from APET 08LPT-12. Then one can check the corresponding table of that APET and find the accident sequences leading to largest risk of Cs release.

Due to the complementary nature of an APET, when the frequency of one branch goes down, the frequency of other branches goes up. The sum thus exhibits less variation than individual terms. Even when the sum and its weighted release are calculated correctly, addition hides the complex conditional behaviour of the APET, caused by requirement *sum of conditional probabilities = 1*. It is essential that the variation in end points of APETs is preserved, since it is the individual end points that form the collection of possible releases - not their sum, even if they belong to the same release category.

9.3.2 IF APET does not include Source term calculation …

In a Level 2 PSA, source term is divided into two dimensions, which are assumed independent of each other: frequency and release fraction. Other dimensions may also be included, like timing, energy and height of release. Frequency is calculated using one model (APET) and source term is calculated separately for representative accident sequences. This approach produces usually ten to twenty different [release fraction, frequency] pairs, from which a complementary cumulative distribution is created. This forms the point value result.

In the simplified case, the uncertainty analysis is often done separately and independently for release fraction and frequency. After this type of analysis one does not have [release fraction, frequency] pairs, but separate distributions for frequency and release fraction. In this case the problem is not as critical as in Fig. 23, since the distributions represent smaller areas in the complementary cumulative distribution, as shown in Fig. 30. However, if the release categories contain more than one accident sequences, the “points” in Fig. 29 are not actually points, but pieces of complementary cumulative distributions.

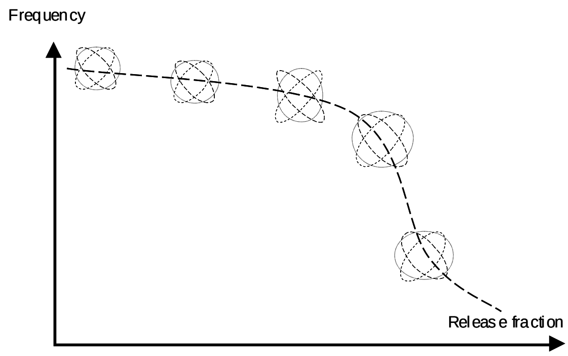

Fig. 30 Effect of different correlations on complementary cumulative distribution based on point values

It is possible to construct usable estimates for the integrated risk even when the base analysis is done with point values. However, it must be kept in mind that each APET in Level 2 PSA is complementary in nature: when the frequency of one accident sequence decreases, the frequency of other sequences increase. If the sequences lead to different releases, the whole APET is correlated.

Due to the complementary nature of APETs, the release categories are also correlated. This means that when the frequency of one release category increases, the frequencies of other release categories decrease. Thus, when speaking of uncertainty of the total release, the whole uncertainty analysis should be coupled to the logic of the APETs. Independently performing uncertainty analysis for release categories presents correctly the distribution of one release category, but does not describe its "inverse" effect on other release categories or the total release.

Sensitivity analyses may be even less useful, especially if they ignore the complementary nature of APETs and release categories.

In any case, it is possible to make simplified analyses and draw conclusions. It is important that the analysts are aware of the nature of the problem and possible side-effects of e.g. changing just the value of one variable without recalculating the complementary structure of APETs or release categories. Since the complementary nature of APETs tends to stretch the variation, independent approach may produce too narrow distributions and too flat complementary cumulative distributions.

9.4 USE OF EXPERT JUDGEMENT

9.4.1 Introduction

The Probabilistic Safety Assessments (PSAs) of nuclear power plants are based on reliability models of the plants safety systems, models of accident progression and physical models on various phenomena. Expert Judgement (EJ) is inevitably encountered in PSA: the models are based on engineering assumptions about the phenomena and the plant. Moreover, the selection of models' input parameters requires judgement due to inadequate empirical or statistical data, the choice between several models is done on the basis of judgements and, finally, the results of the analysis are interpreted and applied in decision making through judgement. PSA requires expertise from many different fields, which makes the application of judgement a complex and difficult task.

NUREG-1150 approach is the most well-known effort in this context (USNRC 1990). The EU Benchmark Exercise on Expert Judgement Techniques in Level 2 PSA examined various aspects related to the use of expert judgement in PSA (Cojazzi et al. 2001). The Risk Oriented Accident Analysis Methodology (ROAAM) is a systematic expert judgement method used in a few Level 2 PSA studies (Theofanous 1996).

9.4.2 NUREG-1150 method

To deal with judgement and to understand its impact on the analysis results and to take it into account in the safety related decision making, it is important that expert judgements are made explicitly. The NUREG-1150 report (USNRC 1990) presents the following principal steps for the formal expert elicitation process:

- Selection of Issues.
- Selection of Experts.
- Training in Elicitation Methods.
- Presentation and Review of Issues.
- Preparation of Expert Analyses.
- Expert Review and Discussion.
- Elicitation of Experts.
- Composition and Aggregation of Judgements.
- Review by Experts.

The EJ process followed in NUREG-1150 was exceptionally comprehensive and as formal as practically possible. Thus it can be regarded as the reference method when there are plenty of resources. An application of NUREG-1150 process for Belgian level 2 is presented in the Appendix.

9.4.3 EU benchmark exercise

To improve the identification and investigation of aspects related to the use of expert judgement in PSA and to encourage the use of proper analysis tools through the European Union, an international Benchmark Exercise on Expert Judgement Techniques in Level 2 PSA (BE-EJTs) project was arranged by the European Commission (Cojazzi et al. 2001). The main objectives of the project were: 1) the documentation of the different methods and techniques for handling judgement actually adopted among European PSA practitioners, 2) the comparison of the effectiveness of different ascertained expert judgement approaches in terms of the PSA needs of, for example, consistency, traceability, reproducibility and credibility and 3) the evaluation of the level of effort implied by the different methodologies by means of an analysis of the required resources. The project was organised in three phases: a survey phase (pre-phase), a first phase devoted to parameter estimation assessment and a second phase devoted to benchmarking expert judgement methods on a scenario development case.

The BE-EJTs project produced a large amount of data and results. During the course of the project a number of relevant reports and publications were issued by the partners taking part in the different phases of the project. The main findings and results of the whole project are summarised in the extended final report of the project (Cojazzi & Fogli 2000).

The project recognised that many EJ techniques are available, for tackling the issue of elicitation and aggregation of expert judgements in a structured way, but practical applications in Level 2 PSA have not been a common routine in the considered European Countries. There, EJ is often applied in an unstructured and even informal way.

One of the most important results of the BE-EJT project was the documentation of a number of structured expert judgement approaches. Both the principles of the methodologies as well as their application to both BE-EJT phases were documented thoroughly. A comprehensive framework was set up during the project for the comparison and assessment of structured expert judgement approaches. The framework did not only consider numerical comparison of final judgements but also aimed to assess the quality characteristics of the structured expert judgement process considering qualitative criteria, such as the applicability and traceability of the method, and quantitative criteria (e.g. the effort required for learning the approach).

According to the results obtained, the effects of structured EJ techniques were evident in comparison with individual estimates. Moreover the documentation and the controlled quality of any structured process made the results more credible and acceptable than individual assessments.

9.4.4 ROAAM methodology

The Risk Oriented Accident Analysis Methodology (ROAAM) (Theofanous 1996) provides a special approach to eliciting and combining expert judgements in the context of assessing and managing risks from rare, high-consequence hazards. It is suggested that rather than the usual 'formal treatments' on how to combine expert opinions that diverge widely, such 'uncertainty' must be approached in each case as a research question that encompasses frame of assessment, approach methodology, risk management, and safety goals, with the aim of obtaining resolution in a clear, consistent, and complete manner. Resolution of major severe accident issues relies heavily on developing an understanding of the underlying physics of the relevant containment phenomena. Implementation of the methodology is based on a comprehensive review effort that involves essentially all experts in the field, through an iterative and fully documented process towards resolution.

9.4.5 Conclusions

Due to the complexity of the phenomena handled by Level 2 PSA, use of expert judgement for treating uncertainties is needed e.g. to complement the results of simulation tools, or if no simulation results are available. Improvements in the simulation tools would help to reduce the use of expert judgement.

There are several challenges in the incorporation of expert judgements in a justifiable manner. Experience with the applications of the ROAAM methodology points to the need for a procedure and shared framework to facilitate the expert collaboration. Such practice should diminish communication gaps between the experts and enhances mutual understanding and comprehension of the physical phenomena. Further, the confidence in the final results will be increased when a well structured and documented EJ is part of the PSA.

9.4.6 References

[113] Cojazzi, G. & Fogli, D. Benchmark Exercise on Expert Judgement Techniques in Level 2 PSA, Extended Final Report, EUR 19739 EN, European Commission, Joint Research Centre, Luxembourg (2000)

[114] Cojazzi, G., et. al. Benchmark exercise on expert judgement techniques in Level 2 PSA, Nuclear Engineering and Design 209 (2001) 211-221

[115] Theofanous, T.G. On the proper formulation of safety goals and assessment of safety margins for rare and high-consequence hazards, Reliability Engineering and System Safety 54 (1996) 243-257

[116] USNRC, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, Final Summary Report, NUREG-1150, Vol.1 (1990)

9.4.7 Example: use of expert judgement for Belgian Level 2 PSA

9.4.7.1 Introduction

In the framework of the Belgian Level 2 Probabilistic Safety Assessment (PSA) update, a generic Accident Progression Event Tree (APET) has been developed for all Belgian units to evaluate the Containment Performance (CP-APET) and the Fission Product (FP-APET) release categories for a representative range of severe accidents.

The approach adopted in Belgium is based on the American one that can be found in the NUREG-1150 study [117]: the accident progression analysis is performed by means of a large and detailed APET. Such an approach provides a comprehensive representation of the possible sequences and ensures that the influences on their evolution are dealt with in detail and correctly propagated through the tree.

In that approach, expert judgement can be used for the quantification of basic events. However, before coming to expert judgement quantification, both the literature information and the plant specific engineering calculations (or a combination of both) have to be considered. Use of expert judgement is limited to basic events for which confidence level based on the other sources of information (literature and plant specific engineering calculations) is not sufficient or not satisfactory.

9.4.7.2 Methodology for expert judgement

Expert judgement is "expression of opinion, based on knowledge and experience, which experts make in responding to technical problems. Specifically, the judgement represents the expert's state of knowledge at the time of response to the technical question." [118].

To reduce the potential for inconsistency and promote a systematic approach to basic event quantification, the Expert Judgement (EJ) process applied for the current Level 2 PSA is based on the one of NUREG-1150 [117]:

1. Selection of issues for EJ (confidence level assessment).
2. Selection of experts.
3. Elicitation training.
4. Preparation of EJ issues.
5. Presentation meeting.
6. Expert judgement.
7. Final meeting.
8. Elicitation.
9. Aggregation and post-processing.

Some of these steps are detailed hereafter.

a) Selection of issues for EJ (confidence level assessment)

The selection of issues consists of defining the basic events that will be submitted to EJ. Before coming to an EJ process, both the literature information and the plant specific engineering calculations (or combination of both) have to be considered. Therefore, that selection takes place based on the possibility or not to find sufficient and satisfactory information in literature and/or by plant specific calculations.

According to [118], expert judgement is used: "If one or more of the following situation exists:

- No other means are available for quantifying an important issue.
- The information available is characterised by high variability.
- Some experts question the applicability of the available data.

- The existing results from code calculations need to be supplemented, interpreted or extended due to recent experimental results or deficiencies in the codes; or
- Analysts need to determine the current state of knowledge."

Practically, the assessment of the confidence level over the outcome of an issue identifies the need to proceed or not to EJ and is useful to further determine a global confidence level about the Level 2 PSA quantification results. If EJ is not necessary, the confidence level on the outcome of the issue under study is determined based on Table 35; otherwise, EJ process must be applied (see "Expert Judgement guidelines" paragraph).

Confidence level
• The expert is **extremely confident** because the outcome is supported by: • Detailed engineering calculations; • Separate analysis which supports the outcome; e.g. literature information; • Consideration of all uncertainties.
• The expert is **very confident** because the outcome is supported by: • Detailed engineering calculations, or published experimental data; • Consideration of all uncertainties.
• The expert is **pretty confident** because the outcome is supported by: • Detailed engineering calculations, or published experimental data that confirms the outcome.

Table 35 Confidence levels that do not require expert judgement

b) Selection of experts

The participants to the EJ are selected according to their knowledge on the issue in relation with the basic events. Three participants per basic events are foreseen, in addition to a referee who mainly takes care of the aggregation of the results. Those participants can be:

- Members of the Severe Accident Group: the members are selected according to the issues in which they have experience (follow-up of research projects concerning the issue, participation to projects dealing with the issue, …);
- Technical referees or senior experts in a specific issue such as thermal-hydraulics, radioprotection, structure, mechanics…;
- External experts with well-known international experience in the issue.

c) Elicitation training

The elicitation training dedicated to the selected experts has several purposes:

- Familiarisation with the EJ and its use in Level 2 PSA;
- De-biasing training: how to recognise and overcome familiar biases (overconfidence, use of a single source of information);
- Practical exercise, with attention to the decomposition of issue.

The training is used as a necessary basis to provide rules for the EJ process with the aim to ensure a coherent quantification despite the fact that participants are not the same for all the basic events.

d) Preparation of expert judgement issues

The experts involved in the EJ must be able to point out exactly what is expected from them at the end of the process (assigning split fraction probabilities, assigning values to parameters, setting distributions over uncertain parameters...).

The principle of the method suggested here consists of the decomposition of the issue into its controlling sub-parameters or phenomena before applying a probability assignment. This results in a number of decomposed elements among which many can usually be quantified using a more direct link to the well justified analysis methods or experimental results. The resulting issues can be expressed as a Decomposed Event Tree (DET) where only the unresolved/uncertain issues (decomposed BEs) are to be discussed during the EJ process.

The decomposition process is therefore important in both minimising the judgement by developing a structured framework which represents what is known about the issue (decomposition structure) separately from the pure judgement (probability assignment); and in providing a way of displaying clearly where it has been used. Different decomposition methods can be used. For example, the identification of uncertain parameters (contributing to a total BE probability) to which distributions will be assigned. Another example is the use of the DET approach in which the tree branch probabilities are to be assessed. Moreover, the combination of these two examples may lead to a third decomposition method.

However, the decomposition method is not always necessary for the EJ if the responsible technical person for the issue considers that the original decomposition (suggested in the APET) is sufficiently detailed.

e) Expert judgement guidelines

Expert judgement guidelines have been developed to take into account the following steps of the EJ process:

5. Presentation meeting.
6. Expert judgement.
7. Final meeting.
8. Elicitation.
9. Aggregation and post-processing.

The steps are detailed in the following paragraphs regarding the task of every participant.

Presentation meeting [Experts, Referee, Technical Responsible person]

1. The technical responsible person:
    a. Introduces the issue, explains the decomposition of issues and defines exactly what is to be quantified.
    b. Presents the available references and supporting calculations and makes sure that the experts have access to the latter.
2. The referee:
    a. Makes sure that the experts have a good understanding of the issue.
    b. Reminds the milestones and the deadline.

Expert judgement [Experts]

1. The experts:
    a. Must work on the issues independently from each other.
    b. May use the documentation presented by the technical responsible person, but may also use their own documentation.
    c. May use the supporting calculations presented by the technical responsible person, but may also perform their own.

Final meeting [Experts, Technical Responsible person, Referee]

1. The experts:
    a. Present their reasoning and pertinent references but without their conclusions.
    b. Agree on the required information that has to be provided for the elicitation.

2. The technical responsible person:
    a. Makes sure that all the available information is known by each expert.
3. The referee:
    a. Ensures that the experts share information without revealing their conclusions.

Elicitation [Experts, Referee]

1. The experts:
    a. Use the probability values recommended during the elicitation training which can help to put figures on their judgement.
    b. May consider Table 36 whenever asked to set distributions over uncertain parameters[16].
    c. Must perform their task according to the planning; any discrepancy must be explained and discussed with the referee.
    d. Must write after the final meeting a report presenting their results, explaining their arguments and quoting the references/calculations used in their judgement. This report has to be sent to the referee.

Table 36 Methods for selecting distributions

Method 1: Use of continuous distributions when there is no knowledge except for the bounds of a variable (uniform), or when there is knowledge of the bounds and of the most expected value (e.g. triangular, if values close to the bounds can be argued to fall off in likelihood).
Method 2: Use of continuous distributions when there is a strong underlying random controlling process (normal, lognormal …).
Method 3: Use of special distributions, either continuous or discrete, when there are physical reasons why certain values are expected to be more likely than others.

2. The referee:
    a. Reads the EJ report of each expert and, if necessary, asks the expert some questions about his approach, reasoning and assigned value/probability.
    b. Ensures that the assignment of the value/probability is in agreement with the amount of available complementary data which support the outcome.
    c. Asks the experts to review their judgement in case of large discrepancies of the assigned value between the experts or if the references/arguments are too poor (without communicating the values assigned by the other experts).

Aggregation and post-processing [Referee, Technical Responsible person]

1. The referee:
    a. Aggregates the values/probabilities of each expert with equal weight. The results are presented in the final version of the quantification report.
    b. Produces a summary of the expert results that is included in the quantification report.

[16] One has to be cautious in the use of continuous distributions which are not bounded (such as a lognormal distribution) because they could correspond to unphysical values, even if the probability related to these values is low.

c. Gives a level of confidence for the whole issue according to Table 37 that is included in the quantification report.

2. The technical responsible person:

   a. Uses probability tools if necessary to get the final probability from the aggregated values that are included in the final version of the quantification report.

If, in spite of the review of their judgement, the large discrepancies among the assigned probabilities of the experts are still observed or if the referee finds that the references/arguments are still too poor, a sensitivity analysis on the assigned probability is made for the BE. If the sensitivity analysis identifies a significant impact on the results, additional experts can be requested and other sources of information can be looked for to refine the EJ. The sensitivity analyses are performed later in the APET quantification results analysis.

Table 37 Confidence levels after expert judgement

Confidence level
4. The referee is **confident** with the experts results because: • The discrepancies of the assigned value/probability between the experts are small; • They are supported by literature and/or engineering calculations; • The uncertainties on the assigned value/probability are limited.
5. The referee is **quite confident** with the experts results because: • The discrepancies of the assigned value/probability between the experts are small; • They are supported by literature and/or engineering calculations; • The uncertainties on the assigned value/probability are large.
6. The referee is **little confident** with the experts results because: • The discrepancies of the assigned value/probability between the experts are large but the aggregation allows to obtain acceptable results; • The uncertainties on the assigned value/probability are large; • Sensitivity studies could be applied to the assigned value/probability to assess its impact on Level 2 PSA results.
7. The referee is **not confident** with the experts results because: • The discrepancies of the experts results are still* too large; • The uncertainties on the assigned value/probability are large; • Sensitivity studies must be applied to the assigned value/probability to assess its impact on Level 2 PSA results.

*It means that, in a first path, the referee asked the experts to review their judgement.

9.4.7.3 References

[117] USNRC, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, Final Summary Report, NUREG-1150, (1990)

[118] Ortiz N.R., Wheeler T.A., Breeding R.J., Hora S., Meyer M.A. and Keeney R.L. (1991), Use of expert judgement in NUREG-1150, Nuclear Engineering and Design, 126, pp 313-331

[119] Bolado R. and Badea A. (2008), Review of expert judgement techniques, SARNET-PSA2-D113

9.5 INITIAL ASAMPSA2 END-USERS SURVEY

This appendix provides the main conclusions of the initial ASAMPSA2 End-Users survey. The conclusions have been expressed in terms of “what should be the content of the ASAMPSA2 guideline”. The present version of the ASAMPSA2 guideline may not answer all demands but this can be reviewed and discussed during the second ASAMPSA2 workshop. This appendix will be updated in the final version of the guideline.

9.5.1 Executive summary of the survey

As part of work package 1 (WP1), a questionnaire comprising 116 questions was circulated among ASAMPSA2 partners and other organisations (End-Users: power plant operators, regulatory bodies) that have a stake in the performance or use of Level 2 Probabilistic Safety Assessments (PSAs) for Nuclear Power Plants (NPPs) and a database encompassing all responses was developed to analyse the results. As of October 14th, 2008 the survey was answered by 30 organisations listed in Table 1 out of more than 100 End Users to whom the questionnaire was sent. Amongst the 30 respondents which are included in this preliminary summary, 4 represent regulatory bodies, 6 are from TSOs who are involved in PSA mostly for regulators, 10 are from utilities, 5 from TSOs who are involved mostly for utilities and vendors, 4 from TSOs who may work for utilities, vendors and regulators, and 1 represents a vendor. This composition should result in a reduction of possible biases and one-sided opinions, and at the same time give substantial input from End-Users of the PSAs (utilities, regulators and vendors). The respondents represented opinions of 12 European countries (11 EU), but we note that 8 (almost 27%) have a stake in Finnish PSAs. Germany is represented by 5 organisations (17%) and France by 3 (10%). This may introduce some bias with respect to the point of view of “local practices and interests”.

This document provides a summary of the findings. The survey was organised in 5 sections, one for each of the possible main headings of the guidelines (general considerations, plant familiarisation, interface with Level 1 PSA, analysis of accident progression, and source terms and risk integration). Results are presented in the same order, with the exception of some major issues regarding the first section, which are of highest importance on the overall frame of the guidelines, and which the survey was not able to resolve. A synthesis of Responses is presented in Section 6 together with a suggested scope of the two PSA L2s (“full” and “limited”) by PSI.

The responses have been entered in an ACCESS database, which also performs some statistical manipulation of the data. In addition to summarising the results, the issues-resolution discussions at the October 28-29 meeting in Hamburg are incorporated, thus this document provides input and recommendations for WP2 and WP3 of the project, i.e. for the definition of full and limited scope Level 2 PSA. It was not easy in many cases to reconcile answers and make sense of the observations and justifications.

To this effect, we make the following preliminary notes and explanations:

1. Despite the number and make up of the respondents, the survey may have resulted in a poor technical basis for work on the guidelines in many areas. This is not only because only 30 out of more than 100 End Users to whom the questionnaire was sent provided their responses, but also since no answers were provided for 21% of the questions. In addition, for approximately 50% of the answers no technical input or reasoning was provided (see table for detailed statistic), even though it had been made clear at the beginning that some was expected, which made the interpretation of the responses rather difficult. We would like to point out one extreme respondent, who provided only 23 answers for 116 questions, and none of these answers was

accompanied by the expected technical reasoning. In total only 58% of the answers provided a "workable" input. The following shows a summary of the questionnaire and responses.

Number of questions where reasoning was requested to justify the answer:

Chapter 1: 21

Chapter 2: 4

Chapter 3: 10

Chapter 4: 17

Chapter 5: 9

TOTAL 61

RESULTS

No answer:	744	21%
Total no reasoning	942	51%
Clear answers	1925	55%
Answers without reasoning	630	34%

2. As a result of this, out of 57 questions that tried to identify users needs (what should be included or discussed in detail in the guidelines), 19 (33%) were judged unresolved in the initial evaluations and were subject to discussions during the Hamburg workshop. The others had been resolved either on the basis of absolute majority (> 66%) or on simple majority supported by the nature of the comments of the uncommitted respondents, if given. Except one, all these unresolved issues were resolved at the Hamburg workshop. Responses provided by the workshop participants aided in the resolution of the issues as presented in this document. This document presents the consensus of the ASAMPSA2 community plus the end users who participated at the workshop. PSI provided a suggested resolution to the unresolved issues based on the outcome of the discussions for the issue related to why L2PSA was conducted and which should relate to the top level objectives. Post workshop comments did not provide any comments to the resolution that was proposed.

3. This document is therefore able to identify users' needs for detailed discussions in the best practice guidelines, but for the most part the expected technical input for these issues cannot be found in the answers of the respondents. The interesting technical suggestions are identified in the summary tables, and the guideline developers should refer to the complete data base.

4. An unexpected observation about safety culture emerged from the responses: many of the utilities may currently perform PSA only because it is mandated. The discussions in the workshop made it clear that one large utility (EDF) has the intention to apply it for risk informed applications, as is already mandated in Finland. From post-workshop comments, other utilities may follow suit. However, the objective "risk informed applications" may be a more generic definition of the objective "risk reduction options", which is identified as one of the top six objectives for performing Level 2 PSA, and is part of regulatory requirements. Perhaps the guidelines may reduce these two top level objectives to just one, when considering applications of the PSAs.

5. The answers to the questions did not provide any firm conclusion that objectives on improving safety culture set in the 1990s by the IAEA and OECD have been fully met. The most striking example of this attitude can be found in the discussion of results for questions 1.2, 1.3, and 1.10 in section 6. Some respondents from the utilities' side do not find any use for their existing PSAs. This fact seems recognised by some organisations (e.g. in Belgium) as being a specific issue with the management of some of the plants.

6. This may be in part due to the fact that the prevailing thought concerning Level 2 PSA is on mitigation of risks (in the ALARP philosophy), rather than prevention. Prevention is normally addressed in Level 1 PSA, and therefore most practitioners may think that risk is already minimised without particularly addressing prevention of risk in Level 2 PSA, especially once SAMGs are implemented. It seems that the definition of risk is not generally in the mind of the community, and hence, risks to the environment and the public are largely overlooked.

7. Part of the problem related to point 6 may be, in fact, the possible bias introduced by the composition of respondents (what is called throughout this document "the community"). Only few of the EU authorities have so far defined "hard" safety goals for severe accidents. Of these few, two are missing from the responses to the survey, namely the Netherlands and the UK. In particular, these two countries are the only ones in Europe that have safety goals which absolutely require the performance of a Level 3 PSA to provide a safety demonstration. However, even if participants from the UK and the Netherlands had contributed to the survey, they would have still represented a minority opinion on the subject of safety goals, risk analyses and interface with Level 3 PSA.

8. As a consequence, for the vast majority of the respondents there appear to be scarce interest or reluctance in extending PSAs to the assessment of offsite risks although L2PSA delivers the key information, e.g. release frequencies, magnitudes, etc, to L3PSA. Rather than seeing PSAs as integral efforts, the vast majority of respondents seem to prefer a well separated approach, and do not seem to consider offsite risks (with perhaps some extensions to the Level 2+, where consequences may be assessed in a simplified manner), which in the opinion of some respondents belong only to Level 3 analyses. This consideration appears confirmed by the comments to question 3.7 (interactions between Level 1 and Level 2, review of Level 1, understanding of Level 1 results, consistency of the different parts of a PSA, etc.). At any rate, the survey points out (and the workshop participants agreed) that the guidelines should not be overly concerned with the interface Level 2 - Level 3.

9. It appears also that the interest to perform assessment of risks to the environment and the public is limited in some countries because, from the discussions in question 1.8, it seems that in general authorities tend not to strictly enforce legislation even when it exists.

10. As in the SARNET survey [1], a large part of the responses provide a rather vague understanding of the difference between uncertainty in occurrence of events, and uncertainty in the probability of occurrence of the same events (or perhaps the difference between epistemic and aleatory uncertainties). It seems that

most of the effort is spent on the first aspect. What is missing in most guidelines (and this may also be reflected in most Level 2 PSAs) is guidance for a robust quantification of probabilities, but only if the probability of occurrence is shown to be risk significant. This was agreed upon at the Hamburg workshop.

11. As a show of contradiction, most of the respondents provide a consistent answer: the ultimate goal should be to assess offsite risks and consequences, however an agreement on the basic problem 'what should common safety goals be' could not be established. In fact, the preferred risk measure is only a surrogate metric (LRF or LERF), and has little meaning in terms of risks. It is not clear, as already noted about the definition of LERF in SARNET, whether common definitions of "large", "early" and "frequency" can be achieved in the community. Due to the disparate and probably irreconcilable points of view on the subject (especially due to the local regulatory context), the ASAMPSA2 project may have to exclude harmonisation of safety goals. However, by contractual agreement, the guidelines must discuss at some level the existing practices, especially in relation with depth of analyses.

9.5.2 Technical issues to be considered in the ASAMPSA2 guideline

The following tables summarise technical conclusions of the End-Users survey.

Table 38 Final recommendations on users needs with respect to general considerations

Section	Need
Not in the questionnaire -Presentation of results	**Guidelines should propose some guidance on the different way of presenting and analysing the results**
Not in the questionnaire - Quality assurance	**One chapter should be dedicated to quality assurance aspects**
1.1 Should WENRA recommendations be accounted for?	Yes, but not in detail, because WENRA recommendations are at a very general level only.
1.2, 1.3, 1.10a, 1.10b Prioritisation of objectives of Level 2 PSA (see Section 6)	**Cross referencing responses from the point of view of performers, end users, authorities, project management, and present applications, indicates that Level 2 PSA may be performed for the most part to fulfil regulatory requirements, hence the local definition or lack of definition of safety goals has a strong influence on the depth of the analyses. The outcome of the survey from respondents suggests that the best practice guidelines to be developed by WP2+WP3 of ASAMPSA2 project should concentrate on regulatory requirements. One concern is raised by stakeholders in Finnish PSAs, that the community as a whole (including ALSO all responses related to these stakeholders) shows no interest in risk informed applications, while this issue should be the primary objective** of their PSAs. Should ASAMPSA2 accommodate this concern and add specific minimum requirements for this type of application, or should the application be subsumed into the most stringent requirements? **Note however that one of the top objectives included in regulatory requirements is "to provide input to risk reduction options", which should cover the concern.**
1.4 Is the list of tasks exhaustive	Yes, the list covers all foreseeable tasks hence the items should provide the headings for the contents of the guidelines. Please check database on this question, with respect to CONTENT of some of the specific tasks.

Section	Need
1.5 (Related to 1.2, 1.3, 1.10a, 1.10b) Characterisation of tasks vs. top level objectives, or how tasks should be performed See Section 6	A table providing the required level of details (minimum requirements) as a function of applications shall be included in the guidelines when discussing top level objectives (i.e., Table 6.10 in Chapter 6). This table should show in detail only the objectives identified as most important from 1.2, 1.3, 1.10a, and 1.10b and any special request by stakeholders in Finnish PSAs, if agreed upon. A working table (Table 6.10) is provided in Section 6. PSI welcomes comments for Table 6.10 from the respondents to questionnaire.
1.6 Should risk of the environment be the primary goal of a PSA?	Yes, this should be kept in mind in the redaction of the guidelines and their content.
1.9 Definition of common safety goals	No attempt for harmonisation of local safety goals in the scope of ASAMPSA2 but a description of current practices in the different countries should be provided (also as input to other EU projects, and also NPSAG) at a minimum, together with proper presentation of results.
1.11Should standardisation of tools be recommended	Not as a recommendation. Sharing of experiences is preferable.
1.11 Should sharing of resources be recommended	Yes.
1.12Should guidelines include an appendix on codes	The community is not entirely convinced but there are eight volunteers. This appendix should not be provided, however strengths and limitations should be discussed in detail, see 1.14. Therefore the volunteers may provide their expertise to task 1.14.
1.13Should use of minimal cutsets be encouraged in Level 2 PSA	Both integrated and non integrated approaches should be described in the guidelines, with a presentation of their advantages and disadvantages. Use of MCSs as opposed to accident sequences (PDSs) should be described and possible inconsistencies introduced in the analyses by using either approach must be discussed.
1.14 Should there be guidelines on the use of codes	Yes, an appendix should be provided, describing strengths and limitations issue by issue but not entering into too many details. A detailed description of codes would be too resource intensive for ASAMPSA2. Specific attention should be paid to PSA needs (in contrast to deterministic studies). Discussions with other groups (e.g. GAMA) may be useful.
1.15 What uncertainties should be included?	The impact of uncertainty analysis for final L2PSA application should be described in the guidelines. The impact is dependent on the objectives of the L2PSA study

Section	Need
	The guidelines should distinguish between: • uncertainty analysis for the study of specific issues (where sensitivity analysis may be used, e.g. to address completeness and model uncertainties), • quantification of uncertainties in the event trees, if necessary to fulfil objectives • propagation of uncertainties through L2PSA quantification and presentation of results, if necessary to fulfil objectives. The experience in practices, even though limited, should be described in the guidelines.
1.16b Should the guidelines provide guidance on sensitivity analyses	Yes.
1.16c Should the guidelines recommend the use of sensitivity analyses to aid in the quantification of uncertainties?	Yes.
1.17 Should the guidelines include a section on peer reviews	No, specialised guidelines already exist

Table 39 Users needs with respect to plant data

Section	Need
2.1a Which plant data should be considered crucial?	Guidelines developers please refer to data-base. A frequent response appears to be with respect to containment fragility and leak paths, and these should be covered in detail.
2.1b Should the guidelines stress in detail the description of containment systems (and related operator interventions).	Yes
2.1c Should the guidelines stress in detail the description of accident management systems (and related operator interventions)	Yes
2.1d Should guidelines insist that historic test data is used for containment leak rates	Yes. It appears that in practice historic data is seldom used. **The data should also be cross-checked with local requirements for leaktightness of the containments.**
2.1e Should functions outside of the primary containment be credited, and a recommendation made in the guidelines?	**Yes**

Table 40 Users needs with respect to interface with Level 1 PSA

Section	Needs
3.1 Should guidelines include a discussion of pitfalls (including, but not limited to the interface between Level 1 and Level 2)?	**Yes. All specific difficulties encountered by practitioners should be described and existing technical solutions discussed.**
3.2a Common Level 1 - 2 mission times	**No. The ASAMPSA 2 guidelines may use SARNET outcomes on the definition of final End States for L2PSA (stable plant state, no more significant releases expected).**
3.2b Should the community adopt a common definition of mission times	**See above**
3.2c/d What mission time would be appropriate?	**See above**
3.3a Should the guidelines define a criterion for the definition of core damage	No, but definitions may have to be discussed in the guidelines as a function of top objectives of the analyses
3.3c Should the guidelines provide guidance to identify Level 1 sequences that do not lead to severe accident	No, these sequences may be beyond the scope of Level 2 PSA and its objectives
3.4a Should the guidelines provide detailed guidance for selection of representative sequences	Yes
3.4b Should ASAMPSA2 provide specific examples	Yes, refer to database for volunteers
3.5 Are there organisations willing to share experiences on shutdown states	10 organisations said yes. Refer to the data-base for identification and inputs that can be provided
3.6 Interfaces depending on Level 1- 2 integrated approach or not	**The guidelines should describe both integrated and not integrated methodologies. Advantages and disadvantages of both methodologies should be described.**
3.7 How to deal with conservatism in level 1 analyses	**The guidelines should emphasise the need for a good communication between L1 and L2PSA teams and also with radioprotection specialists (consequences of accidents)**

Table 41 Users needs with respect to accident progression

Section	Needs
4.1a Closure of issues in accident progression	An issue is closed when L2PSA developers may find enough knowledge or validated codes for the assessment of risks (depending on the metrics of risk, e.g. containment failure probabilities, releases, others) related to this issue. The guidelines should discuss the "closure" of the issues on a plant-type basis
4.1b Guidance should be given in screening uncertainties to select those having most contribution to risk	Yes
4.1c Analysis and quantification of all uncertainties is necessary?	The guidelines should underline the fact that there is no need for a quantification of all uncertainties but only for important aspects regarding the results of the study. The guidelines should stress that the analysis and models may depend on the objectives of the L2PSA and may be plant specific.
4.2 What are the relevant phenomena?	"Relevant phenomena" are plant specific and a definition depends on the objective of the study. The guidelines shall specify that only validated codes should be used in support of L2PSA performance and that blind use of codes must be avoided. For this a discussion of major deficiencies and limitations of integral codes should be provided.
4.3 Should cost-benefit analyses be discussed	No
4.4a Should guidelines provide templates for event trees	To progress in harmonisation of EU L2PSAs, the development of generic skeletons of event trees for inclusion in the guidelines seems to be an efficient approach. Warnings should be provided on plant specific issues.
4.5a/b Would the community endorse common generic split fractions	To progress in harmonisation of EU L2PSAs, the development of generic split fraction seems to be an efficient approach. Warnings should nevertheless be provided on plant specific issues and objective specific needs (e.g. the requirements imposed by safety objectives or dependencies on accident progression may have an influence on the orders of magnitude of low probability events or effects).
4.6a Should guidelines provide specific guidance on some containment failure modes	Yes on all four items indicated in the question
4.7 The guidelines should include a discussion on possible influence on containment fragility from other internal and external events	Yes

Section	Needs
4.8.1 Appendix on MCCI	Yes
4.8.2 Appendix on FCI	Yes
4.8.3 Appendix on vessel failure	Yes
4.8.3a Appendix on vessel uplift and DCH	Yes
4.8.4 Appendix on induced SGTR and passive ruptures	Yes
4.8.5 Appendix on pressure suppression pool phenomena	Yes
4.8.6 Appendix on hydrogen combustion	Yes
4.8.7 Appendix on impact of SAMGs	Yes
4.9a Section on formal expert judgement	Yes. ASAMPSA2 project can rely on SARNET outcomes for this subject.
4.9b Include some guidance for the internal expert judgement with examples	Yes. See above.

Table 42 Users needs with respect to source terms

Section	Needs
5.1a The way of grouping the radioactive isotopes is an issue that should be addressed in detail in the ASAMPSA 2 guidelines	**Yes. This item should be described with reference to integral codes and specialised codes. Prioritisation of important isotopes (with respect to consequences) should be described with some care.**
5.2 Appendix on Iodine	Yes
5.3 Appendix on Ru	**Yes. It may be explained why some organisations should have an interest in Ruthenium. The available state of the art should be presented. The concern is about releases from the core but also from spent fuel pools.**
5.4 Guidance on isolation failure	Yes.
5.5a Guidance for source terms uncertainties	Yes, 8 organisations volunteer to share their experiences
5.6a Should the guidelines adopt the ASME policy on source terms uncertainties	Communication with US NRC on this topic may be useful. (ASME position may change). No recommendation yet.

Section	Needs
5.6b Should the guidelines adopt the German guidelines policy on source terms uncertainties	No
5.7a Should source terms be provided with auxiliary data (for Level 3)	**To be discussed with respect to top level objectives.**
5.7b Should there be guidance on propagation of uncertainties to Level 3	**Data needs for L2+, L3PSA tools should be identified (e.g. current EU programs about emergency preparedness for nuclear installations). No specific attempt to describe in detail interfaces with L3, which is not one of the applications identified by the community as of interest.**

9.5.3 References

[120] 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