ASAMPSA_E SEVENTH FRAMEWORK PROGRAMME	Advanced Safety Assessment Methodologies: extended PSA	EURATOM

"NUCLEAR FISSION"

Safety of Existing Nuclear Installations

Contract 605001

Summary report on the impact and experience feedback of the previous ASAMPSA2 project

Reference ASAMPSA_E

Technical report ASAMPSA_E / WP40 / D40.1 / 2013-3

IRSN Reference PSN-RES/SAG/2013-00413

H. Löffler, GRS
S. Morandi, RSE
H. Bonneville, IRSN
E. Raimond, IRSN
L. Burgazzi, ENEA

Period covered: from 01/07/2013 to 31/10/2013	Actual submission date: 31.10.2013	
Start date of ASAMPSA_E: 01/07/2013	Duration: 36 months	
WP No: 40	Lead topical coordinator : H. Löffler	His organization name : GRS

Project co-funded by the European Commission Within the Seventh Framework Programme (2013-2016)		
Dissemination Level		
PU	Public	Yes
RE	Restricted to a group specified by the partners of the ASAMPSA_E project	No
CO	Confidential, only for partners of the ASAMPSA_E project	No

ASAMPSA 2 Quality Assurance page

Partners responsible of the document:	GRS
Nature of document	Summary report on ASAMPSA2 experience
Reference(s)	Technical report ASAMPSA_E/ WP40 / D40.1 / 2013-3 IRSN Reference PSN-RES/SAG/2013-00413
Title	Summary report on the impact and experience feedback of the previous ASAMPSA2 project
Author(s)	H. Löffler. S. Morandi, H. Bonneville, E. Raimond, L. Burgazzi
Delivery date	November 06, 2013
Topical area	severe accidents, probabilistic safety assessment, ASAMPSA2
For Journal & Conf. papers	No

The objective of the present document is to provide a summary on the impact and experience feedback of the previous ASAMPSA2 project and to identify lessons learned during the ASAMPSA2 project which may be beneficial for the ASAMPSA_E project.

The structure of the present document follows the structure of the ASAMPSA2 documentation. There is a short summary for each of the topics covered in ASAMPSA2, followed by conclusions with regard to ASAMPSA_E.

Additional sections summarize the impact of ASAMPSA2 on the ASAMPSA2 partners, and on the PSA community in general.

A final section provides some comments on the achievement of harmonization in L2PSA and mentions experience gained in the working process of ASAMPSA2 which may be interesting for ASAMPSA_E.

Visa grid			
	Main author(s) :	Verification	Approval (Coordinator)
Name (s)	H. Löffler	O. Mildenberger	E. Raimond
Date	06. Nov. 2013	06. 11. 2013	13/11/2013
Signature	[signature]	[signature]	[signature]

Modifications of the document

Version	Date	Authors	Pages or paragraphs modified	Description or comments

Distribution list

EC scientific officer for ASAMPSA_E

Name	First name	Company
Hugon	Michel	EC

Representatives of EC ASAMPSA_E partners

Name	First name	Company
Alzbutas	Robertas	LEI
Armingaud	François	IRSN
Banchieri	Yvonnick	EDF
Bardet	Lise	IRSN
Bareith	Attila	Nubiki
Baumont	David	IRSN
Benitez	Francisco Jose	IEC
Benzoni	Stéphane	EDF
Bernadara	Pietro	EDF
Bogdanov	Dimitar	TUS
Bonnet	Jean-Michel	IRSN
Bonnevialle	Anne-Marie	EDF
Borysiewicz	Mieczyslaw	NCBJ
Bosso	Sandrine	Tractebel
Brac	Pascal	EDF
Brinkman	Hans (Johannes L.)	NRG
Burgazzi	Luciano	ENEA
Cazzoli	Errico	CCA
Claus	Etienne	Tractebel
Clement	Christophe	IRSN
Corenwinder	François	IRSN
Coulon	Vincent	EDF
Cordoliani	Vincent	Areva
Decker	Kurt	Vienna Univ.
Del Barrio	Miguel A.	Iberdrola
Denis	Jean	IRSN
Dirksen	Gerben	Areva

Duflot	Nicolas	IRSN	Petar	Kaleychev	TUS
Duluc	Claire-Marie	IRSN	Klug	Joakim	LRC
Dupuy	Patricia	IRSN	Knochenhauer	Michael	LRC
Durin	Thomas	IRSN	Kolar	Ladislav	UJV
Dybach	Oleksiy	SSTC	Kollasko	Heiko	Areva
Gallois	Marie	EDF	Kowal	Karol	NCBJ
Georgescu	Gabriel	IRSN	Kubicek	Jan	UJV
Godefroy	Florian	Areva	Kumar	Manorma	LRC
Gryffroy	Dries	Bel V	La Rovere	Stephano	NIER Eng.
Guigueno	Yves	IRSN	Lajtha	Gabor	Nubiki
Guimier	Laurent	IRSN	Lanore	Jeanne-Marie	IRSN
Grindon	Liz	AMEC	Laurent	Bruno	IRSN
Groudev	Pavlin	INRNE	Loeffler	Horst	GRS
Häggström	Anna	LRC	Lopez	Julien	EDF
Halada	Peter	VUJE	Matuzas	Vaidas	LEI
Hashimoto	Kazunori	JANSI	Ménage	Frédéric	IRSN
Henssien	Benjamin	EDF	Michaud	Laurent	Areva
Hibti	Mohamed	EDF	Mildenberger	Oliver	GRS
Hugon	Michel	EC	Mitaille	Stanislas	Tractebel
Hultqvist	Göran	FKA	Morandi	Sonia	RSE
Hurel	François	Arexis	Mustoe	Julian	AMEC (UK)
Hustak	Stanislav	UJV	Nitoi	Mirela	INR
Ivanov	Ivan	TUS	Nonclercq	Philippe	EDF
Jan	Philippe	EDF	Olsson	Anders	LRC
Jaros	Milan	UJV	Oury	Laurence	Tractebel .

REPRESENTATIVE OF ASSOCIATED PARTNERS

Name	First name	Company
Hirata	Kazuta	JANSI
Hashimoto	Kazunori	JANSI
González	Michelle M.	US-NRC
Yamanana	Yasunori	TEPCO

CONTENT

INTRODUCTION

The objective of the present document is to provide a summary on the impact and experience feedback of the previous ASAMPSA2 project and to identify lessons learned during the ASAMPSA2 project which may be beneficial for the ASAMPSA_E project.

The structure of this present document follows the structure of the ASAMPSA2 documentation. There is a short summary for each of the topics, followed by conclusions with regard to ASAMPSA_E.

1 SUMMARY OF ASAMPSA2 DOCUMENTATION AND CONCLUSIONS FOR ASAMPSA_E

The objective of the ASAMPSA2 project was to develop best practice guidelines for the performance and application of Level 2 probabilistic safety assessment (L2PSA), for internal initiating events, with a view to achieve harmonization at EU level and to allow a meaningful and practical uncertainty evaluation in a L2PSA. The project has been supported and funded by the European Commission in the 7th Framework Programme. It has been established through a collaborative effort of 21 European organizations.

At the end of the ASAMPSA2 project, the guidelines have been submitted to an international external review open to European nuclear stakeholders and organizations associated to the OECD-CSNI working groups on risk and accident management. An international workshop was organized in Espoo, Finland, hosted by FORTUM, from 7th to 9th of March 2011 to discuss the conclusions of the external review. This final step for the ASAMPSA2 project occurred just before the Fukushima Daiichi disaster (11th of March 2011). All lessons from the Fukushima accident, from a severe accident risk analysis perspective, could not be developed in detail in the final version of the ASAMPSA2 guideline.

The guideline is published on CORDIS: http://cordis.europa.eu/fp7/euratom-fission/funded-reports_en.html. It is also available on www.asampsa2.eu. It includes 3 volumes:

Volume 1 - General considerations on L2PSA [1]

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

Volume 2 -Technical recommendations for Gen II and III reactors [2]

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

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

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.

One important quality of the document is that is has been judged acceptable by organizations having different responsibilities in the nuclear safety activities (utilities, safety authorities or associated TSO, research organizations, designers, nuclear service companies, …).

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

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

The ASAMPSA2 guidelines have to be considered as a technical complement of other existing "high level" guidelines like [4] or certain national guides. They propose practical solutions and try to define what could or should be done to obtain a state-of-the-art study. It was not the intention of the authors to define any quantitative or qualitative safety requirement. This activity is the responsibility of the national safety authorities.

Conclusion for ASAMPSA_E:

The ASAMPSA2 guidelines are a huge document. This is due to the wish to work at a technical level. The ASAMPSA_E project also has a technical objective, and therefore should lead to a large documentation as well.

For ASAMPSA_E it seems reasonable to propose like in ASAMPSA2 a set of acceptable existing solutions for all issues to be addressed without trying to propose a precise step-by-step procedure.

The ASAMPSA_E project has much more internal structures than the previous ASAMPSA2, and in contrast to ASAMPSA2, specific documents are foreseen for particular topics. Therefore, although the total volume of documentation in ASAMPSA_E is expected to be large as well, it will be better structured and more easily accessible.

1.1 VOLUME 1 - GENERAL

The following sections provide a short summary of the first volume of the ASAMPSA2 documentation [1]. In each sub-section there is a conclusion for ASAMPSA_E.

1.1.1 THE THREE LEVELS OF PROBABILISTIC SAFETY ASSESSMENT

A definition of the three levels of probabilistic safety assessment can be found in IAEA Safety Standard SSG-4 [4]: "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 recognized:

(1) In Level 1 PSA, the design and operation of the plant are analyzed 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 L2PSA, 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. L2PSA 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."

Conclusion for ASAMPSA_E:

The common three levels of PSA are also a practical concept for extended PSA. There is no indication that the common level definitions are not adequate. However, in order to take into accout also acccidents in the spent fuel pool, a more general definition such as fuel damage should be preferred over core damage.

1.1.2 STRUCTURE OF A L2PSA AND RELATED ACTIVITIES

The intention of this chapter in [1] is to give an overview of a L2PSA project. The following issues are addressed:

- Quality assurance programme
- Plant familiarization
- Definition of the L2PSA objectives
- Accident sequences analysis, analysis of phenomena, source term analysis
- Containment analysis
- Human reliability analysis
- Systems analysis
- Event tree modeling
- Quantification, result presentation and interpretation
- Documentation
- Management of a PSA in support of the objectives
- Independent review
- Communication of L2PSA results

<u>Conclusion for ASAMPSA_E:</u>

The principal structure of a L2PSA is applicable also for those issues which have to be taken into account in an extended PSA, as considered within ASAMPSA_E. However, a few remarks are due:

- Specific issues which concern the spent fuel pool or open RPV have not been addressed.
- Vulnerability of vital structures (e.g. containment, emergency diesels, water intake buildings, secondary and auxiliary buildings, site accessibility for human actions needed for severe accident management …) due to external events is not considered.
- Multi-unit accidents have not been addressed in ASAMPSA2.

Taking into account such issues will not alter the general approach and structure of L2PSA as proposed in [1], but there may be a shift of perspectives and priorities. It is one of the objectives in ASAMPSA_E to appropriately address such issues and to incorporate them into the L2PSA project structure.

1.1.3 THE CURRENT SITUATION REGARDING L2PSA ACTIVITIES AND APPLICATIONS

In [1] there is a chapter which presents a review of the current background regarding L2PSA activities and applications. It introduces the general situation at international level. The chapter provides some global views on the different stakeholders' positions.

<u>Conclusion for ASAMPSA_E:</u>

There are several references which address L2PSA in general, without detailing the NPPs' operation mode or the type of events. Their relevance for ASAMPSA_E should be checked. In addition, there are few documents which explicitly mention shutdown states of the reactor or external events – these documents are certainly relevant in ASAMPSA_E. However, there is almost no information about spent fuel pool accidents in ASAMPSA2. Deliverable 40.2 of ASAMPSA_E has the title: "Summary on published guides". This deliverable will be a suitable document for addressing the current situation in general, while the technical deliverables will have to address particular issues.

1.1.3.1 <u>IAEA Safety Guide on Application of L2PSA for Severe Accident Management</u>

The Safety Guide [11] provides recommendations on the use of L2PSA 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. In particular the Safety Guide recommends to use the results of L2PSA 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 L2PSA or by any other method.

1.1.3.2 IAEA Safety Guide on L2PSA and Applications

Within WP40 of ASAMPSA_E, L2PSA is addressed. Therefore, the IAEA safety guide on L2PSA and applications is of particular interest. This Safety Guide [4] includes all the steps of the L2PSA process, and includes also the determination of the detailed source terms that would be required as input to a Level 3 PSA. This Safety Guide addresses only the aspects of PSA that are specific to L2PSA. The Safety Guide describes all aspects of the L2PSA that need to be carried out if the starting point is a full scope L1PSA. The objective of this Safety Guide is to provide recommendations for meeting the requirements in performing or managing a L2PSA project for a nuclear power plant (NPP). The Safety Guide is structured in accordance with the major tasks of a L2PSA.

1.1.3.3 OECD Technical Opinion Paper on L2PSA

A significant publication is the Technical Opinion Paper (TOP) on L2PSA [12]. 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 L2PSA 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."

1.1.3.4 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. 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 [5],
- The Safety Objectives for new Power Reactors [6].

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 [5]:

"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.

1.2 The second document on the Safety Objectives for new Power Reactors 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.

(…)

3. Accidents with core melt

- accidents with core melt which would lead to early or large releases have to be practically eliminated;
- 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. (…)"

Conclusion for ASAMPSA_E:

WENRA Reference Levels are being updated to take into account the situation after the Fukushima accident. Many Reference Levels now address DECs (design extension conditions) which cover hazards and severe accidents. The chapters on WENRA in ASAMPSA_E will have to be updated accordingly. The link between the DECs and extended PSA concepts should be discussed in detail by the ASAMPSA_E partners.

1.1.3.5 National situation

This chapter in [1] provides examples of L2PSA and associated rules in different countries. Much information is available in the respective sections. A few examples which are of particular interest for ASAMPSA_E are summarized here.

For both NPPs in the Netherlands (one has been decommissioned), a full scope PSA (levels 1, 2 and 3) for all operating states (power and low-power/shutdown) and for all internal, external and area events has been performed. The level 3 PSA of the HFR (a tank in pool research reactor) has been limited to full power states only. In the late 90s, the CSN and the NPP agreed to develop a program for the creation and use of PSA in Spain, 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.

In Sweden, PSA should include level 1 and level 2, and 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, including area events and external events.

In the UK, the PSA for newer 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.

In Hungary 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. L1 and L2PSAs are required for a NPP covering all operational states, modes and initiating events.

Conclusion for ASAMPSA_E: The information available on national situations in ASAMPSA2 guidelines need to be updated (recent evolution) and completed for ASAMPSA_E issues (spent fuel pool, external hazards …).

1.1.3.6 Impact of Fukushima accident on L2PSA activities

In [1], taking into account early experiences from the Fukushima accident, it is concluded from a PSA perspective point of view:

- PSA practice should no longer be limited a priori to a certain set of events or sequences (e.g. restriction to full power plant status, or neglecting certain initiators like external floods). Only the PSA itself can provide justification for discarding events or phenomena. Within a PSA, a dedicated screening process should find out all significant issues. The only acceptable a priori restriction of the scope could be the need to keep certain security issues secret.
- There should be a strong incentive to fill knowledge gaps in fields which have not found much attention until now:
    - Accident sequences in shut down mode, including open RPV,
    - Accidents in the spent fuel pool,
    - Fission product behavior, reducing the existing large uncertainties in release fractions to the environment,
    - Accident prevention and mitigation by “unconventional” human actions.
- Operating staff and crisis teams should be trained in response to extreme events and severe accident management.

Conclusion for ASAMPSA_E: The Fukushima accident is a main reason for the ASAMPSA_E project even if importance of external events on NPP risks was previously hightlighted by some experts.

1.1.4 RISK MEASURES

The sections 4, 5, and 6 in [1] are related to issues which are covered by WP30 in the ASAMPSA_E project. The sections have the following headings:

- Risk measures / safety indicators, Presentation and Communication of L2PSA results
- Complementary risk measures / safety indicators based on extended L2PSA

- A proposal for a Common Risk Target

These sections describe the different risk measures / safety indicators that may be calculated by a L2PSA and considered as state-of-the-art. The definition of risk measure is a key issue for the communication of the L2PSA results. In addition, some complementary measures for L2PSAs are provided, where the level 2 information is complemented with additional information to derive some results in the direction of results that are expected from a level 3 PSA.

A proposal of a Common Risk Target was made by Jirina Vitazkova and Erik Cazzoli representing the CCA company within the project ASAMPSA2. However, it should be noted that it does not fully reflect the opinion of the majority of the community participating in the ASAMPSA2 project.

<u>Conclusion for ASAMPSA_E:</u>

For the ASAMPSA_E project it can be assumed that probably all statements in [1] related to this general topic are valid. However, as indicated above, there was no complete consensus in the ASAMPSA2 community with regard to a common risk target. Within WP30 of ASAMPSA_E (“general issues”) there is a possibility to address this issue and to verify or modify the views given in [1].

1.1.5 L2PSA APPLICATIONS

This chapter in [1] tentatively discusses the possible applications of the L2PSA. For WP40 in ASAMPSA_E, many sections are of interest. They cover general aspects which can also be applied to the issues to be addressed in ASAMPSA_E. Where appropriate, it is indicated which ASAMPSA_E deliverable will address the issue.

- L2PSA quality and content for various end user needs
- Requirements for presentation of results
- Checking the validity of the conclusion regarding the known weakness of the tools, quantification that have been used.
- Identification of containment failure modes, plant vulnerabilities, validation of the design
- Assessment of releases
- Development or validation of severe accident measures. Section 7.6 in [1] directly addresses the topic which will be addressed in ASAMPSA_E within the deliverable D40.5 (SAMG validation by L2PSA).
- Link between L2PSA and research programs

  [1] provides a list of R&D topics which are of high interest for L2PSA development. According to the scope of [1] this list does not include issues related to external initiating events. This topic will be covered in ASAMPSA_E in deliverable 40.6 (SD states and SFP and recent R&D).
- Capitalization of knowledge – Living L2PSA – training

  L2PSA for normal operation is a mature technique, and it can and should be used for training responsible staff, as indicated in [1].

- Emergency preparedness: L2PSA analyses and results are very valuable for planning off-site emergency measures in order to protect the public. This issue could be addressed in ASAMPSA_E within the deliverable D40.5 (SAMG validation by L2PSA).

1.1.6 SPECIFIC ISSUES RELATED TO SHUTDOWN STATES

One of the topics to be addressed by ASAMPSA_E is L2PSA for shutdown states. It will be reported in deliverable D40.6. [1] contains a 3-page chapter dedicated to this topic. In particular it highlights

- Open containment
  - Depending on the outage management, it may be difficult or time consuming to close the containment.
  - Low power and shutdown PSA is typically divided into different parts according to plant operating mode. Those with open containment are:
    - Cold shutdown,
    - Refuelling.
- 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 vaporization from the RPV begins, access by rescue teams to the RPV (or to the containment in general) may no longer be possible.
  - 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 air).
- System availability
- Success criteria for phenomena mitigation
- Source term evaluation for shutdown sequences.

Conclusion for ASAMPSA_E:

The [1] subchapter on low power and shutdown L2PSA will serve as starting point for the ASAMPSA_E D40.6 deliverable.

1.1.7 INITIAL ASAMPSA2 END-USERS SURVEY

[1] contains a summary of the initial end-user survey, performed at the beginning of the ASAMPSA2 project.

Conclusion for ASAMPSA_E: Since ASAMPSA_E will have its own interaction with end-users, the results of the ASAMPSA2 survey may not be applied directly. However, the experience in performing the survey should be transferred to the survey to be done within ASAMPSA_E in WP10.

1.2 VOLUME 2: BEST PRACTICES FOR THE GEN II PWR, GEN II BWR L2PSAS – EXTENSION TO GEN III REACTORS

Volume 2 of ASAMPSA2 [2] concentrates on technical issues of L2PSA. It is the core section of the ASAMPSA2 documents with approximately 400 pages.

This very large size is due to the fact that all technical issues which influence core melt accidents and hence L2PSA are covered in the document. Moreover, the views and experiences of the different partners in ASAMPSA2 have been compiled. Therefore, the present summary can give only a tentative impression of volume 2.

1.2.1 DEVELOPMENT OF THE LEVEL 1 AND LEVEL 2 INTERFACE

Three basic principles are listed which have to be observed when building the interface from level 1 to level 2:

- Consistency: the endpoint of L1PSA and the initial point of L2PSA must be identical. There must be no gap between the two levels. (e.g. no undefined section of the accident sequence should exist between the RPV coolant level falling below a certain limit, and melting of fuel.)

- Completeness: the endpoint of L1PSA must contain all information needed for L2PSA. (e.g. issues belonging to the level 1 phase should be analysed completely within level 1, and not be analysed within the level 2 tasks.)

- Unbiased: the set of information defining the endpoint of L1PSA should not be biased. (e.g. the information should not be “pessimistic”, or “conservative”. It should rather reflect best estimate, including uncertainties if needed. This may sometimes be difficult to achieve since the L1PSA often contains inherent caution, like pessimistic minimal systems requirements for avoiding core melt); this principle is especially important when L2PSA is used for risk ranking approaches.

<u>Conclusion for ASAMPSA_E:</u> The general statments above are fully applicable also for extended PSAs. Nevertheless extension of L1PSA scope may add complexity in L1-L2 PSA interface. Practicable approaches should be discussed in ASAMPSA_E.

1.2.2 PROBABILISTIC ACCIDENT PROGRESSION MODELING – QUANTIFICATION AND PRESENTATION OF RESULTS

<u>Conclusion for ASAMPSA_E:</u>

The present section covers rather general topics of L2PSA. It is expected that most of the statements from ASAMPSA2 will be valid for ASAMPSA_E as well. The deliverable D40.3 (Proposal for ASAMPSA_E guidance) is suitable for discussing whether and how these issues can be adapted.

1.2.2.1 <u>Development of the accident progression event trees</u>

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

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

The events considered in the APET are of different nature:

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

The APET allows the description of severe accidents through L2PSA sequences, from a plant damage state (PDS), which consists of a grouping of L1PSA sequences (see 2.2.1) up to the plant final state after the accident, including possible containment failure. L2PSA sequences are grouped in turn into release categories (RC) which represent the most characteristic L2PSA result.

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

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

1.2.2.2 Realisation of the accident progression event trees

The accident progression event trees (APET) need to consider the different phenomena and containment failure modes in each phase along with potential mitigating functions and operator actions, including dependencies between phases, e.g. hydrogen burning in an early phase will lower the risk for hydrogen explosion in succeeding phases.

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

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

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

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

Depending on the probabilistic tools in-use, it may be possible to introduce the source term calculation in the APET. Such an approach is possible with tools like EVNTRE, SPSA and KANT. In that case, fast-running source terms codes are used as user function in the event tree.

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

1.2.2.3 Release Category Definition

Most L2PSAs finally aim at providing information about the release of radionuclides into the environment. In principle each individual sequence of an APET would lead to a specific type of release. However, even if it was possible to perform this detailed analysis, it is better to group sequences together which have similar releases. Sequence groups which have similar releases are called release categories. In a subsequent step a specific source term of radionuclides is associated to each release category.

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

A release category defines a grouping of APET end points, within which it is expected that the source term into the environment is similar. This grouping brings together accident sequences with similar fission product release and transport mechanisms.

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

1.2.2.4 APET quantification and results presentation

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

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

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

Tables correlating release categories with PDS or with initiating events provide useful evaluation tools and sources of information. Such tables can present absolute or relative values, i.e. with frequency or percent values. Examples of such result documentation are:

- Table which represents the relative contribution of each initiating event in each release category.
- Table which displays the conditional probabilities of each initiating event to enter each release category.

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

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

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

In addition there may be some simplified deterministic calculations of offsite doses, calculated in a manner similar to the offsite doses as calculated for Design Basis Accidents. If this approach is used, it is recommended that the IAEA guidelines for estimates of doses in safety demonstrations (see for instance among other IAEA safety series documents, IAEA-TECDOC 953/1997 [7] and IAEA-TECDOC-955/1997 [8]) should be followed. This approach is a surrogate for Level 3 PSAs, without the need to assess in detail the site-specific issues. Such results can be used to assess:

- Whether a release category falls above or below a pre-defined dose threshold for members of the public (compliance with safety limits),
- Ranking of release categories,
- Importance measures for systems, events and phenomena,
- Importance of SAM measures.

1.2.2.5 Consideration of uncertainties

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

There is a widespread notion that L2PSA issues are particularly uncertain. But the relative degree of uncertainty in L1PSA respectively in L2PSA is very much plant specific. There are, for example, containment designs which have a very high and rather certain conditional failure probability in core melt accidents – consequently for such reactors the uncertainty of the L2PSA analysis is small.

L2PSA projects occasionally refrain from explicitly performing uncertainty analyses because the required resources are considered substantial, and / or because the uncertainties seem to be not quantifiable. The second argument is not very convincing, if at the same time the analysis is presented with "best estimate" or "point values": how could one defend, for example, a point value for a containment failure probability of 10%, if no idea exists about the uncertainty range of that value?

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

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

1.2.3 HUMAN RELIABILITY ANALYSIS

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

In the context of L2PSA, HRA mainly focuses on the analysis of the severe accident management (SAM) actions taken into account in the accident scenarios. The analysis is similar to the analysis of post-initiator actions carried out in L1PSA, even though SAM actions have several features different from actions prescribed in Emergency Operating Procedures (EOPs).

Conclusion for ASAMPSA_E:

ASAMPSA_E addresses accident situations where human actions are particularly difficult to perform:

- Availability of necessary equipment may be limited in shutdown states
- After external events which are so severe that they cause accidents, the availability of plant staff may be limited and / or their stress is even more pronounced than after internal initiating events (e.g. recall the situation in Fukushima or imagine the plant area after an aircraft impact)
- For the accident conditions mentioned above rule-based actions will hardly be available, and knowledge based actions are difficult to evaluate.

Therefore, human reliability analysis (including an evaluation which human actions could be assumed at all) is a challenging issue in ASAMPSA_E.

1.2.3.1 Human actions in L2PSA

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

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

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

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

The main focus in PSA is on EOO's, while EOC's are generally considered out of the scope due to the difficulties to systematically and comprehensively identify EOC's. EOO's are typically modelled in PSAs because they are easily defined and limited by the requirements of the emergency operating procedures. The U.S.NRC HRA Good Practice document [9] recommends that EOCs should be addressed in future PSAs and as a minimum a search should be performed for conditions that make EOCs more likely.

1.2.3.2 Issues that must be carefully treated in HRA for L2PSA

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

- The dependency between L1PSA and L2PSA. Two dependency cases may be distinguished:
    - Operator actions common to L1PSA and L2PSA, e.g. recovery of core cooling. The difference is that in L2PSA there may be more symptoms to help diagnosis, and SAMG as well as crisis organisation may support the crew to perform the action. HRA in L1PSA and L2PSA should be performed in an integrated manner to consistently assess the issue for the level 2 action.
    - Operator actions relevant only in level 2 but that are dependent on the plant damage state (PDS). The key issue is to check whether the PDS provides enough information for accurate assessment of the context for the tasks. From HRA point of view, PDS definition may need to be refined to account for issues such as associated event sequences (failed safety functions), initiating events, equipment failures and previous operator errors. Dominant PDS-specific MCSs need to be analysed to assess the preconditions for level 2 operator actions.
- Procedures used in level 1 actions vs. level 2 actions. The operator actions modelled in L1PSA (before core damage), tend to follow EOPs, which are quite specific in terms of procedure steps and if-then logic statements. The operator actions modelled in L2PSA (after core damage) use SAMGs as a basis. SAMGs are usually more general in nature than EOPs. During the severe accident the decisions are based on the SAMGs and the training provided to all relevant staff for use of these guidelines and procedures. It should be considered that these procedures are more complex than during the normal operation or design basis accident and in general the symptoms are not exact. A gap may exist between EOP and SAMG and there may be potential operator reluctance to perform certain actions.
- The assessment of the impact of the action for the case where actions have both negative and positive influence (see SARNET benchmark on hydrogen, for example).
- Treatment of available time vs. failure probability. Many HRA methods use a time correlation curve to assign a base failure probability for a human action. This is better suited for operator actions with relatively short time windows for success, i.e. level 1 human errors. The use of time correlation curves at level 2 should be examined and considered carefully as the accident progression can have much longer time scales. One solution is to define a minimum human error probability for long time windows of several hours or even days.
- Treatment of uncertainties. All the above issues include uncertainties that need to be addressed, preferably in the same manner as other uncertainties in L2PSA. Reliability parameter uncertainties are handled in the parameter uncertainty analysis.

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

The principle distinction is that EOP should be applied step by step while SAMG can offer several options that should be chosen in relation with the crisis organisation. Each step in the guidelines is part of the overall process that should be used to reach a decision regarding the appropriate actions to take during a severe accident.

1.2.4 QUANTIFICATION OF PHYSICAL PHENOMENA AND CONTAINMENT LOADING

This chapter in [2] presents most of the physical phenomena that influence the severe accident progression and containment loading. The reader finds some phenomenological description of each related phenomenon and some recommendations on how it can be modeled in a L2PSA.

Conclusion for ASAMPSA_E:

In ASAMPSA_E the deliverables 40.4, 40.5 and 40.6 will address many of the issues mentioned in [2]. It will be one of the main tasks to identify where the common approaches are suitable, and where particular solutions are needed in an extended PSA. At present, it is already possible to indicate the following trends:

- The in-vessel accident progression will be similar for accidents in full power and in shutdown mode if the RPV head remains closed.
- If the RPV head is open in shut down modes, there are some additional phenomena which may be important:
  - The heat load from the open RPV (convection and radiation) may challenge sructures above the RPV (containment upper walls in PWR, reactor building roof in BWR).
  - The access of air to the melting core will change some chemical processes (hydrogen generation, aerosol formation).
- If an external event does not directly affect the interior of the containment, the existing analyses with internal initiating events will probably be applicable.
- The ASAMPSA2 documents do not address spent fuel pool accidents. This issue will be covered separately in D40.6

1.2.4.1 Important strategies for different purposes / End-users needs

The ASAMPSA2 end-user survey [10] identified 6 areas of L2PSA applications to be prioritised:

1. To gain insights into the progression of severe accidents and containment performance.
2. To identify plant-specific challenges and vulnerabilities of the containment to severe accidents.
3. To provide an input to determining whether quantitative safety criteria which typically relate to large release frequencies (LRF) and large early release frequencies (LERF) are met.
4. To identify major containment failure modes and their frequencies, including bypass sequences; and to estimate the corresponding frequency and magnitude of radionuclide releases.
5. To provide an input to the development of plant-specific accident management guidance and strategies.

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

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

<u>To gain insights into the progression of severe accidents and containment performance</u>

Gaining insights into the progression of severe accidents and containment performance requires that the level of detail of accident phenomena modelling is sufficient in the APET and includes some uncertaintiy analysis.

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

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

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

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

In that case, the L2PSA final results may not be realistic and the relative importance of identified risks can be biased.

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

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

- The plant design,
- The definition of “large release”.

If the plant design is extremely robust regarding both core damage prevention and severe accident consequences, the demonstration that some global LRF or LERF safety criteria are met can be done with some simplified (conservative) study of the accident phenomena. Nevertheless, even when the plant design is robust, if the regulator imposes a low level for the « large » release definition then some more detailed analysis of phenomena may be needed. The source term assessment may not be needed for this purpose if the large release definition is only based on the containment failure modes (assuming that the failure of any component that would increase the leak rate of the reactor containment building would lead to “large” release).

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

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

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

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

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 (or "risk reduction option"), L1PSA or L2PSA can be more or less crucial for the application.

It is important that the L1 and L2PSA scope and level of detail cover the risk reduction options being addressed. It should be verified that some specific L1PSA or L2PSA assumptions do not mask, decrease or increase the benefit of a plant modification.

1.2.4.2 Definition and calculation of representative sequences for each PDS

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

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

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

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

Some L2PSAs do not include the modeling of SAMG and human reliability. That allows a high simplification in the definition of the "representative sequences". However, it may be considered that such L2PSA studies are no longer state of the art.

1.2.4.3 In-vessel core degradation phase

This large section in [2] with approx. 80 pages covers all in-vessel phenomena which are relevant in a core melt accident. For each of the issues there is a description of the accident phenomena and recommendations how to address them in the L2PSA. The following sub-sections exist in [2]:

- core degradation
- induced RCS rupture including induced SGTR
- hydrogen production
- vessel cooling from outside
- consequences of in-vessel water injection (coolability, hydrogen production, RCS pressurisation, …)
- containment atmosphere composition and containment pressurisation
- containment venting
- hydrogen combustion
- in-vessel steam explosion and consequences

1.2.4.4 Vessel failure phase

This section in [2] with approx. 30 pages covers all phenomena which are relevant when the RPV fails. For each of the issues there is a description of the accident phenomena and recommendations how to address them in the L2PSA. The following sub-sections exist in [2]:

- RPV bottom failure due to thermal loads
- Issues related to high pressure RPV failure
- Fuel coolant interaction (FCI) and steam explosion

1.2.4.5 Ex-vessel phase (MCCI)

This large section in [2] with approx. 50 pages covers all ex-vessel phenomena which are relevant in a core melt accident. For each of the issues there is a description of the accident phenomena and recommendations how to address them in the L2PSA. The following sub-sections exist in [2]:

- Basemat lateral and axial erosion
- Impact of water injection after onset of MCCI
- Production of steam and incondensable gases
- Evolution of containment atmosphere composition and long-term pressurisation
- Containment venting in the ex-vessel-phase
- Pool scrubbing
- Melt propagation into ducts and channels

1.2.4.6 Corium recriticality and reactivity accidents

This section in [2] addresses the following issues:

- In-vessel corium recriticality
- Ex-vessel corium recriticality
- Core recriticality initiated by rapid heterogeneous boron dilution

1.2.5 CONTAINMENT PERFORMANCE

This chapter in [2] presents three topics which directly affect the containment function: failures in the containment itself, failure of the containment due to beyond-design internal loads, and failures in systems which render the containment ineffective.

Conclusion for ASAMPSA_E:

For ASAMPSA_E most of the statements in [2] will remain valid. However, the following issues are not covered in ASAMPSA2 and will need to be addressed in ASAMPSA_E.

The containment could be challenged by external events (e.g. mechanical impact, heat due to fire, pressure waves). It will be a wide field to be discussed within ASAMPSA_E how to address that in an extended PSA. Some of the issues are:

- Resilience of the containment (including penetrations) against seismic loads and missiles
- Reliability of containment isolation systems in external events
- Vulnerability of electric containment penetrations under heat load (external fire)

1.2.5.1 Initial containment performance: pre-existing leakage and failure of isolation systems

This section in [2] addresses two different issues: First, the question about the possibility and probability that the containment leak rate is higher than designed. Second, the probability that active isolation systems (valves, flaps) fail, so that the containment function is lost.

1.2.5.2 Quasi-static loading / dynamic loading - structural response, structural analysis, fragility curve

According to [2], the timing and the way in which the containment fails is a crucial factor influencing the magnitude of the off-site consequences. Containment design criteria are usually based on a set of deterministically selected load scenarios. For example, pressure and temperature challenges are usually based on the design basis LOCA. External events such as earthquakes, floods, high winds or aircraft crash are considered as well in some cases. Assessment of beyond design accident sequences show that, in some cases, significant containment loading can occur, reaching or exceeding the design loads. Therefore, transient loads like fluid jet impingement, direct containment heating, rapid deflagration or detonation of hydrogen pockets which may occur during severe core degradation accidents, may pose significant threats to containment integrity.

The assessment of such threats is a key element in L2PSA. The loads to the containment are determined in the respective sections on in-vessel and ex-vessel phenomena, and are not repeated here.

1.2.5.3 Containment bypass

According to [2], the following issues have to be addressed with regard to containment bypass:

a) Which systems are potential candidates for generating a containment bypass event?

b) What are the consequences of containment bypass?

c) What is the probability that containment bypass occurs?

d) What is the quantity of radionuclide release (source term) through a containment bypass? For this issue it is important to distinguish whether the bypass flow is directed immediately into the environment or whether it is directed into an intermediate room inside the plant.

1.2.6 SYSTEM BEHAVIOUR UNDER SEVERE ACCIDENT CONDITIONS

This section in [2] discusses the use of systems in severe accidents. There are two different approaches for severe accident management for operating power plants: use of non-dedicated systems and use of dedicated systems. Finally, specific systems such as the core catcher, the filtered venting system and some passive systems are described.

<u>Conclusion for ASAMPSA_E:</u>

In ASAMPSA_E the boundary conditions for systems are different (due to impact of external events, or unavailability in shutdown modes). Therefore, it is advisable to address this issue in ASAMPSA_E.

Some of the associated topics are related to L1PSA (systems unavailability due to the initiating event) and some to L2PSA (impact of accident conditions on systems). Therefore, an appropriate work distribution is needed within ASAMPSA_E. For the level 2 part, the deliverable D40.5 (SAMG validation by PSAL2) is a suitable framework.

1.2.7 SOURCE TERM ASSESSMENT

The source term assessment provides information about the characteristics of the release categories in terms of composition of the release and the time of release. The source term is combined with release category frequencies in the presentation of results. Depending on the scope of the PSA, source terms can be simple, e.g. above or below a certain threshold of released quantity, or sophisticated, e.g. time-dependent release rates of different isotopes for further processing in a PSA level 3.

The source term calculations are usually based on the choice of one or more sequences to be representative for each release category. If there is more than one source term calculation for a single release category, this information can be used to both support the definition of release categories and the assignment of release categories to different release sequences in the APET.

The source term assessment process includes the following steps:

- Choice of representative severe accident sequences within each release category,
- Identify the needs of the source term characterisation due to L2PSA objectives and criteria.
- Calculation of source terms for the representative severe accident sequences.

In [2] there is also a section introducing the concept of allocating a hazard index to source terms using a simplified off-site dose assessment methodology. This issue is sometimes considered within a so-called level 2+ PSA.

<u>Conclusion for ASAMPSA_E:</u>

The section on source term assessment in the ASAMPSA2 guideline is a rather general methodological discussion. As such, it is in principle valid also in the ASAMPSA_E framework. It does not seem necessary to address the methodological issue again in ASAMPSA_E. However, the source term mainly depends on the accident sequence, and they may need to be revisited for external events, for multi-unit accidents and for fuel storage accidents. Consideration of non-radioactive but chemical or toxical source terms in case of an accident can also be a topic of interest in ASAMPSA_E.

1.3 VOLUME 3: EXTENSION TO GEN IV REACTORS

The GIF forum selected six reactor designs as being of special interest in the context of next generation reactors. Of those six projects the ASAMPSA2 project considered four:

1. The Sodium Fast Reactor (referred to as SFR below),
2. The Gas Fast Reactor (GFR),
3. The Lead Fast Reactor (LFR),
4. The Very High Temperature Reactor (VHTR).

The molten salt reactor and the supercritical water reactor were not considered as too few data were available to the ASAMPSA2 participants.

The compliance of those concepts with the GIF main objectives are recalled hereafter:

- The SFR, GFR and LFR systems (i.e. those featuring a fast neutron spectrum) are top-ranked in sustainability because of their closed fuel cycle and excellent potential for actinide management, including resource extension; they are also rated good in safety, economics, and proliferation resistance and physical protection;
- SFR is primarily envisioned in electricity production and actinide management; the SFR system is the nearest term actinide management system; based on the experience with oxide fuel,
- GFR is primarily envisaged in electricity production and actinide management, although it may also support hydrogen production; given its R&D needs for fuel, the GFR is estimated to be deployable by 2040;
- The LFR system is specifically designed for distributed generation of electricity and other energy products and for actinide management, given its R&D needs for fuel, materials, and corrosion control, the LFR system is estimated to be deployable by 2025;
- The VHTR addresses advanced concepts for helium-cooled, graphite moderated thermal neutron spectrum reactors with a core outlet temperature higher than 900°C.

Four representative concepts have been selected as a basis for the work to have some clear data to base the discussion on:

- The EFR (European Fast Reactor) concept for SFR; an European engineering consortium (EFRA) developed the EFR project on behalf of a European utility consortium (EFRUG) from 1988 to 1998, aiming at pooling the experience and resource of several European design and construction companies, R&D organisations and electrical utilities.

- The CEA 2400MWth GFR design (as designed at the end of 2007) for GFR;
- The ELSY project for LFR; a European lead-cooled fast reactor developed in the framework of EU FP6,
- ANTARES project, a commercial project designed by the AREVA company, for VHTR.

1.3.1 MAIN FEATURES OF THE GEN IV REPRESENTATIVE CONCEPTS

In this section of [3], the main features regarding the core and the circuits of the representative Generation IV reactors are provided. Then, a review of the specific core degradation mechanisms to be accounted for in these various concepts is provided, with the final objective to assert the similarities between LWRs and Gen IV reactors for L2PSA model building.

In order to mitigate the consequences of a sSevere accident, several provisions of different natures and related to these specific risks are intended to be implemented in these Generation IV concepts. A paragraph is also devoted to the review of the provisions made to mitigate the consequences of severe accidents.

Finally some specific issues regarding the fission products chemistry and phenomenological trends are exhibited for the source term assessment.

1.3.1.1 Specific degradation mechanisms and damage criteria related to these concepts

Based on expanded (with respect to LWR) failure modes to describe the associated risks related to the four selected concepts, this section of [3] provides elements regarding:

- The “key parameters” and related phenomena associated with core degradation mechanisms;
- The containment building features and potential specific degradation processes;
- The specific provisions implemented in order to reduce the consequences of severe accidents;
- The main elements regarding fission product thermo-chemistry for the source term assessment.

1.3.1.2 Key parameters related to core degradation mechanisms

Even if a L2PSA is focusing on the containment challenges and the assessment of the FPs release into the surrounding of the plant, it is worth noticing that specific core degradation mechanisms could be involved in the selected Generation IV representative concepts compared to LWRs.

The key parameter related to core degradation mechanism are:

- Material inventories,
- Core materials behaviour at high temperature (melting / slumping / sublimation) and potential interactions of fuel/cladding with primary coolant or foreign fluids (water/air/...),
- Interactions between the primary coolant and others fluids,
- Possibility of a core disruptive accident,

- Core criticality concern due to for instance foreign fluid ingress in the fissile region, or to coolant voiding effects.

1.3.1.3 Compliance and potential transposition of containment degradation modes

In this section, equivalence with the usual terminology for LWR for the representative containment failure mechanisms is proposed for theses new reactors.

1.3.1.4 Specific provisions for prevention and mitigation of severe accident consequences

In both PWRs and BWRs, several provisions are used in order to limit the consequences of severe accidents (for instance: the containment spray system - which is aimed at reducing the containment pressure and removing the decay heat and also to enhance the FPs aerosols deposition in the containment building - and the hydrogen control by the use of igniters or catalytic recombiners).

For Generation IV reactors, different “devices” are specifically engineered for prevention of SAs. They can be classified as :

- A supplementary shutdown system,
- a specific design of core assembly to promote the corium spreading and local recovery of a cooling path,
- a core catcher,
- some containment’s engineered safety features,
- means / systems of ultimate heat sink,
- a severe accident management strategy.

1.3.1.5 Important parameters for L2PSA related to the source term evaluation

In the source term analysis, the following issues are considered to be of major importance according to LWRs phenomenological trends:

- Inventory of radioactive materials in the fissile region (at EOL) according to the nature of fuel, maximum burn-up, minor actinides inventory;
- In-vessel radionuclide release and transport mechanisms (related to fuel/cladding and coolant natures);
- Retention and deposition of fission products inside RCS;
- Chemical species (e.g. organic or non-organic iodine…)
    - Iodine and caesium chemistry (Affinity of these isotopes with coolants involved);
    - Chemistry of other isotopes (Te, Sr…): knowledge regarding phenomenological trends in presence of helium, lead or sodium;
- Activation and corrosion products;

- Ex-vessel radionuclide release and transport (related to containment type);
- Aerosols behaviour inside the containment;
    - Deposition and re-suspension of aerosols mechanisms;
    - Effect of energetic phenomena on in-containment fission product behaviour;
    - Activation and corrosion products of concrete surrounding the core vessel (if any) and air of its cooling system;
- Radionuclide release outside the containment (i.e. source term);
- Tritium;
- Potential for FPs scrubbing;
- Additional barriers or structures (retention tanks, close-containment) for radionuclides (e.g. close-containment for GFR is not considered as a confinement barrier but could lead to a potential of FPs retention).

1.3.1.6 Treatment of hazards

In this section of [3] a number of hazards that can cause loss of containment integrity are listed.

Two of those hazards are of prime interest because of their wide-spread effect. They are generally quantified through a dedicated level 1 PSA:

- The internal fire: in addition with potential induced failures of components, systems or electrical supplies, the fire event can potentially cause a containment isolation valve to fail to close (i.e. β-mode) or a slow heat-up of the containment atmosphere. On behalf of these considerations that are proto-typical of all engineering processes, a fire induced by a sodium or graphite interaction with air or water is of major importance for Generation IV concepts owing to the induced effect on the primary vessel and on the containment. At this stage, it is worth noticing that a fire event (as internal hazard, or caused by coolant interaction with another fluid) has also a direct influence on the source term through the chemical form of FP species that could be formed in such a hot atmosphere.
- The seismic event: an earthquake can potentially cause structural failure of the containment or its penetrations. Earthquake is also a great concern for fast neutron spectrum reactors in particular due to the core compaction and the resulting criticality risks.

Hazards are generally classified under “internal hazard”, meaning originating from the plant, as for instance a damaged turbine fan acting as a missile, and “external hazard”, meaning originating from outside the plan like an airplane crash. The risks involved could be radiological as well as of chemical or toxical nature.

A list of the representative hazards for Generation IV reactors is provided (for both internal and external hazards):

- SFR :
    - Internal hazards : fire, hydrogen explosion
    - External hazards: earthquakes, exceptional meteorological conditions, inland or marine flooding, aircraft crash, gas cloud explosion, turbo-alternator missiles, lightning discharge

- Particular case: water/sodium/air reaction in the steam generator building:
- GFR : Even if GFR studies are up-to-now non-site specific, several hazards are particularly feared owing to the potential they could have for a potential core disruptive accident (e.g. earthquake) and to the multiple failures of required systems they could engender. As a basis for further studies, one should envision seismic events, fire in the containment building, and flooding (potentially caused by a climatic event).
- LFR :
  - Internal event : flood and fire.
  - External hazards : earthquakes, high winds, hurricanes, aircraft impacts, and external flooding
- VHTR : a comparison between hazards for LWR and VHTR is made. Two hazards have a higher risk in VHTR : hot and cold gas release, and vibrations. There is one additional hazard not existing in LWR: Core / fuel chemical hazards, including dust explosions and combustible gas generation (H_2 and CO_2).

Conclusion for ASAMPSA_E:

The representative hazards described above should be taken into account.

The risks involved in an extended PSA should be also of radiological as well as of chemical or toxical nature. In consequence some elements regarding chemical risk are provided in an appropriate section in ASAMPSA2. The coupling between both impacts should be discussed (a toxical impact on a site may influence the NPP management).

1.3.1.7 Specifics related to shutdown or refuelling states

All plant operation states should be addressed so that L2PSA results will be balanced. In consequence, this section in ASAMPSA2 exhibits the peculiarities of the selected reactors concepts as regards to shutdown or refuelling states.

1.3.1.8 Review of existing L2PSA applied to SFR, LFR, HTR or GFR

This section of [3] provides a review of exisiting L2PSA applied to representative Generation IV reactors

SFR: To our knowledge, no L2PSA was performed for the EFR concept. However, it should be underlined that several probabilistic studies were performed in the past for the US PRISM concept, for the SNR-300 reactor and recently for the JSFR.

GFR: Regarding the safety approach, a large body of both analytical and experimental safety R&D under the LMFBR safety program was assumed to be directly applicable or easily adapted to GCFR fuel rod behaviour. However, the absence of two-phase coolant flow effects reduced the complexity of the accident analysis tasks and accident behaviour evaluation. Regarding Hypothetical Core Disruptive Accident (HCDA), three classes have been identified which have slightly different phenomenology: ULOF (Unprotected Loss of Flow - either initiated by a loss of helium circulation capability or by depressurisation), UTOP (Unprotected Transient of Power) and LDHR (Loss of Decay Heat Removal). Given the occurrence of a HCDA, the reactor vessel provides a first line of protection against FPs release to the environment. In these former GFR concepts (see appendix C in [3]), the PCRV assures the containment integrity by preventing the molten core penetration and the formation of missiles due to accident energetic or overheating failures. For containment integrity, one design requirement for the containment is that it could be able to withstand the overall helium volume and the energy egress associated with the Design Basis Depressurization Accident (DBDA).

LFR: At present, there are no studies available pertaining to L2PSA for LFR

VHTR: Evidences exist that PSA studies were formerly conducted for:

1) the American HTGR project by General Atomics in 1978,
2) the German HTR-1160 around 1979 (several internal reports have been issued by the Jülich research center);
3) the American MHTGR project around 1995;
4) the PBMR in South Africa.

The PSA for the US MHTGR is publicly available. Otherwise, only a limited set of articles has been retrieved.

1.3.2 EXISTING TOOLS FOR ACCIDENT ANALYSES

1.3.2.1 Extent of the knowledge and potential limitations in the modeling of severe accidents

For LWRs, L2PSA models were built in combination with an important R&D effort, including experiments and code development (mechanistic codes and also integrated tools), validation processes of physical models and finally uncertainty assessment for risk quantification. The situation is slightly different for other reactor concepts. This section of [3] aims at providing an overview of the pertaining difficulties that could be encountered for L2PSA model building in the frame of the Generation IV reactors.

SFR: Regarding the extent of knowledge and the modelling level, it seems essential to distinguish the following situations:

- The leading phenomena are those ensuring an adequate core cooling in protected or unprotected situations (i.e. with or without reactor scram). In particular, reactor cooling to maintain fuel cladding integrity is of major importance. Investigations of transient-overpower events have also been an important area of investigation. For the less severe overpower transients, significant data from transient tests in EBR-II could provide significant confidence in the ability to model fuel performance and the consequences of failure (e.g. such tests included transients on fuel with breached cladding to determine the potential for fuel loss to the coolant). Reactivity effects of mechanical changes in core structure, sodium density effects and changes in fuel structure have also been extensively studied starting with the investigation of fuel pin bowing in the American EBR (Experimental Breeder Reactor).
- The severe overpower and under-cooling transients received a great deal of attention in the 70s and 80s, especially given the potential for leading to core disruptive accidents (e.g. through the results of

Transient Reactor Test facility (TREAT) and other similar facilities). For such transients, analysis uncertainties are high because of the event complexity, the rapidity with which numerous different phenomena occur, and the difficulties of performing and instrument experiments. Analyses of phenomena evolving during a HCDA are also complex, particularly due to the potential corium criticality (linked to the geometry).

- Another difficulty in modelling severe accident progression is the lack of hazards studies, e.g. earthquakes impact. There is also a lack in experience feedback for developing reliability or performance data with a good confidence level, particularly for L2PSA. Globally, in fast breeders, it is easier to improve prevention than mitigation.
- For transition phase (extended meltdown in the core region before fuel relocates downwards): A large document has been issued in the mid-90s as a common work between researchers involved in the SFR transition phase studies in Europe. A part of the document consists of a broad survey of the state of the knowledge and evaluation of future needs. A short summary of the main shortcomings identified at the time is also provided, in order to give a clearer view on what has yet to be understood through experiments, modelling and implementation in codes.
- For source term assessment: A PIRT is currently taking place in the US and should bring an up-to-date overview of the present state of knowledge.

GFR: As for SFR, one should first distinguish mechanisms for accidental insertion of reactivity from those leading to reactivity effects due to the loss of fuel integrity. Insertion of reactivity could occur due to Control Rods Assemblies withdrawal, large water ingress or core radial compaction. The loss of fuel geometry is mostly related to the decrease of decay heat removal performance, then leading to core materials melting and slumping in the fissile region. As a result, several phenomena should be accounted for in calculational tools for consequence assessment. Compared to LMFBRs or LWRs, a substantial lack of feedback and experimental validation is associated with dedicated tools for core damage progression assessment.

LFR: Main limitations are related to the analysis tools (i.e. integral codes), since the analyst should be aware of the phenomena addressed in the code and their modelling approach and limitations. The user should have a sound knowledge of the strengths and weaknesses of the code, which should not be used out of the range of situations and conditions for which it has been designed. It should be noted that any limitations in the level 1 PSA will be carried forward into the L2PSA. This will need to be taken into account in the intended uses and applications of the L2PSA. If the L2PSA has been based on a L1PSA that has a lower scope than this, these limitations need to be taken into account in the application of the L2PSA. Main limitations related to PSA concern the uncertainties (i.e. parameter, modelling accuracy and completeness) and the time treatment which considers the chronology of events instead of actual timing: this implies the consideration for dynamic event trees.

VHTR: A full-scope identification and ranking of the phenomena involved in the VHTR concepts by an expert panel has been conducted recently in the US. For what is related to fuel particles behaviour, there remain quite a lot of uncertainties. Moreover, some areas where it's thought that more knowledge is needed are connected with the design (for instance evaluation of the core by-pass flow). The phenomena knowledge has usually been ranked as medium on a 3-level scale and only three times scored at low level for the following phenomena:

- reactor vessel cavity air circulation and heat transfer;
- reactor cavity cooling system with "gray gas" in cavity;
- cooling flow restarts during loss of forced circulation.

Conclusion for ASAMPSA_E: This part is not relevant for ASAMPSA_E which does not cover these types of reactor.

1.3.2.2 Existing and available tools

This section of [3] details the limitations of the available codes with respect to their applicability for L2PSA consequence assessment (i.e. potential large CET, discrepancy of scenarios and phenomena, assessment of the containment response in addition to the calculation of FPs release in the environment).

In compliance with L2PSA scenario quantification for LWRs, a combination of mechanistic, integrated (e.g. ASTEC, MELCOR or MAAP developed for LWRs) and simplified tools appear suitable for consequence assessment. The mechanistic integrated codes are used for detailed analyses (e.g. for the definition and interpretation of experimental tests) while the integrated tools are able to represent the whole accident scenario and predominant phenomena (starting from the core degradation and including the containment response modelling and the evaluation of FPs release outside the containment).

Then, a major concern for tools that would be used for the accident propagation and consequence assessments of Gen IV concepts is related to the necessity to handle the foreseeable and potentially large uncertainties in scenario depiction, time frame of the core degradation and FPs release and propagate them in tools devoted to physics, as in the quantification tool. This point is generally devoted to the code systems including modules for scenario, phenomena and potential cliff-edge or "branching" effects.

To date, Gen IV reactors mostly rely on code developed for LWRs or LMFRs (SFRs and LFRs) and adapted to other fluids and materials. The LMFRs concepts take benefit from validated codes that were developed in several countries during the 70-80s (US, Japan or France for SFR, Russia for LFR).

Conclusion for ASAMPSA_E: This part is not relevant for ASAMPSA_E which does not cover these types of reactor.

1.3.3 SCREENING OF THE COMPLIANCE WITH L2PSA GUIDELINES OF LWR

This section provides examination of compliance with methodology guidance for a L2PSA building, as regards to the design phase of these reactors, regarding the design features and the main phenomena involved in the selected Gen IV representative concepts.

1.3.3.1 Compliance with PWR phenomena and systems for L2PSA building

According to LWR state-of-the-art accident progression depiction in L2PSA models, questions for the accident progression could be described (and modelled in the L2PSA model's event trees) according to the following mechanisms:

- In-vessel core degradation
- Vessel rupture phase
- Ex-vessel phase (MCCI)
- Containment performance (tightness)
- Systems behaviour in severe accident conditions

A compliance table has been built up, with 5 values of "compliance levels" ranging from "1" which means that recommendation are highly compliant with the phenomena involved for Gen IV concepts, to "5" that is signifying that no compliance could be exhibited. To sum up, the phenomena expected in the primary circuit can be completely different from LWR ones, because of the presence of coolants (liquid or gaseous, i.e. with or without potential phase change), systems and phenomena of different nature.

Physical phenomena involved in a severe accident transient occurring on a LWR concept are mostly non relevant to VHTR technology. Similarities may be pointed out for some phenomena but particulars are generally so far away that they require to be studied in very different ways. Subjects connected with human factors and source term assessment are of general interest and LWR may provide interesting information. Otherwise, all which is connected with APET is of interest.

1.3.3.2 L2PSA structure

The commonly-adopted approach for a L2PSA (performed after L1PSA) is:

- Definition of the initial conditions by binning of L1PSA end states into Plant Damage States (PDS);
- Development, construction and quantification of event trees: Containment Event Trees (CET, i.e. small event trees) or Accident Progression Event Trees (APET, i.e. large event trees);
- Definition of source term categories or release categories;
- Binning of containment states related to specific containment failure modes (thanks to the determination and the evaluation of containment failure modes).

This approach is universally applicable to the Gen IV reactors.

1.3.3.3 Human Reliability Assessment

As demonstrated by a number of PSAs, both qualitatively and quantitatively, human actions play a very important role in the safe operation of current Nuclear Power Plants (NPPs). Therefore Human Reliability Analysis (HRA) becomes an extremely important task for the realistic assessment of the plant safety in PSAs. Unfortunately, human reliability is a very complex subject, which cannot be addressed by fairly straightforward reliability models like those used for components and systems. So, even if uncertainties still exists in some areas, the described methods well represent the situations in which the operators are to perform preventive accident management actions.

This is not generally true for actions that can be effective in the mitigation of severe accidents; such actions are not always clearly addressed in the Emergency Procedures Guidelines or in the Emergency Operating Procedures.

For Gen IV reactors, Emergency Operating Procedures are neither defined nor developed and validated by interviewing and observing control room personnel performance when challenged by events potentially leading to plant damage states.

1.3.3.4 Quantification of physical phenomena and uncertainties

Among the several sources of uncertainties for L2PSA, one should distinguish:

- Parameters (data) uncertainties;
- Model uncertainties (i.e. associated with phenomenological models for the physical-chemical processes and related assumptions);
- Model completeness uncertainties (even if such uncertainties cannot be quantified within a given PSA scope, but by performing additional analyses of excluded events to demonstrate their insignificance);

1.3.3.5 Passive Safety systems

Uncertainties regarding the performance of safety systems will constitute a new challenge owing to the fact that several Gen IV designs employ passive safety characteristics and passive safety systems to a much greater extent than current nuclear facilities. The failure assessments of passive components or systems require a complex combination of physical and human factor ingredients. This poses an issue for methodology because there is less experience in modelling passive systems compared to active systems. Moreover, system-specific operating data are sparse and may not provide statistically useful information. In LWRs, this aspect was not a major concern and the assessment of the reliability or performance of passive systems was mainly “deterministic”. However, for Gen IV reactors that should rely more on passive systems, a deterministic demonstration would lead to substantial conservatisms. Therefore, in a constant evolution of modelling and safety improvement, it should be foreseen that probabilistic assessment through uncertainties propagation (Monte-Carlo sampling, …) would help the L2PSA quantification.

1.3.3.6 Calculation tools and Uncertainties

It is worth recalling that uncertainties may also be relative to the extent of knowledge based on experimental results, on code development techniques (and unavoidable simplifications they will handle) and finally to their validation matrices or crossed comparisons (i.e. benchmarking). This point seems to be a major drawback for Gen IV reactors (and related L2PSA models) compared to LWR ones.

1.3.3.7 Role and extent of expert judgment

Expert judgement plays an important role in assessing the progress and probabilities of events in a L2PSA. This is even more relevant for Gen IV reactors due to the limited experience in comparison with LWRs.

2 SUMMARY ON THE IMPACT AND EXPERIENCE FEEDBACK OF THE ASAMPSA2 DOCUMENTATION

2.1 SUMMARY OF CONCLUSIONS FROM ASAMPSA2 DOCUMENTS FOR ASAMPSA_E

The following conclusions basically are a summary of the most significant statements given in section 1 of the present document.

2.1.1 CONCLUSION FROM ASAMPSA2 VOLUME 1 - GENERAL

The ASAMPSA2 guidelines are a huge document. This is due to the wish to work at a technical level. The ASAMPSA_E project also has a technical objective, and therefore should lead to a large documentation as well.
For ASAMPSA_E it seems reasonable to propose like in ASAMPSA2 a set of acceptable existing solutions for all issues to be addressed without trying to propose a precise step-by-step procedure.
The ASAMPSA_E project has much more internal structures than the previous ASAMPSA2, and in contrast to ASAMPSA2, specific documents are foreseen for particular topics. Therefore, although the total volume of documentation in ASAMPSA_E is expected to be large as well, it will be better structured and more easiliy accessible.

The common three levels of PSA are also a practical concept for extended PSA. There is no indication that the common level definitions are not adequate. The expression “core damage” should be replaced by “fuel damage” where appropriate in order to take into account accidents in spent fuel pools.

The principal structure of a L2PSA is applicable also for those issues which have to be taken into account in an extended PSA, even if few additional issues have to be considered (fuel storage containment threats from outside, multi-units accidents …).

Regarding the current situation of L2PSA on international and national level, there are several references which address L2PSA in general, without detailing the operation mode or the type of events. Their relevance for ASAMPSA_E should be checked. In addition, there are few documents which explicitly mention shutdown states of the reactor or external events – these documents are certainly relevant in ASAMPSA_E. However, there is almost no information about spent fuel pool accidents in ASAMPSA2.

With the perspective of updated WENRA reference levels after the Fukushima accident, the link between the DECs (design extension conditions) and extended PSA concepts should be discussed in detail by the ASAMPSA_E partners.

For the ASAMPSA_E project it can be assumed that probably all statements in ASAMPSA2 related to risk measures general topic are valid. However, there was no complete consensus in the ASAMPSA2 community with regard to a common risk target. In ASAMPSA_E , there is a possibility to address this issue and to verify or modify the views given in ASAMPSA2.

The chapter on PSA applications in ASAMPSA2 tentatively discusses the possible applications of the L2PSA. For ASAMPSA_E, many sections are of interest. However, the focus in ASAMPSA_E will have to be defined in the upcoming end-users survey.

In ASAMPSA2 a short list of issues on shutdown states has been developed, which should be considered in extended L2PSA. It will be a starting point for ASAMPSA_E.

Since ASAMPSA_E will have its own interaction with end-users, the results of the ASAMPSA2 survey may not be applied directly. However, the experience in performing the survey should be transferred to the survey to be done within ASAMPSA_E in WP10.

2.1.2 CONCLUSIONS FROM ASAMPSA2 VOLUME 2: BEST PRACTICES FOR THE GEN II AND GEN III REACTORS

Volume 2 of ASAMPSA2 [2] concentrates on technical issues of L2PSA. It is the core section of the ASAMPSA2 documents with approximately 400 pages.

The sections on level 1 – level 2 – interface and on the probabilistic accident progression modeling in ASAMPSA2 cover rather general topics of L2PSA. It is expected that most of the statements from ASAMPSA2 will be valid for ASAMPSA_E as well. Nevertheless extension of PSA scope may add complexity in L1-L2 PSA interface. Practicable approaches may be discussed in ASAMPSA_E.

ASAMPSA_E addresses accident situations where human actions are particularly difficult to perform. Therefore, the traditional approach as discussed in the ASAMPSA2 documents will have to be evaluated critically in ASAMPSA_E.

In ASAMPSA_E three deliverables will address the quantification of various physical phenomena. It will be one of the main tasks to identify where the common approaches provided in ASAMPSA2 are suitable, and where particular solutions are needed in an extended PSA.

The containment could be challenged by external events (e.g. mechanical impact, heat due to fire, pressure waves). It will be a wide field to be discussed within ASASMPSA_E how to address that in an extended PSA.

In ASAMPSA_E the boundary conditions for systems will often be different from internal events, due to impact of external events, or due to unavailability in shut down modes. Therefore, this impact on systems behavior will be an issue in ASAMPSA_E.

The section on source term assessment in the ASAMPSA2 guideline is rather general. Therefore, it is in principle valid also in the ASAMPSA_E framework.

2.1.3 CONCLUSIONS FROM ASAMPSA2 VOLUME 3: EXTENSION TO GEN IV

ASAMPSA_E does not include the Gen IV reactor concepts. Nevertheless the following topics may be useful for ASAMPSA_E:

- The internal and external hazards that are taken into account in the design of the Gen IV concepts,
- The risks involved in an extended PSA should be of radiological as well as of chemical or toxical nature. The coupling between both impacts should be discussed (a toxical impact on a site may influence the NPP management).

A review of Gen IV advanced reactor concepts with focus on L2PSA shows a number of technical issues and gaps to be resolved:

- Severe accident evaluation
- Accident phenomenology studies
- Fuel behaviour studies, e.g. relocation and interaction with materials
- Analysis of risk of core compaction and power excursion due to re-criticality
- Identification of confinement building weaknesses and damage modes and possibility of by-pass for each concept
- Establishment of mechanistic source terms and source term assessment
- Need for each concept to consider extreme external events

The overall conclusions from the analysis of both core degradation phenomenology and general PSA methodology (like interface between level 1 and level 2, human risk assessment, system modeling and the role of expert opinion) is that the methodology itself is not very much affected by the reactor type contrary to what is related to the accident phenomena.

2.2 IMPACT OF ASAMPSA2 ON THE PARTICIPANTS

The large number of participants and the wide field of topics has been a challenge for the project management, and also for the individual partners who had to fit into work plans, time schedules and budget constraints. The necessity to produce a common product fostered exchange between the partners and created working groups. This networking is certainly beneficial for the partners and will improve further information exchange and potential cooperation.

Many sections of the guideline have been drafted by more than one partner. The necessary exchange of views and the discussions which sometimes evolved in order to achieve the common draft improved the partners' knowledge.

Almost all sections of the ASAMPSA2 documents have been endorsed unanimously. However, a few issues remained controversial. This has been mentioned as appropriate in the documents. For the partners with a minority opinion it is certainly satisfactory to see their views expressed in this way.

Some of the participants originate from countries with a well-established L2PSA culture, including national rules or guidelines in this field. It is unrealistic to expect that those countries and their responsible authorities will dismiss their established rules and adopt the ASAMPSA2 guideline instead. However, when these national guidelines will eventually be updated, the groups in charge will very likely make sure that their new rules at least do not contradict the ASAMPSA2 documents. In this sense there may be some harmonization effect.

Participants from countries with less experience in L2PSA will be able to benefit from the ASAMPSA2 knowledge transfer during the project and from the knowledge contained in the final documentation. But one should be aware that also these countries have access to international guides (IAEA, OECD) and to published national guidelines (e.g. from Germany or the US) and to their vendor's documents. At present, short after the official publication of ASAMPSA2 volumes, it is not possible to estimate their impact on the national situations.

2.3 IMPACT OF ASAMPSA2 ON THE PSA COMMUNITY IN GENERAL

The final ASAMPSA2 documentation on the ASAMPSA2 website has the delivery date 2013-04-30. Only few months have passed before writing the present report – therefore not much feedback is possible outside the ASAMPSA2 community yet.

Nevertheless, the following impact can be expected:

- The large body of information which is compiled in the ASAMPSA2 documents can almost be understood as a complete summary of the present state of knowledge – including many practical examples, but also with remarks about still existing knowledge gaps. Therefore, the ASAMPSA2 documents might become some type of reference for young professionals, or for organizations with limited resources in the L2PSA field.

- A large number of L2PSA have been performed, but only a fraction of it is available outside the narrow circle of producers and reviewers and authorities. Certainly, valuable information exists also in those PSA which are not published, but due to a policy of keeping many L2PSA secret, they cannot develop beneficial influence. In contrast, the ASAMPSA2 documentation is publicly accessible. This is a good basis for impacting many future developments.
- PSA producers and / or authorities with little experience in the field typically take international guidelines as a reference. The ASAMPSA2 guideline can be a complementary reference to the well-known IAEA documents. Since the ASAMPSA2 volumes contain more practical examples and provide more background they may become a valuable help for practitioners.
- Volume 3 (Extension to Gen IV reactors) is one of the very rare documents which explicitly address and compile L2PSA for new reactor types. While this novel compilation does not meet the standards of a practically usable PSA guideline, it makes it evident that some of the fundamental approaches in severe accident PSA do not necessarily apply to innovative designs. Therefore, it is certainly of interest for the developers of Gen IV reactors and for the prospective licensing authorities in order to prepare an adapted PSA approach for future reactors.
- There has been communication between the ASAMPSA2 project and the so-called “end-users” of the L2PSA in a so-called “end-user survey” and in a final workshop. This communication increases the awareness that ASAMPSA2 and the related documents may be beneficial.

Conclusion for ASAMPSA_E: Participation in many international collaboration activities (OECD, IAEA, FISA, EUROSAFE, …) helped the ASAMPSA2 communication, and the ASAMPSA2 effort was well known by the PSA international community.

The ASAMPSA_E should benefit from the same context. The difference with ASAMPSA2 is that extended PSA is an emerging concept and is not associated to a state of art methodology. The day-to-day communication on ASAMPSA_E should contribute to the publicity on the project but also induce some stakeholders to extend their PSAs.

2.4 TECHNICAL AND SCIENTIFIC ACHIEVEMENTS OF ASAMPSA2 OF INTEREST IN ASAMPSA_E

The ASAMPSA2 documentation is a very comprehensive compilation of practically all aspects which are relevant in a L2PSA for full power states and internal initiating events. Therefore, it can be used as a valuable source of information. This will also apply for ASAMPSA_E which may adapt the general structure of addressing topics. It will probably be very helpful to have such a structure available when setting up work plans and tables of contents.

During ASAMPSA_E it will also be interesting to compare the achievements of ASAMPSA_E with the previous ASAMPSA2 statements.

On the other hand, ASAMPSA2 addressed topics which are rather settled in the community and it did not have the ambition to create new solutions. This is different in ASAMPSA_E which will try to discuss partly novel challenges where practical solutions may not yet exist or at least not yet be considered state of the art. ASAMPSA2 will constitute some type of background for this process, but it cannot provide a sound basis for such tasks.

3 CONCLUSIONS AND LESSONS LEARNED FROM ASAMPSA2

3.1 L2PSA HARMONIZATION

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

The main contribution of ASAMPSA2 should therefore be the reduction of the lack of consistency between existing practices on L2PSA in the European countries.

Originally, there has been the idea to define two approaches within the ASAMPSA2 project: one approach should define a “limited scope” L2PSA, and the second approach should be a “full scope” L2PSA. If this policy had been pursued successfully, it had perhaps contributed to harmonization in the sense of defining minimum requirements for the “limited scope” L2PSA. It is a well-known mechanism that large groups tend to produce compromises which minimize the requirements to the parties concerned. This type of minimum requirement had probably been less demanding than the existing rules in most countries. One might rightly question the usefulness of such minimum standards. However, due to various reasons, this original idea of producing a “limited scope” and a “full scope” guideline has been abandoned.

The final ASAMPSA2 product is more a compilation of many existing approaches, than a streamlined recommendation for certain well-defined methods. Many sections in ASAMPSA2 explicitly provide different methods and show various solutions. One difficulty that has been identified is that the guidelines do not propose one precise step-by-step procedure: the user is supposed to take and use the relevant information depending on his objective. This question has been discussed during the project and it was accepted that, due to the complexity of L2PSA content (it represents a whole NPP (systems and operators), thousands of accidents situations and severe accident phenomena …) and depending on the final application, different technical solutions can be implemented.

Most countries do not apply quantitative safety goals related to L2PSA results. Therefore, one might expect that due to lack of national interest and with little hazard to interfere with existing rules it should be possible to reach a consensus. However, risk metrics and safety goals have been one of the few issues which have been discussed controversially in the ASAMPSA2 group, and the ASAMPSA2 guideline contains a minority statement to this topic.

Lesson learned:

Among the partners there are different levels of experience and various views on the scope and objective of L2PSA. The structure of the project did not provide approaches (e.g. negotiation groups) and frameworks (e.g. minimum requirements for the harmonization). Consequently, the ASAMPSA2 guideline is not very precise. But due to the complexity of L2PSA, and depending on the final application, it is obvious that different technical solutions can be valid

Trying to find consensus on controversial issues proved to be very difficult and time consuming in ASAMPSA2. If such issues show up, each view should be given room in the documentation, and minority statements should be allowed. If unresolved issues remain, they should clearly be documented.

3.2 ASAMPSA2 WORKING PROCESS

Within the ASAMPSA2 group an assignment of topics to the individual partners has been agreed in consensus. Some topics have been handed over to small sub-groups who organized the work on their own. This has been a successful partition of the work load. As a consequence, each partner could contribute according to his experience, and the final documentation covers a very large field of topics. There has been no controversial debate on the distribution of topics.

On the other hand, no requirement has been developed on how to produce the individual contributions. Neither the volume of the contributions, nor their structure has been pre-defined. Only the global structure of guidelines was defined and discussed. A certain evolution occured during the progress of the project. For example, it was decided quite late to develop a specific volume quite general on L2PSA (application of L2PSA was considered as a crucial topic). Consequently, the individual contributions had no similar appearance. Finally, much effort was needed at the end of the project to produce a consistent document. For this purpose, an editorial group had to be established which had not been foreseen in the project a-priori.

Unfortunately, the Fukushima disaster struck during the final phase of the project, when almost all technical contributions had been delivered, but the final documents still were in preparation. Many of the experts involved in ASAMPSA2 were affected by various activities in the wake of the accident. This caused significant delay in the finalization of the ASAMPSA2 documents. Fortunately it has been possible to postpone the delivery dates. Because this was a unique situation, it is not reasonable to draw general conclusions.

Lesson learned:

Assignment of topics to individual partners seems to be non-controversial.

If no requirements exist for the individual partners how to produce their contributions, the partners will contribute more willingly and easily (because they can start from their own experience, internal documents, …). But consequently significant effort may be needed at the end of the project to produce a consistent document.

As a minimum, if no precise requirements is defined for each defined contribution, the global content of all guidelines developed in ASAMPSA_E needs to be discussed quite early in the project.

4 LIST OF REFERENCES

[1] ASAMPSA2 BEST-PRACTICES GUIDELINES FOR L2PSA DEVELOPMENT AND APPLICATIONS
Volume 1 - General; Technical report ASAMPSA2 WP2-3-4/D3.3/2013-35

[2] ASAMPSA2 BEST-PRACTICES GUIDELINES FOR L2PSA DEVELOPMENT AND APPLICATIONS
Volume 2 - Best practices for the Gen II PWR, Gen II BWR L2PSAs. Extension to Gen III reactors; Technical report ASAMPSA2/WP2-3/D3.3/2013-35

[3] ASAMPSA2 BEST-PRACTICES GUIDELINES FOR L2PSA DEVELOPMENT AND APPLICATIONS
Volume 3 - Extension to Gen IV reactors; Technical report ASAMPSA2/WP4/D3.3/2013-35

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

[5] WENRA, Reactor Safety Reference Levels, January 2007, www.wenra.org.

[6] WENRA, Reactor Harmonisation Working Group, Safety Objectives for new Power Reactors, December 2009, www.wenra.org.

[7] IAEA-TECDOC-953, Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency

[8] IAEA-TECDOC-955, 1997, Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident

[9] U.S.NRC (2005). Good Practices for Implementing Human Reliability Analysis (HRA), NUREG-1792

[10] ASAMPSA2 - Results and Synthesis of Responses from the End-Users to the Survey on End-Users Needs for Limited and Full Scope PSA L2, S. Guentay, PSI, Villigen, Switzerland, Reference ASAMPSA2, Technical report ASAMPSA2/WP1/13/2008-13, PSI TM-42-08-10, October 2008.

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

[12] NEA/CSNI, Level-2 PSA for Nuclear Power Plants, Technical Opinion Paper No.9 (ISBN 978-92-64-99008-1).