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

"NUCLEAR FISSION"

Safety of Existing Nuclear Installations

Contract 605001

Final guidance document for extended Level 2 PSA
Volume 1

Summary report for external hazards implementation in extended L2 PSA, validation of SAMG strategy and complement of ASAMPSA2 L2PSA guidance

Reference ASAMPSA_E

Technical report ASAMPSA_E / WP40 / D40.7 / 2017-39 volume 1

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Author(s)	H. Löffler (GRS), E. Raimond (IRSN) (see also authors of [27], [28] and [29], especially E. Cazzoli (CCA), N. Rahni (IRSN) and M. Kumar (LR)), main authors of these reports.
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Summary :

The present document is a summary of the deliverables produced within the ASAMPSA_E project for extended L2 PSA. These deliverables are [27] : D30.7vol2, “Implementing external Events modelling in Level 2 PSA”, [28] : D30.7vol3 : “Verification and improvement of SAM strategy , [29] : D30.7vol4 : “Consideration of shutdown states, spent fuel pools and recent R&D results”.

Among many others, the following summary statements are provided:

Analyses of external events:

- No need for new methodology,
- It is necessary to develop L1 PSA first and then clearly defined boundary conditions for the L2 PSA must be generated,
- The remaining challenge is how to address adverse environmental conditions due to external hazards.

Multi units:

- No practical methodology exists to treat the problem,
- A new methodology is necessary to be developed first for the L1 PSA. This should, from the beginning, take into account the specific needs of L2 PSA so that the boundary conditions for subsequent level 2 analysis can be generated adequately.

SAM strategies verification and improvement:

- L2 PSA methodology can usefully by applied and experience exists for internal initiating events L2 PSA,
- How to address adverse environmental conditions due to external hazards - needs for new methodology or examples of experience,
- How to model the decision process when there is a conflict of interest - needs for new methodology or examples of experience.

For L2 PSA in shutdown states with open RPV, some new technical issues (fission product release, thermal load to structures above RPV) have to be addressed.

Spent fuel pool issues have been developed, in particular:

- Heat load from the melting spent fuel to structures above (e.g. to the containment roof) is a severe challenge for the plant and for the present-day, methodology is missing.

Recent R&D achievements with relevance for L2 PSA:

- Basic research has been continued in the radiochemistry (iodine and ruthenium chemistry) field, but the existing models are not yet fully suitable for routine application in L2 PSA, even if chemical forms are now better identified,
- Hydrogen and carbon monoxide issues within the containment are routinely taken into account in PSA. However, related issues outside the containment seem to require additional attention.

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3	2017-03-03	E. Raimond	all	Verification and editing modifications.

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CONTENT

GLOSSARY

APET	Accident progression event tree
BEEJT	Benchmark Exercise on Expert Judgment Techniques
BWR	Boiling water reactor
CAV	Cavity Package
CDS	Core damage states
CHRS	Containment Heat Removal System
CO	Carbon Monoxide
COR	Core Behaviour Package
CRT	Common risk target
DCH	Direct containment heating
DiD	Defence in depth
ECCS	Emergency core cooling system
EDMG	Extensive damage mitigating guidelines
EOP	Emergency operating procedure
ERO	Emergency response organization
FCVS	Filtered Containment Venting System
FLEX	Diverse and Flexible Coping Strategies
FP	Fission product
FSGs	FLEX Support Guidelines
HFE	Human failure event
HRA	Human reliability assessment
IE	Initiating event
IVR	In-vessel retention
IVMR	In-Vessel Melt Retention
LERF	Large early release frequency
LOCA	Loss-of-coolant accident
L2 PSA	Level 2 probabilistic safety analysis
MCCI	Molten Corium Concrete Interaction
MCR	Main control room
NPP	Nuclear power plant
PAR	Passive autocatalytic recombiners
PDS	Plant damage state
POS	Plant Operating State

PWR	Pressurized water reactor
QA	Quality assurance
RCS	Reactor coolant system
RPV	Reactor pressure vessel
R&D	Research and development
SAM	Severe accident management
SAMG	Severe accident management guideline
SBO	Station black out
SSC	Structures, systems and components
SFD	Spent fuel damage
SFP	Spent fuel pool
SG	Steam generator
SGTR	Steam generator tube rupture
SSM	Swedish Radiation Safety Authority
USA	United States of America
VVER	Water-water energetic reactor (Russian design)
WOG	Westinghouse Owner Group

SUMMARY

An extended PSA applies to a site of one or several Nuclear Power Plant unit(s) and its environment. It intends to calculate the risk induced by the main sources of radioactivity (reactor core and spent fuel storages) on the site, taking into account all operating states for each main source and all possible relevant accident initiating events (both internal and external) affecting one unit or the whole site. The combination between hazards or initiating events and their impact on a unit or the whole site is a crucial issue for an extended PSA

This report gathers conclusions of the 3 ASAMPSA_E reports on L2 PSA which complete existing ASAMPSA2 guidance [4], [5], [6] :

- D30.7vol2, “Implementing external Events modelling in Level 2 PSA” [27],
- D30.7vol3, “Verification and improvement of SAM strategy [28],
- D30.7vol4 : “Consideration of shutdown states, spent fuel pools and recent R&D results”[29].

Some outcomes are summarized hereafter.

Analyses of external events in L2 PSA

The following conclusions were reached for the modeling of external events in L2 PSA in ASAMPSA_E document D40.7vol2 [27]:

a) from the point of view of procedures/methods/approaches used currently in L2 PSA, there is no need of new methodologies in terms of PDSs, accident progression event trees development and evaluation;
b) the present guidelines identify the need of additional vulnerability/fragility analyses of systems, structures and components (like spent fuel pool, reactor containment, instrumentation, FCVS, etc.) needed for SAM strategies application in relation to all external hazards of various degrees of loads and intensity;
c) from the point of view of HRA more and higher stressors should be taken into account, e.g. within HRA models that use shaping factors. Assessment of human actions related to external events should be critically evaluated. SAM human interventions in particular seem to be appropriate as sensitivity analyses only in case of extreme conditions, especially if the utility has not implement a specific training program for such conditions.
d) from the point of view for multi-unit site analyses, it was concluded that:
   - no practical methodology exists to treat the problem,
   - no completely INDEPENDENT units on sites with several units are in operation; therefore, existing PSAs need QA re-assessment with respect to commonalities (and not only the potential for common cause initiating events),
   - a new methodology is necessary to be developed first for the L1 PSA, and clearly defined boundary conditions for L2 PSA must be defined there, considering that risk (and not only “site” frequency) of the whole site should be evaluated,

- a major conclusion in this respect was made: simplification of models is inevitable,

e) from the point of view of proper analysis of results, it was found to be useful to assign one additional identification character to the PDS codes keeping track of each and every internal and external hazard in order to make it possible to analyze at the end the contributors to the total risk by initiator related to the given PDSs,

f) from the point of view of proper analysis of results an application of proper risk metrics is necessary in order to make the best possible use of the PSA findings, especially to identify the main sources of risks and to support well founded decision making (see ASAMPSA_E document D30.7vol3 [30] on risk metrics).

SAM strategies verification and improvement in the context of L2 PSA is addressed in ASAMPSA_E document D40.7vol3 [28]. Within the issue of applying PSA for the implementation of SAM there are – among others - the following remarkable challenges:

- Safety grade equipment and also operational equipment should be taken into account.
- Is the SAM analysis restricted to the plant operating staff, or is a crisis team (internal or external to the plant) part of the PSA modelling?
- How to address adverse environmental conditions due to external hazards?
- How to model multi-unit issues (mutual support and/or spread of negative impact from an affected plant to the next one(s))?
- How to model the decision process when there is a conflict of interest (e.g. limited amount of water is available, but two SAM actions require water)?
- How to deal with opposing requirements (a classical issue is venting the containment: it leads to immediate environmental releases, but prevents later catastrophic release)?

The deliverable summarizes experience of each partner involved in SAM strategies verification and improvement, in order to derive some good practices and required progress in particular related to L2 PSA. There are concluding remarks on the following issues in this report:

a) Emergency team activation, rooms habitability, instrumentation, …
b) Human actions
c) Feeding steam generators with water
d) Corium cooling / water injection strategy
e) RCS depressurization
f) Control of flammable gas
g) Containment function (isolation, ventilation/filtration of auxiliary buildings …)
h) Containment pressure control
i) Radioactive release issues
j) SAM strategies for spent fuel pools
k) Links with external hazards

l) Links with equipment qualification

For L2 PSA in shutdown states, according to deliverable D40.7vol4 [29], two plant conditions are to be distinguished:

- Accident sequences with RPV head closed,
- Accident sequences with RPV head open.

When the RPV head is closed, core melt accident phenomena are very similar to the sequences going on in full power mode. Therefore, the large body of guidance which is available for full power mode is largely applicable to shutdown mode with RPV closed as well.

When the RPV is open, some of the L2 PSA issues become irrelevant compared to full power mode, while others come into existence. The following issues obviously are less significant as compared to closed RPV head:

- high pressure core melt sequences with the large number of associated complications;
- retention of radionuclides inside the reactor coolant loop; and
- restoration of heat removal system.

The situation is different for aspects which do not exist or which are less pronounced in sequences with RPV closed, such as:

- fission product release out of the RPV,
- containment issues.

Fission product release out of the RPV

Release fractions for closed RPV cannot be transferred to open RPV sequences. It is justified to assume that all fission products which are released from the degrading core will be transferred to the containment atmosphere. Moreover, in BWRs with closed RPV, the release in most accident sequences passes through the wetwell, thereby scrubbing large fractions of the radionuclides. This significant mitigating feature also does not exist when the RPV is open.

Containment issues

It is recommended that extended PSA Level 2 for sequences with open RPV carefully evaluate temperature evolutions in structures above the RPV. Heat radiation as well as convection out of the open RPV shall be considered. Typical integral accident simulation codes may be applied for this purpose; however care has to be exercised in the nodalization of the flow paths above the RPV.

Spent fuel pool issues

SFP guidance was not included in the scope of ASAMPSA2, so the SFP L2 PSA discussion is complemented in ASAMPSA_E. Section 5.2.14 contains a list of the issues which have been developed within deliverable D40.7 [29].

- Fuel degradation process, including energy and fission product release from melting spent fuel into containment. There is concern about the impact of air on the fuel degradation process.
- Hydrogen generated in a SFP inside the containment is in principle covered by the arrangements foreseen for core melt accidents. If the SFP is located outside the containment in the reactor building or in specific buildings, a significant risk of deflagration or even detonation exists. Altogether, there is a high probability for catastrophic releases if a SFP outside the containment begins to melt.
- Heat load from the melting spent fuel to structures above (e.g. to the containment roof). Several analyses show that the heat load from the SFP upwards to structures above (containment dome, or roof of reactor hall) is significant. Analytical models should include thermal radiation and apply a suitable nodalization to model convection.
- Release pathway for radionuclides from degrading spent fuel to environment. If the SFP is located inside the containment, the potential release paths to the environment are almost the same as for core melt accidents in the RPV. If the SFP is located outside the containment, the potential release paths to the environment depend very much on plant specific properties, e.g. ventilation systems, building doors, roof under thermal impact, size of rooms on the path etc.
- Concurrent accident progression in spent fuel pool and reactor system. Fuel melt occurs only if the plant status is in severe disorder. It seems difficult to prove that not both the reactor and the SFP would be affected by such disorder. This is especially the case for external hazards. Additional loadings due to SFP steam generation and melting processes will add an additional challenge for containments which house the SPF.
- Core concrete interactions for spent fuel pool accidents. It has to be taken into account that local peak heat fluxes at the upper edge of the melt pool in the SFP can exist. The erosion process and the failure of SFP structures should be assumed accordingly.
- Criticality: qualitative analysis should be performed to demonstrate that SFP criticality is not likely in case of PWR spent fuel pool
- Safety assessment of spent fuel pool during decommissioning: An interesting issue still to be solved is whether after a certain extended time the decay heat is so low that even without water significant fuel damage and radioactive release would not occur.

Recent R&D achievements with relevance for L2 PSA:

Recent development and the ongoing research with relevance on extended L2 PSA are evaluated based on the various on-going and completed research projects e.g. ASAMPSA_E, SARNET (Severe Accident Research Network), SARNET-2, OECD and European projects (public results only), NUGENIA roadmap and ASAMPSA2. A synthesis of acquired knowledge and remaining gaps is provided in deliverable D40.7vol4 [29] and summarized as follows:

1. L2 PSA guidance is missing on quantitative analyses of releases into the waters and ground and its related source term characteristics.
2. The long term resilience of containments against fuel degradation accidents are not adequately covered in existing L2 PSA. Although it is noted that some activities are going on in this field, the state of the art seems unfit for producing guidance for now.

3. Basic research has been performed in the radiochemistry (iodine and ruthenium chemistry) field, but the existing models are not yet suitable for routine application in L2 PSA. Source term R&D programmes conducted in the last two decades have shown that iodine oxide particles, gaseous organic iodides and gaseous ruthenium tetroxide may contribute significantly to the environment source term in case of venting. The filtration efficiency review and update of the filtered containment venting systems is the scope in European ongoing projects (MIRE and PASSAM). Furthermore the potential revolatization of the various deposited iodine and ruthenium species has to be further assessed for conditions representative of a severe accident. Despite the recent achievement of major experimental programs and significant advances in understanding of source term issues, additional research is still required as recently reviewed in an international workshop [63] for the consolidation of source term and radiological consequences analyses. Guidance cannot yet be provided for these issues. It is prudent to associate a high degree of uncertainty to releases of these species.
4. Hydrogen and carbon monoxide issues within the containment are routinely taken into account in PSA. However, related issues outside the containment seem to require additional attention.
5. The uncertainty analysis in L2 PSA shall provide information on the possible deviation in accident progression on the NPP and impact on the accident consequences. Solutions to this issue with respect to L2 PSA have been investigated within the EU project BEEJT. However, several sources of uncertainties cannot be easily addressed or quantified.

1 INTRODUCTION

An extended PSA applies to a site of one or several Nuclear Power Plant unit(s) and its environment. It intends to calculate the risk induced by the main sources of radioactivity (reactor core and spent fuel storages) on the site, taking into account all operating states for each main source and all possible relevant accident initiating events (both internal and external) affecting one unit or the whole site. The combination between hazards or initiating events and their impact on a unit or the whole site is a crucial issue for an extended PSA

The objective of the present document is to compile guides for extended L2 PSA which have been elaborated within the ASAMPSA_E project. It is based primarily on the following three deliverables of the ASAMPSA_E project, prepared in the work package WP40:

- D30.7vol2 : “Implementing external Events modelling in Level 2 PSA” [27],
- D30.7vol3 : “Verification and improvement of SAM strategy [28],
- D30.7vol4 : “Consideration of shutdown states, spent fuel pools and recent R&D results” [29].

The scope of the previous ASAMPSA2 project has been defined in [5]: “…guidelines for the performance and application of Level 2 probabilistic safety assessment (L2 PSA), for internal initiating events, …” In the framework of the present ASAMPSA_E project, this scope has been extended beyond what was addressed in ASAMPSA2, i.e.:

- Taking into account accident sequences initiated by external events. Considering SAM strategies to mitigate severe accidents and optimizing such strategies by application of L2 PSA. “Optimizing” is closely linked to the issue of applicable risk metrics and “figures of merit” which were discussed in ASAMPSA_E work package WP30.
- Taking into account accident sequences in the core in shutdown states and in the spent fuel pool.
- Updating the existing ASAMPSA2-documents according to recent R&D achievements.

It has to be noted that the present document is related to L2 PSA. L2 PSA addresses issues beginning with fuel degradation and ending with the release of radionuclides into the environment. Therefore, the present document does not evaluate the potential external and internal hazards which occur before the fuel begins to degrade. Such questions belong to L1 PSA, which will define boundary conditions for the L2 PSA. They were addressed by other work packages and documents within the ASAMPSA_E activities.

The ASAMPSA_E-partners want to express a certain concern with regard to existing guidance and L2 PSA practice. In spite of many IAEA and other documents and guides related to extended PSAs, apparently these have until now not been applied satisfactorily. Analyses in the field of extended L2 PSA are scarce, and several achievements may not be available publicly or even not inside the ASAMPSA_E community. Under such boundary conditions it was ambitious and difficult to set up valid guidance. Nevertheless the authors feel that the present document will be useful for future L2 PSA developments.

In the ASAMPSA_E final End-user's workshop [34] another basic concern has been raised: A full scope PSA will comprise issues with rater different degrees of methodological maturity and uncertainty. It may not be suitable to integrate all the different aspects into one single frame, because there might be certain issues which dominate the whole picture. While this is an important outcome in itself, it would probably overshadow many valuable insights on a more detailed level. Therefore, it might be wise to separate a PSA into appropriate sections.

2 GENERAL CONSIDERATIONS

2.1 GUIDANCE BASED ON FINDINGS OF END-USER'S SURVEY

2.1.1 EVALUATION OF END-USER'S QUESTIONNAIRE

An evaluation of the end-user's questionnaire has been performed within WP10. In [1] there is a detailed representation of all questions and answers. The present section represents the L2 PSA issues within [1] and provides information where these topics have been addressed in the ASAMPSA_E deliverables.

Just seven questions out of 100 in the questionnaire address L2 issues. The following statement numbers refer to the question numbers in [1].

The following statements summarize the evaluation of answers to the questionnaire. This compilation is very short by purpose, so that the statements can clearly be identified.
Of course such a short conclusion cannot consider all relevant comments or deviating opinions. The reader is encouraged to take note of the relevant sections in [1] order to get the full picture.

Based on an evaluation of the end-user's questionnaire which has been performed within WP10 [1], it is recommended to integrate the issues mentioned below. The question number of the questionnaire is provided in brackets for easy referencing. A pertinent comment from the L2 PSA point of view is added:

(85) It is reasonable to integrate the effect of internal hazards induced by external hazards.
Comment: This issue is particularly relevant for L1 PSA, where hazard combinations could cause core damage. For L2 PSA this issue seems to have limited relevance, because the potential impact of the induced internal hazards on L2 PSA will be reflected in the boundary conditions defined by L1 PSA.

(86) The evolution of external and/or internal hazards should be considered in accident progression.
Comment: Again, this issue is particularly relevant for L1 PSA, where an evolution of the hazard (e.g. earthquake aftershocks, continuously rising flooding level) can contribute to core damage. For L2 PSA this issue seems to have

limited relevance except for long term plant stabilization modelling after fuel melt (for example earthquake aftershocks). In principle, it could be integrated into an event tree analysis by adding branching points which address additional failures caused by the hazard evolution.

(88) Human reliability analysis for mitigative actions should be taken into account in extended L2 PSA.
Comment: This issue is addressed explicitly for L2 PSA in document [D40.7vol3][28]

(89) A long term management of radioactivity should be considered within L2 PSA.
Comment: It is state of the art to perform L2 PSA until no significant further release of activity is to be expected. Therefore, in theory, the abovementioned requirement is fulfilled. But in practice L2 PSA end when dynamic processes have finished, at maximum after a few days. Further long term processes, like corrosion of confinement components (reactor building, pipes, exchangers, …) are not routinely addressed.

It seems to be possible to derive the following additional overall conclusion on extended L2 PSA from the questionnaire: The community is not sure about the impact of an extended L2 PSA – this is understandable, because such complete analysis is very rare. If someone undertakes to perform such an analysis, the present methods are generally sufficient. The analysis should be as complete as possible – this is probably motivated by the lack of pertinent experience, so that it is difficult to a priori exclude certain issues from the analysis.

2.1.2 SUMMARY OF END-USER'S DISCUSSIONS MAY 2014

The ASAMPSA_E end user's meeting has been held in May 2014 in Uppsala, Sweden. A document has been issued which summarizes the findings [2].

The following table represents those issues which relate to L2 PSA and which are considered very important (Type A) or relevant (Type B) or of minor importance (Type C). There is also an indication which work packages apart from WP40 are covering the respective topic. Some comments from the L2 PSA point of view are introduced (marked "L2 comment"), and information is provided where the issue is addressed in the WP40 documents.

Table 1 Guidance needs identified by end-users May 2014

N°	GENERAL CONSIDERATIONS ON EXTENDED PSA	WP	Type
7	ASAMPSA_E shall address methodology for simultaneous accident progression in core and SFP. **L2 comment:** This issue is addressed in section 5.1.5	22 40	A
	INTRODUCTION OF HAZARDS IN L2 PSAs	**WP**	**Type**
1	ASAMPSA_E shall identify issues associated to external hazards that may need	40	A

	significantly different treatments in comparison with L2 PSA methodologies for internal IE, e.g - Induced effects (internal hazards) by external hazards, - Earthquake aftershocks, - External hazards impact on containment function **L2 comment on induced effects and aftershocks:** The end-users recommend that these issues should be addressed by L2 PSA, but it seems that they are more relevant for L1 PSA and should be covered there. In addition, it seems extremely ambitious to provide good practice for such issues. This issue has not been addressed.		
2	- Extended L2 PSA shall include long term management of radioactivity in the containment and release in environment. - ASAMPSA_E shall consider in long term strategies both in-vessel retention and ex-vessel retention. **L2 comment:** Present L2 PSA address radioactivity and retention issues. However, they end when dynamic processes have finished, at maximum after a few days. Further long term processes challenging safe cooling of debris and retention of radionuclides are not routinely addressed at present.	**40**	**A**
3	ASAMPSA_E shall examine existing containment venting strategies optimization versus L2 PSA results (status today: different strategies, depending on NPPs – is it consistent with L2 PSA results?) **L2 comment:** This issue is addressed in [28]	**40**	**A**
4	ASAMPSA_E shall examine SAMG sufficiency, especially for shutdown state (However, SAMG has to be known to develop pertinent event trees …) **L2 comment:** This issue is addressed in [28]	**40**	**B**
5	For shutdown states of reactor, ASAMPSA_E shall propose guidance for : - Open RCV or RCS situations : FP release (effect of air ingress), thermal radiation effect on the containment integrity (open RCV case, heat load) **L2 comment:** This issue is addressed in [29].	**40**	**A**
6	ASAMPSA_E shall examine how the conditional probability of SFP fuel degradation after core melt can be calculated (depending on common system core/SFP, on location of SFP – inside vs outside containment) **L2 comment:** The calculation of conditional probabilities of SFP fuel degradation after core melt involves system analysis for the SFP. Therefore, the methodology to be applied is L1 PSA (fault tree). Only some boundary conditions for this analysis may depend on the core melt effects und therefore can be provided by L2 PSA. ASAMPSA_E shall examine how far, in case of SFP fuel degradation (inside a containment), the containment function can survive (depending on pressurisation, hydrogen production, thermal radiation load …) **L2 comment:** This issue is addressed in [29].	**40**	**A**
N°	**COMMON ISSUES FOR MULTI-UNITS PSA**	**WP**	**Type**
1	ASAMPSA_E shall clearly identify deficiencies of single units PSA and promote development of multi units PSA. **L2 comment:** This issue is addressed in [27].	**22** **40**	**A**
3	ASAMPSA_E shall consider experience of countries like Canada having already developed multi-units PSA. **L2 comment:** This issue is addressed in [27].	**22** **40**	**B**
4	ASAMPSA_E shall in particular examine HRA modelling demand for multi-unit PSA (e.g.	**22**	**A**

N°		WP	Type
	team sufficiency if shared between units, site management complexity, equipment restoration possibilities, inter-reactor positive or negative effects …) **L2 comment:** This issue is addressed in[27], [28]	**40**	
N°	**COMMON ISSUES FOR HRA MODELLING (FOR ALL EXTERNAL HAZARDS)**	**WP**	**Type**
1	ASAMPSA_E shall examine how to improve HRA modelling for external hazards conditions to tackle the following issues : - the high stress of NPP staffs, - the number of tasks to be done by the NPP staffs, - the impossibility, for rare events, to generate experience or training for operators actions (no observation of success/failure probability (e.g. simulator), - the possible lack of written operating procedures (or non-precise procedures), - the possible wrong information in the MCR or maybe the destruction of the MCR, - the methodologies applicable to model mobile barrier installation (for slow developing event), - the methodologies available to model use of mobile equipment (pumps, DGs) and conditional failure probability (human and equipment), - the methodologies applicable to model equipment restoration (long term accident sequences, specific case of multi-units accidents, …). **L2 comment:** As already noted by the end-users, this is mostly a L1 issue. L2 parts of this topic are addressed in [28]	22 and 40 (TBD)	A

2.1.3 COMMENTS IN END-USER'S REVIEW AND WORKSHOP SEPTEMBER 2016

Drafts of the deliverables on L2 PSA in WP40 have been submitted to review by end-users. A summary document [34] contains the relevant comments and suggestions made during reviews and in the workshop in Vienna in September 2016. The present document and the updated deliverables within WP 40 take into account those statements. Therefore, this present document reflects the end-state of the ASAMPSA_E project achievements related to L2 PSA.

2.2 DEFINITION OF PLANT DAMAGE STATES (PDS)

The content of this section is relevant mostly if the L1 and L2 PSA analyses are not integrated. Moreover, the discussion of definition of PDS is valid for the analyses of initiating events occurring at full power and low power (which normally is part of the shutdown analyses). Since the definition of, and collection of data for the PDS are tasks that may fall upon different teams that perform the analyses (L1 and L2 PSA teams), this section provides a general summary intended primarily for L2 PSA analysts.

This section summarizes some views for the definition of PDS which are common to all external hazards.

It must be stressed, as was done for analyses of internal events, that this task involves close interaction between the teams performing the analyses. L2 PSA team has knowledge about boundary conditions necessary for characterization of accidents after core damage, and L1 PSA team know how accidents progressed up to that point and why core damage occurred. Therefore, this part of the works profits from feedback and potentially iterative work between the two teams in the course of defining the PDSs.

To this point, it is recommended that the L2 PSA team in general takes cognizance and understands thoroughly the definition of systems success criteria used in the Level 1 study, and in particular for accidents initiated by external events, what are the potential initiator-dependent systems failures (failure of systems that occurred as a direct impact from the initiator) and independent failures (failure of systems that may have occurred after accident initiation, at a time that for the most part cannot be specified by Level 1 analyses).

It is also strongly recommended that the L2 PSA team familiarizes itself with the results of Level 1 in terms of individual accident sequences or Minimal Cut-Sets (MCSs) that show the chain of failures (initiator, initiator severity, dependent systems failures, component failures, and operator errors) that ended in core damage.

Operator errors in L1 PSA are of particular importance for L2 PSA analyses if anticipated operator interventions that could be considered as part of SAMGs are introduced in L1 PSA in conjunction with interventions that are part of EOPs. This is the case for instance for containment venting, initiation of containment sprays, or initiation of firewater (or equivalent emergency system) injection in the RCS prior to core damage in BWR plants. In these plants for example, since many of the accident sequences from external events result in L1 PSA consequences similar to complete Station Blackout accidents with failure of all safety high pressure injection systems, the only option for preventing core damage would be to depressurize the RCS and initiate firewater as soon as possible. The danger is that this system may be over-credited in Level 2, if accident progression to the time of core damage is not thoroughly understood by the L2 PSA teams.

In addition, it is also strongly recommended that the L2 PSA team responsible for the definition of PDSs understand the role of auxiliary systems (such as compressed air, auxiliary and component cooling water systems, etc.) in the process of preventing core damage in particular accident scenarios, since these may fail as dependent on the initiator, without immediate failure of the primary safety systems.

For the purpose of “presentation of results” and “analysis of results” (especially for importance analysis) it is strongly suggested to include one additional characteristic in the definition of PDSs that describes the group of initiators. For instance, the following groups of initiators can be identified: internal fires, internal floods, seismic, aircraft crash, floods, tornadoes/high winds and corresponding identifiers to these should be used in the PDS codes in the analysis to differ them in order to recognize within the analysis, which PDS is addressed to which initiator, since the same sequence can be related to more types of initiators.
Moreover, if a group of initiators is subdivided in L1 PSA models into severity classes (e.g. seismic initiators class 1 or S1 considers seismic events with ground acceleration between 0.1 and 0.2 g, seismic class 2 or S2 considers events with ground acceleration between 0.2 and 0.3 g, etc.; or aircraft crash class 1 or A1 considers potential impact of small civilian airplane including crop dusters, class 2 or A2 considers impact of small military airplane, etc.), it is recommended that the PDS characteristic preserves the division into these classes.

The definition of PDSs that has been used for the internal events analysis has to be verified for applicability to L1 PSA accident sequences that are initiated by specific external events. The combination of dependent and independent systems failures due, for instance, to seismically induced sequences may require the definition of additional PDSs that were not considered possible for internal events. In addition, all external events may induce additional failures that were not considered for internal events (such as direct containment failure, containment isolation failure, piping failure inside or outside the containment, unavailability of main control room).

Finally, site personnel may be required to perform actions (such as venting of the containment prior to core damage) that would not be considered under accidents initiated by internal events and that change the status of the containment before the beginning of Level 2 analyses.

Note that some of these boundary conditions (especially with respect to the status of the containment function and attempts to perform interventions that could be considered as part of accident management, hence as part of Level 2) may in general not be of interest specifically to the Level 1 models, therefore it is the responsibility of the Level 2 analysts to alert their Level 1 colleagues on the need to tag or flag accident sequences where containment has been challenged and failed, or where some accident management actions have been exhausted.

Severe accident management strategies aim at protecting the containment during the accident progression: the L2 PSA teams shall identify precisely what is needed for this purpose (reactor building integrity, pipes and penetration tightness, primary circuit depressurization equipment, containment venting, instrumentation, recombiners, valves, electrical or air supply, human access to some rooms for some manual actions …) and examine if the external hazards can induce damage. This information shall then be available in the PDS characteristics.

Considering the Fukushima Dai-ichi accident, also additional structures should be taken into account in addition to containment status – e.g. status of spent fuel storage/pool, which, in current analyses may not be commonly included, except e.g. for the Swiss Mühleberg one-unit BWR plant where a complete analysis was performed in 2011-2013 including all external events, all operational modes including fuel damage in the fuel pool (for details see: [38], [39], [40]). The location of the spent fuel storage should be considered and included in PDS characteristics – if outside the containment or inside the containment. Location of the spent fuel pool outside the containment represents a quite significant potential source of risks in case of hydrogen generation and its immediate release into reactor building with no additional measures from the point of view of defence-in-depth (missing last physical barrier of containment).

Additional characteristics for defining PDS with particular importance for L2 PSA do not seem to be needed. Any example we could think of would be an accident with somehow catastrophic consequences in Level 1 (everything fails), so that any issue impacting Level 2 would be “mute”. For instance fires after large aircraft impact in the reactor building would have no additional meaning, since in this case either the containment is penetrated / fatally damaged (failure of all pipes assumed due to failure of reactor building and systems located in the building), or the fire should have been taken into consideration in Level 1 (failure of equipment due to fire following the aircraft impact).

2.3 ANALYSIS AND PRESENTATION OF RESULTS

The ASAMPSA2 document (Vol. 2, [5]) contains an extended section on this topic. In spite of this, the present sections are a complement to and re-enforcement of the discussions given there because the need for proper analysis and presentation of results is of crucial significance for PSA quality.
Some solutions in this respect are discussed in detail in WP30. Even though these issues are addressed in other ASAMPSA_E documents as well, these topics are expanded here as integral part of the performance of L2 PSA.

WP30 of ASAMPSA_E was involved in a general discussion of risk metrics and PSA results [30]. The present section concentrates on those topics which are of particular relevance for L2 PSA.
“The objectives of a probabilistic safety analysis are to determine all significant contributing factors to the radiation risks arising from a facility or activity and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined.” ([8] IAEA SSG-4, para. 1.2).

In IAEA INSAG-3 [31], chapter 3.3.4, item 84, the following paragraph can be found:
“Probabilistic analysis is used to evaluate the likelihood of any particular sequence and its consequences. This evaluation may take into account the effects of mitigation measures inside and outside the plant. Probabilistic analysis is used to estimate risk and especially to identify the importance of any possible weakness in design or operation or during potential accident sequences that contribute to risk (which should be more precisely interpreted as: that might cause excessive contribution to risk).”

Therefore, the results should be provided at least partly in form of risk(s). For PSA we should accept in general (i.e. irrespectively of specific risk measures) the definition of risk as defined in INSAG-12 [31] §14, pg. 8: “the risk associated with an accident or an event is defined as the arithmetic product of the probability of that accident or event and the adverse effect it would produce”.

An important deficiency noted in analyses, as concluded within ASAMPSA2 project, is that in spite of the IAEA definitions and requirements, the results are currently depending on PSA objective, and “risk” evaluation complying with one of the IAEA fundamental principles is currently performed in various ways because there is no

common understanding of the "adverse effect". The second deficiency related to L2 PSA results is, that no common harmonized risk metrics exists to compare the level of safety. As a surrogate, currently a frequently used parameter is LERF (Large Early Release Frequency), which is only semi-quantitative without an exact definition of "Large" and "Early" without harmonized values of frequency throughout the European countries.
The observations mentioned above apply to the status of many present-day PSAs. Considering these shortcomings in traditional PSA, it is justified to discuss adequate risk metrics within the "extended" scope of ASAMPSA_E.

Existing PSA methodology is able to provide results for any type of risk metrics. As it is discussed in [4] ASAMPSA2 Vol 1, various results in various forms are produced within the L2 PSA assessments depending on the scope/objective of L2 PSA; among these the most commonly analyzed are:

- Frequency of containment failure - first containment failure, dominant containment failure modes.
- Individual containment failure modes and related frequencies.
- Magnitude and frequency of releases for the different containment failure modes.
- Frequency of releases - based on releases, in/out of APET evaluation, based on kinetics, on containment failure time, on delay before obtaining an activity release limit; this category covers L(E)RF.
- Containment matrix (probability of containment failure modes as a function of accident initial conditions or CDS).

This means that the results, by showing different phenomena or parameters, are usually not comparable in a process of cross-checking and thus consistency and comparability of the results of different L2 PSA studies cannot be ensured.

L2 PSA should carefully check the local requirements. Several panels have been, and are still, compiling and comparing the various practices. In this respect different limits and practices in different countries exist and it depends on local authorities what kind of results they ask for and indeed what quality, depth and extent of the analysis of results is required.
If, for instance, in the local legislation LERF is used for L2 PSA results, then it depends, what else the authority asks in the legislation to show about the results (importance analysis, contribution of chosen PDSs to final frequency, contribution of chosen containment failure modes to final frequency etc…). Sometimes nothing more than LERF results are specifically required.

Since for the most part regulatory requirements concentrate on the demonstration that a target on large release frequency is met, and no demonstration is asked for total risk or even risk profile (frequency versus releases), accident sequences may not be analyzed according to their contribution to total risk. Then it is not possible to conclude that the plant is really balanced thus complying with the general safety objective, i.e. there are no specific sequences identified with a significant contribution to total risk. Consequently, decision making focusing on limited risk metrics will dismiss other accident related consequences. Even though the results might be in accordance with safety requirements of an authority (e.g. LERF or LRF values), they might not satisfy some of the basic safety principles and objectives as mentioned above, and decisions made on such basis may be misleading.

Unfortunately, however, no harmonized or unanimously accepted risk metrics exists. The related discussion is provided in reports D30.7vol4 [32] (DiD), D30.7vol3 (Risk metrics) [30], or D.30.7vol1 [33] (Decision making) of the ASAMPSA_E project, where also recommendations are given for suitable results presentation.

Within the ASAMPSA2 project the idea of Common Risk Target (CRT) was proposed by Jirina Vitazkova and Erik Cazzoli representing the CCA company within the project ASAMPSA2, described in Chapter 6 of the ASAMPSA2 Guidelines (Vol. 1, [4]). The methodology used to derive the proposed Common Risk Target (CRT) was fully worked out within a dissertation thesis and published in 2013 in the journal Nuclear Engineering and Design [35]. The methodology is based on grouping sequences leading to releases according to INES scale grades. This helps to recognize if the plant is really balanced – i.e. if none of the release groups causes a significant contribution to the total risk. The CRT parameter is based on the constant risk principle (Farmer's curve) and its quantitative value is comparable with other industrial risks by transforming releases in TBq to consequences. In the context of the CRT (and the IAEA INES definition [36]) it is necessary to use radiological equivalent toxicity of I_{131} and include all the released radioactive elements.
The CRT method is mentioned here as an example for calculating the total risk because it is in particular related to L2 PSA. Within WP30 of the ASAMPSA_E project there is a more comprehensive discussion of various risk metrics ([30] D30.7vol3)

2.4 DEFENCE-IN-DEPTH AND L2 PSA

L2 PSA have identified several deficiencies in existing plants with regard to severe accidents, although these plants apply the DiD concept. This is not surprising because practically no existing plants (Generation I and II NPPs) were designed against such events. Identified issues include also instances where DiD is not or not well implemented for such conditions. One example is the fact that the fuel cladding made out of Zr, which is considered to be a reliable second physical barrier within the first safety layer of DiD concept under normal operation as well as under conditions of Design Basis Accidents, becomes a source of risk at beyond design basis temperatures because together with steam it is a source of hydrogen. From this point of view the fuel cladding should not be considered to be a safety barrier with respect to severe accident conditions or PSA. Another example is, for many plants, the insufficient containment pressure load capacity in severe accident conditions, which leads to the necessity of a venting system. This means that the containment, which constitutes the last barrier in the DiD, is not well suited to manage severe accident conditions, except if the filtration capacity of the venting system is so good that offsite impact becomes negligible. Nevertheless, the severe accident which occurred in the TMI plant demonstrated a successful DiD concept: the inner barriers were lost due to fuel melting, but the containment remained intact and functional.

The large majority of severe accidents initiated by external hazards can be represented by sequences which are very similar to transients initiated by internal initiators, or loss of offsite power sequences. For such external hazard scenarios the DiD issue is not different from the well-known internal initiator topics. There is, however a subsection of external hazard scenarios which can directly threaten the containment, i.e. the outermost (last) barrier in the first place. If this last barrier fails first, it may be difficult to demonstrate that the remaining inner barriers still constitute adequate protection levels. Therefore, the applicability of the DiD concept may be questioned under such conditions.

3 GUIDANCE ON THE IMPLEMENTATION OF EXTERNAL EVENTS MODELING IN EXTENDED L2 PSA

D40.4 [27] provides guidance in the implementation of external events modeling in extended Level 2 PSA for states at power, and is a complement of the ASAMPSA2 guidelines in this area. Issues that belong only to Level 1 analyses will not be discussed, and it is assumed that the only relevant issues to be resolved are those subsequent to core damage and/or fuel degradation, i.e. after permanent loss of core cooling and/or decay heat removal functions. Following the accident at Fukushima, the community has realized that much attention should be given to the areas of operator interventions and accidents that may develop at the same time in more than one unit if they are initiated by one or more common external events. For this reason, the attention is mostly focused on interface between Level 1 and Level 2, human response analysis and some consideration is given to Level 2 modeling of severe accidents for multiple unit sites, even though it is premature to provide extensive guidance in this area.

3.1 EXTERNAL EVENTS TO BE CONSIDERED

A complete list of events to be considered for extended PSA has been proposed by the ASAMPSA_E project in [37]. It has been decided in ASAMPSA_E deliberations to group most of them into six main groups that are discussed within the ASAMPSA_E guidelines in separate documents, and these are shown in Table 2.

Nevertheless, from the point of view of L2 PSA the specific initiator is not important, since the analysis starts at the time of “core damage”, and what is important is to know the boundary conditions at that time (i.e., it is important to know how the accident reached that point, regardless of what initiated the chain of failures). Therefore, it should be kept in mind that the present L2 PSA guidance is not just specific to the six groups of events shown in Table 2, but covers all events that result in core or fuel damage due to loss of coolant level and/or decay heat removal functions.

Table 2 Groups of external initiating events considered in details in ASAMPSA_E

Initiator group	Initiating events or natural phenomena included
Seismic	Seismo-tectonic events
External floods	Extreme precipitation; events that cause swelling of waterways and/or lakes (in general including elevation of sea level); failure of dams; tsunami
Extreme weather	Effects of high or low temperature; high wind and tornadoes; excess snow
Lightning	Stroke of lightning on power lines, switchyard, transformers, electromagnetic disturbances to electronic components
Biological hazards	Biological (animal, plant) infestation within the installations and water supplies
External explosion, aircraft crash, external fires	Man-made events such as external explosions, civilian and military aircraft (large and small, including crop dusters) crashes, external fires

3.2 IMPACT OF EXTERNAL EVENTS ON L2 PSA ISSUES

It is assumed that the team or teams performing the L2 PSA for external events will be already familiar with the procedures and protocols to be used in the analysis for internal events. All the relevant information can be found in Vol. 1 of the ASAMPSA2 guidelines [[4], Sections 2.1 through 2.15] and the technical approach is discussed in Vol. 2 [[5], Sections 2 through 7].

The vast majority of core damage sequences induced by external events behave as sequences that are induced by a loss of power event. A smaller number of sequences (especially for the most severe initiators with extremely large consequences) behave as containment bypass or containment failure prior to core damage.

It is recommended that a limited set of specific accident sequence sensitivity analyses should be performed to verify that the results of analyses for internal events apply at least for risk dominant Level 2 sequences within the uncertainty bounds. E.g. for extreme weather conditions, the initial and boundary conditions for severe accident calculations should cover a range of outside environmental temperatures representing occurrence frequencies to a given frequency threshold.

All phenomena related to in-vessel accident progression, vessel failure, and ex-vessel accident progression should be then reviewed, including source terms.

It should be noted that current Level 2 guidelines (e.g. the IAEA [8] and Swiss ENSI guidelines [7]) have no *specific* requirement and *very few* recommendations for the performance of L2 PSA for external events, indicating that the expected impact of external events on the performance of Level 2 is not as great as it is for the performance of Level 1. Nevertheless, some relevant issues will be addressed in the following sections.

3.2.1 DEFINITION OF PLANT DAMAGE STATES (PDS)

The definition of PDSs that are used for the internal events analysis has to be verified for applicability to Level 1 accident sequences that are initiated by specific external events. The combination of dependent and independent systems failures due, for instance, to seismically induced sequences may require the definition of additional PDSs that were not considered possible for internal events. In addition, all external events may induce additional failures that were not considered for internal events (such as direct containment failure, containment isolation failure, piping failure inside or outside the containment, unavailability of main control room). Finally, operators may be required to perform actions (such as venting of the containment prior to core damage) that would not be considered under accidents initiated by internal events and that change the status of the containment before the beginning of Level 2 analyses. Nevertheless, plant damage state properties from a comprehensive L2 PSA for internal initiators will be mostly applicable to external initiators as well. New relevant combinations of properties may show up, but the number and type of properties will not have to be modified significantly.

The content of this section is relevant mostly if the L1 and L2 PSA analyses are not integrated. Moreover, the discussion of definition of PDS is valid for the analyses of initiating events occurring at full power and low power (which normally is part of the shutdown analyses). Since the definition of, and collection of data for the PDS are tasks that may fall upon different teams that perform the analyses (L1 and L2 PSA teams), this section provides a general summary intended primarily for L2 PSA analysts.

Definition of PDS involves close interaction between the teams performing the analyses. L2 PSA team has knowledge about boundary conditions necessary for characterization of accidents after core damage, and L1 PSA team knows how accidents progressed up to that point and why core damage occurred. Therefore, this part of the works profits from feedback and potentially iterative work between the two teams in the course of defining the PDSs.

For the purpose of “presentation of results” and “analysis of results” (especially for importance analysis) it is strongly suggested to include one additional characteristic in the definition of PDSs that describes the group of initiators. For instance, the following groups of initiators can be identified: internal fires, internal floods, seismic, aircraft crash, floods, tornadoes/high winds and corresponding identifiers to these should be used in the PDS codes in order to recognize within the analysis, which PDS is addressed to which initiator, since the same sequence can be related to more types of initiators.

Moreover, if a group of initiators is subdivided in L1 PSA models into severity classes (e.g. seismic initiators class 1 or S1 considers seismic events with ground acceleration between 0.1 and 0.2 g, seismic class 2 or S2 considers events with ground acceleration between 0.2 and 0.3 g, etc.; or aircraft crash class 1 or A1 considers potential impact of small civilian airplane including crop dusters, class 2 or A2 considers impact of small military airplane, etc.), it is recommended that the PDS characteristic preserves the division into these classes.

The definition of PDSs that has been used for the internal events analysis has to be verified for applicability to L1 PSA accident sequences that are initiated by specific external events. The combination of dependent and independent systems failures due, for instance, to seismically induced sequences may require the definition of additional PDSs that were not considered possible for internal events. In addition, all external events may induce additional failures that were not considered for internal events (such as direct containment failure, containment isolation failure, piping failure inside or outside the containment, unavailability of main control room).

Finally, site personnel may be required to perform actions (such as venting of the containment prior to core damage) that would not be considered under accidents initiated by internal events and that change the status of the containment before the beginning of Level 2 analyses.
Note that some of these boundary conditions (especially with respect to the status of the containment function and attempts to perform interventions that could be considered as part of accident management, hence as part of Level 2) may in general not be of interest specifically to the Level 1 models, therefore it is the responsibility of the Level 2 analysts to alert their Level 1 colleagues on the need to tag or flag accident sequences where containment has been challenged and failed, or where some accident management actions have been exhausted.

It should be noted however that, when here it is stated that the Level 1 analyses can provide information that is important to define boundary conditions for the Level 2 analyses, especially where the containment status is concerned, it is meant always within the bounds of Level 1 specific analyses and competences. For instance, a specific structural analysis for failure of the containment due to earthquake has to be performed for Level 1 and is required e.g. by the Swiss ENSI A-05 Section 4.6.2.1 of [7] o discuss the SSC fragility analyses, and it is stated that both structural failure of the containment and failure of pipes that would lead to containment bypass must be considered and assessed. These fragility assessments however are meant to provide information relevant to failures that can influence reactor systems or operations of related components. Even with the ENSI requirements, as far as containment failure is concerned, only gross structural failure is normally considered in Level 1, because this may cause failure of pipes and components (or even the reactor vessel) housed within the containment. The potential for cracks and leaks of the containment is <u>not</u> generally included, and therefore <u>the Level 1 SSC fragility studies cannot provide this information</u>. Responsibility for the assessment of leaks from the containment, including failure of penetrations, following an external initiating event should be assigned to the Level 2 assessment. Considering the Fukushima Dai-ichi accident, also additional structures should be taken into account in addition to containment status – e.g. status of spent fuel storage/pool, which, in current analyses may not be commonly included, except e.g. for the Swiss Mühleberg one-unit BWR plant where a complete analysis was performed in 2011-2013 including all external events, all operational modes including fuel damage in the fuel pool (for details see: [38], [39], [40]). The location of the spent fuel storage should be considered and included in PDS characteristics – if outside the containment or inside the containment. Location of the spent fuel pool outside the containment represents a quite significant potential source of risks in case of hydrogen generation and its immediate release into reactor building with no additional measures from the point of view of defence-in-depth (missing last physical barrier of containment).

Additional characteristics for defining PDS with particular importance for L2 PSA do not seem to be needed. Any example we could think of would be an accident with somehow catastrophic consequences in Level 1 (everything fails), so that any issue impacting Level 2 would be “mute”. For instance fires after large aircraft impact in the reactor building would have no additional meaning, since in this case either the containment is penetrated / fatally damaged (failure of all pipes assumed due to failure of reactor building and systems located in the building), or the fire should have been taken into consideration in Level 1 (failure of equipment due to fire following the aircraft impact).

3.2.2 CONTAINMENT ANALYSES AND PHENOMENA

In order to address the potential of leaks from the containment following and external initiating event, it is recommended that, in addition to containment fragility analyses for events that occur within the containment (internal missiles, internal pressurization, explosions, etc.) fragility analyses should be performed within Level 2 to assess:

- The probability of cracks crossing and traversing the entire wall of the containment resulting in leaks and isolation failure following specific external events initiators (a complement of Level 1 seismic fragility analyses), and
- The probability of failure of any of the containment penetrations (cable, pipeline) leading to containment isolation failure in case of specific external events initiators.
- The probability of failure of any of the containment access doors (man-holes, hatches) leading to containment isolation failure in case of specific external events initiators

These analyses should be performed only with respect to external initiating events that have a direct impact on the containment (e.g., they need not be done for biological infestation events, lightning, external explosions ...).. Additional mechanistic codes analyses will be needed in case new or additional external-events specific PDSs have been identified. Protocols and best-practices applicable to these processes are found in Sections 2, 3 and 4 of [5]. For PDSs that are common between internal and external events, there could be an impact of external events on physical phenomena in level 2 after core damage; e.g. the timing of events could be affected.

3.2.3 HUMAN RELIABILITY ANALYSIS

The techniques used to assess human actions in L2 PSA are discussed in detail in Chapter 3 of the ASAMPSA2 guidelines, Vol. 2 [5]. For the most part, the current models are adequate but some points that are specific to the conditions to be expected during accidents initiated by external events should be carefully reviewed as discussed below.

L2 PSA accident sequences that are induced by external events should be examined in order to verify and take into account whether site personnel interventions have not already been credited in Level 1 either as part of the EOPs or as recovery actions. In addition, the availability of systems that may be credited for Level 2 may be impacted by the specific initiators (availability of signals, non-plausible/misleading signal(s), failure of components, loss of site personnel, problems with the Technical Support Center). Special attention should be given to the availability of sufficient resources (systemic and human) for multi-units sites. After some adaptation, existing HRA methods should be able to cope with Level 2 issues after external impact.

However, the real challenge seems to be proper modelling of the actual situation. As by definition of Level 2 PSA, the external event has been so powerful that it has caused failure of systems, structures, signals etc., resulting in core damage. Therefore, the staff might have to face extremely serious conditions and degradation of plant systems, possibly including disrupted communication lines, inaccessibility of resources, and missing personnel. In addition, external or internal radioactivity levels may preclude interventions that involve work outside or inside the buildings. It is obvious that human reliability under such conditions is very uncertain.

The HRA methods and data used in the analyses should be critically examined with regard to their applicability under the described circumstances. Potential screening criteria (or additional criteria and performance shaping factors for the quantification of probabilities of operator failure) for this task may be:

- Is only the plant itself affected by the external event (e.g. aircraft impact), or is the whole region affected (seismic, flooding, typhoon), which would leave the plant without external support?
- Is the external event fast (e.g. aircraft impact, seismic), or slow (e.g. heat wave), and was there an opportunity for preparation against the external event?
- Is the external event itself also affecting human performance (e.g. extreme storm, snow, smoke, debris, radioactivity or victims of casualties or even corpses on the site)?
- How is the crisis team (who is in charge of ordering and initiating the SAM actions) going to respond to the potentially extreme conditions?
- How efficient can be the rescue teams from outside of the plant considering the amplitude of damages, kinetics of accident, radiations, …
- How is accident management affected by interventions performed potentially by unskilled or not properly trained personnel?

With respect to the last concern in the list given above, it must be re-iterated that some actions may have to be performed by unskilled operators (e.g. the fire brigade). A large weight should be given to the issues of training and skill of the operators or personnel who are involved in the management actions, and much less weight to the time available to perform them, even though in many cases this time may be very long. It should be noted that relatively long time available needs not necessarily be an asset, since a longer time for implementation might mean also more potential mistakes or may induce a too optimistic or lax attitude of the personnel involved (including the crisis team). It seems quite clear that L2 PSA assumptions for HRA will depend on the quality of training of the utility emergency teams and on the existence of procedures that would allow crisis team to take decision in due time and avoid an aggravation of the situation.

Given all the uncertainties introduced by the quantification of the potential shaping factors that would properly describe and characterize the SAM interventions, and given that the SAM actions in L2 PSA *per-se* are implemented for mitigative purposes, it might be advisable in sequences with extremely high level of stress to perform the basic analysis without consideration of SAM human interventions, especially if the utility has not implement a specific training program for such conditions. Under such adverse conditions SAM should be investigated in sensitivity analyses that would show what are the potential (but not assured!) benefits of the implementations. This will also provide good information of the resilience of the plant containment safety function.

The examination of the shaping factors that drive human responses under severe accident conditions is essential for integrating SAMGs and for implementing HRA approaches in light of external hazards.
The list of human performance shaping factors for L2 PSA that should be carefully reviewed before implementation in the models includes (see also deliverable D40.5 [28] relative to SAMG implementation):

- physical and psychological stressors that are likely to influence performance in severe accidents need to be realistically modelled; if the accident is extending over multiple days it will impose severe mental and physical fatigue on control room operators, field staff, and personnel in the plant's emergency response centre ; note that "level of stress" per se may not be modelled as a performance shaping factor, nevertheless the issue is whether stress is properly taken into account especially for accidents initiated by external events.

- control room operators and field personnel are also exposed to physical stressors (e.g., loss of lighting and high radiation) as well as psychological stressors associated with risk to their health and lives and those of their co-workers and families, posing an extra load on the control room operator performance;
  in particular operator actions need to consider the possible environmental factors, posed by the extreme harsh working environment conditions, including radiation levels and high temperatures;
  for example, operators could not take some critical control actions from the control room; instead, they should take manual actions in the field ; radiation releases in the plant and limited access of the personnel to equipment could hamper the ability of personnel to perform their duties, both in the control room and in the field; some field activities could require multiple teams because of difficult onsite conditions; flooding, debris, and other hazards caused by the external event and by the severe accident phenomena, like the hydrogen explosion, limit access to some parts of the buildings and challenge the field response.

- communication to transmit information and instructions in an accurate and timely manner plays an important role in shaping actions at certain points during the accident response;
  this item encompasses communications and real-time information systems to support communication and coordination between control rooms and technical support centres, control rooms and the field, and between onsite and offsite support facilities; it should be noted, that the hierarchy of responsibility for some actions and issuing instructions should be known and clear to everybody; to this aim the availability of the communications equipment that the staff will need to effectively respond to the accident (e.g., radios for response teams, cellular telephones, and satellite telephones) must be ensured;

- operators training: the operators could encounter situations that go well beyond their training for responding to off-normal conditions; in responding to severe accidents at nuclear plants, operators are likely to face complex, unanticipated conditions (e.g., multiple interacting faults, failed or degraded sensors, goal conflicts, and situations not fully covered by procedures) that require them to engage in active diagnosis, problem solving, and decision making to determine what actions to take; this implies that emergency response procedures should involve all the scenarios which include core damage and operator training should routinely exercise the whole range of SAMG response options and involve as well multiple unit scenarios;
  here is a need for HRA methods that more accurately model the kinds of complicating situational aspects that are likely to arise in severe accidents and the psychological processes that underlie performance in these situations.

- real-time information about conditions at nuclear plants for monitoring critical thermodynamic parameters related to the severe accident progress and phenomenology, as fuel rod – water interaction, hydrogen build up and combustion, fission product release, molten fuel relocation and MCCI (molten core concrete interaction), etc.; it should be also noted that based on some conditions (e.g., radiation levels), all operations in the open on site may be stopped and non-necessary personnel evacuated.

The performance of all the related instrumentation for the diagnosis of severe accident and monitoring at deteriorated plant conditions should be taken into account in the probability of human interventions.

This present section on human reliability analysis has close links to section 4(“Guidance on the verification and improvement of SAM strategy with L2 PSA”), and in particular to section 4.4 (Links with external hazard).

3.2.4 QUANTIFICATION OF EVENT TREES

This section deals only with assessment of the conditional probabilities of the branches in the accident progression event trees. A basic necessary precondition for this task is proper estimation of physical phenomena including containment performance, and of human reliability. The quantification approach is, in principle, similar for external hazards and the conventional PSA. However, some particular remarks have to be formulated:

Containment fragility analyses should be performed for Level 1 (fragility due to external loads during accident initiation) and Level 2 (fragility due to internal loads during accident progression). This fact was recognized already in the NUREG-1150 analyses [19]. In the Level 2 APETs developed for the five plants, provision was made to model these potential additional containment failure modes with the addition of a top event that allowed for quantification of the conditional probability of “Pre-existing containment leaks and containment isolation failure” prior to accident initiation, in addition to the quantification of the other modes of potential containment failure.

As already mentioned in the section on human reliability, the success probability of human actions under the conditions of an externally initiated accident is extremely uncertain and difficult to estimate. Furthermore, since in many cases the external event was powerful enough to cause so many failures in SSCs to induce core melt, it is not at all certain whether equipment needed for SAM (if not designed for that conditions) or any other action is available and functional. In this case, the role of personnel and the crisis teams in fact becomes insignificant.

An example of problems connected with SAM interventions is containment venting: even if the venting system is designed properly to cope with core melt accidents initiated by internal events (e.g. if it can manage steam, hydrogen, fission products, and can retain volatile radionuclides), and even if the actions necessary to operate a venting system are simple and very quick; nevertheless the impact of the external event may have affected e.g. valves, piping, filters, or the stack. The consequence of misled venting exhaust containing hydrogen has been clearly visible in the Fukushima Dai-ichi accident.

This example shows that adequate quantification of event tree branches may be much more complicated when taking into account disturbances from external impacts. Given the restraints in time and budget which normally exist when performing PSA, it seems to be not realistic to expect a complete quantification of a full set of external event sequences. At best, it may be possible to address particular selected issues, e.g. the conditional probability of successful venting after a certain initiator (e.g. external events initiators with relatively low intensity). Facing the difficulty of quantifying the event tree, one might assume that accident management actions will not be possible at all, which could be an unjustified conservatism.
For this reason, it would be advisable to separate higher intensity initiating events (i.e. duplicate event trees and quantify them differently for core damage sequences due to higher intensity external initiators), and assume that for these accident management actions will not be possible at all. The event trees with lower intensity events should then try to assign some success probability to accident management, including the following issues:

- potential damage to instrumentation and control devices,
- potential damage to structures where the necessary equipment is stored,
- potential damage to the equipment itself,
- impairment or even death of key personnel, and
- disruption of communications and means and ways to move the equipment around the site.

3.3 ISSUES INVOLVING MULTI UNITS SITES

Most of the plants currently operating are multi-unit sites and it is really urgently needed to consider this issue in PSA. No complete satisfactory methodology or guidance exists as of the date this was written. This document has made full use of the information that can be gathered from [41] and the related literature, which includes experience from Canadian PSAs. Although it is a general issue for PSA, it is addressed here in the section on external events because external events seem to be the most significant contributor to sequences affecting more than one unit.

The accident at Fukushima has shown that accidents at multi-units sites should be given special consideration, given the possibility of common cause failures among the different plants in different states of operation. Moreover, it is possible that the units at a site are in different operation modes: in one unit, the risk may be due to the SFP (core unloaded), in another unit, the core may be the main threat.
Each of the plants at the site behaved differently and final consequences (releases to the environment, at least according to current information) varied for each of the units at the site. The different behaviors and responses of the units and final core status including extent and type of containment damages in Fukushima Dai-ichi prove that even though the units are very similar and they are at the same place and being threatened simultaneously by the same initiator, there are many unforeseen factors that may influence the progression and final result of a severe accident. All these factors may raise the doubt whether the current PSAs are "realistic" at all even for a single unit, and perhaps the community should return to a more conservative approach. Nevertheless, this section attempts to provide, if not guidance, some points that should be considered when addressing multi-units in PSA, and some suggestions on procedures for resolving some of the issues connected with such PSAs.

At first glance a PSA for multi-units sites seems to be merely a technical issue and a question of resources to simply adding and combining sequences in more than one unit on a site. However, when looking more closely it becomes clear that significant challenges are involved. These challenges arise from the fact that a plant housing more than one unit can be subjected to the following sets of problems arising from intra-unit (i.e. within individual units) and inter-units (i.e. from connections among units) dependencies or correlated phenomena:

- **Common cause initiator:** an event (external or internal) affects more than one unit on the site (common initiating event) ; some SSCs fail due to the initiator, but these dependent (on the initiator) failures occur randomly and in different combinations in the units ;consequently, the accident progresses in different ways in each unit ; no other common dependent failure occurs, either in systems, components or structures ; all recovery actions by the operators progress completely independently in each unit ; one or more units may reach core damage conditions while the others do not (Level 1) ; after core damage, accident progression still goes on independently within each unit, and SAM interventions proceed independently (Level 2) ; if such a scenario could be irrefutably proven, then the results for the whole site are only a matter of combinatorial analysis.
- **Common cause failure of systems:** there could be inter-connections among systems, and common cause failure of systems as a whole could occur due to the same initiator ; one simplistic example which is valid for BWRs and PWRs is as follows: the Auxiliary Cooling Water System (ACWS) of two or more units could share the intake from sea or river; if the intake is blocked, the ACWS and cooling of the components in the secondary side (PWRs) or in the balance of plant (BWRs) is lost for all units; after the common initiating event that essentially trips all cooling pumps in all units, other failures may occur in each unit independently and therefore the accidents in the units start along the same path and then progress

according to the other failures ; the list below provides more examples of systems that are typically in common.

- **Common cause failure of operator actions:** resources for recovery may be shared among the units ; if the resources are not available for one unit, due to initiator or other causes, they are not available for any other unit ; the accidents in the units will probably end in the same way (the same PDS) ; here it might be noted that maybe only one unit is affected by e.g. a seismic/external initiator and the other/others not, but an operator failure causes the failure of the originally non-affected units (this being however a L1 PSA issue).
- **Potential correlations and dependencies between components, systems and operator actions**: this could be considered an analogy of the common cause failure of components already considered in L1 PSA for internal events: the initiating event could induce the same type of failure in components due to latent reasons, such as poor maintenance in all units or partial failure of components due to the initiating event.
- **The crisis center that guides the management of accidents is shared among the units**: the issue is whether the crisis center can cope with managing more than one severe accident at a time, and whether, if a wrong decision is reached for one unit, the same wrong decision will be reached for all units (e.g., venting the containment when not necessary).

The last two points are mostly relevant for L2 PSA. Integrated and very detailed models such as have been developed to analyze single unit L2 PSA, or which are suggested within the ASAMPSA2 guidelines cannot be developed for analyses of accident progression of more than one unit at the same time. Moreover, as is recognized by e.g. [42], supporting or adjunct mechanistic or probabilistic models for multi-units analyses are lacking at this time (both for study of accident progression and of consequences). For now the ASAMPSA_E guidelines need to point out that the following more specific issues need to be addressed when considering multi-unit sites:

- The proper definition of “risk” to a site seems to be the integral of all releases multiplied by frequencies for all sources including spent fuel pools.

- the units on a site are not totally independent: at a minimum they share the crisis center and at least external energy supplies/electric grid/transformers and switchyards which are interconnected for back connections and jumps which are used not only for “OUT” energy but also “IN” energy in case of loss of production necessary for self-consumption. The dependencies and feedback between units can be properly modeled only if a single dynamic super-model that can track accident progression in all units at the same time is used; this is practically impossible; therefore, modeling accident progression in one unit at a time seems to be the only solution.

- if the units are not independent - how should the dependence be modelled? Note that in practice the units on one site often are not identical; given that analyses should be performed one unit at a time, the only solution may be to introduce dependencies in an iterative way, by re-quantifying some nodes in the

APETs according to results of a single unit; this would also take care of the fact that units at a site are not necessarily of the same type and make.

- the final maximum released quantity of an accident in two units is about twice the maximum release from a single unit. This implies in fact that the risk, whatever may be the definition of risk, posed by a site with N units <u>may be in first approximation</u> N times the risk posed by a single unit site, and the analysis could be stopped there: when compared to the other uncertainties in releases and frequencies this factor of two or even N is insignificant. So, it does not really appear justified to spend much effort on <u>detailed</u> multi-unit analyses for accidents that progress in more than one unit, and analyses should be simplified as much as is reasonable, and introduction of conservatisms should be considered. Please note that even this first approximation is valid <u>ONLY if Level 1 PSA can provide a defensible and complete analysis of all inter- and intra-units connections, dependencies and correlations that could trigger conditions conducive to CD in all units at the same time, even for internal initiating events.</u>

At the present time, it seems to be already clear that modelling common cause failures (caused by the external event) in more than one unit opens a large field of practical modeling (especially the probabilistic models and tools capable of accommodating the potential size of combinations) and computational problems (extension of existing mechanistic and probabilistic consequence codes). All these layers of complexity may actually be sufficient to warrant stopping at the first approximation of risk estimates (total site risk equal to N times the single unit risk).

ASAMPSA_E suggests some approximations, introduction of conservatism and simplifications as discussed in the next section below. Please note that the scheme shown here is only an example that can resolve some of the issues detailed above.

For these possible procedures here to be valid it is necessary that L1 PSA provides adequate information about accidents that occur or are under way at the same time in more than one unit. It must also be remembered that L1 PSA for the most part deals with prevention of core damage and thus does not necessarily cover all possible sequences potentially significant and in progress after core damage, while Level 2 deals only with mitigation of releases and consequences from severe accidents that cannot be prevented.
Bearing then in mind that models should be simplified, and assuming that the only inter-unit dependencies during accident progression after core damage are in the area of SAM operator interventions, the following procedure is suggested:

1. Clearly establish major objectives of calculations in terms of the risk measures that should be provided (see D30.7vol3 [30]), bearing in mind that not all risk measures may be actually calculated. Nevertheless,

the end product should be the estimation of overall RISK (probability that adverse consequences from all accidents at one site will occur in a given period of time, as defined by IAEA) and comparison with appropriate safety targets. The use of risk based safety goals, in combination with deterministic safety goals, provides a way to develop balanced, technology neutral, expectations for the protection of worker and public health and safety and a means for an independent and integrated assessment of plant safety.

2. Simplify existing single-unit models (APETs), keeping them compatible with the objectives (risk measures compatible with common risk targets) that must be provided (e.g., one potential simplification could be a broad characterization of release classes as performed by EDF (D30.7vol3 [30]), rather than characterization of releases by specific release modes). Analyze APETs one unit at a time (i.e., it is not envisioned that super models may be developed even with a very simplified scheme of characterizing release modes).
3. Identify, from L1 PSA results, accidents that are expected to occur simultaneously in more than one unit: specific super-PDSs should be provided.
4. Define consequence/release dominant containment failure modes from analyses of single-unit APETs and source terms assessment and prioritize these modes in the quantification of APETs: which are the “very large”, “large”, “medium”… release modes, and in which time frame they are expected to occur. The INES scheme [36] (Farmer’s curve) should be used for reference of what is “large”, “medium” etc.
5. Assume that the unit which is expected to fail in one of the failure modes conducive to large releases actually fails first (by containment bypass, by failure of containment isolation, by early containment failure…..). Here an example is given for a two-unit site. The time of release defined in point 4 determines which unit should fail first. For example, in a combination of PDSs in which unit 1 fails in a bypass mode, and unit 2 fails as Station Blackout, the containment failure of and releases from unit 1 certainly precede any possible containment failure of unit 2, and any intervention in the open for unit 2 is thus precluded (see next point). If both units fail as Station Blackout, the conditional probability of early containment failure of unit 1 defines in first approximation the dependent failure probability of interventions in the open for unit 2 (see next point).
6. After the failure in one unit as described in point 5 conservatively assume that, due to the large releases occurring from the first failure, all accident management interventions for all other units that need working in the open will completely cease or will be impeded for an extended period of time (this assumption takes also care of uncertainties in the decision of intervening correctly and at the appropriate time by the crisis center), and therefore will likely fail for all the other units.
7. Quantify event trees according to the assumptions made in point 6.
8. Eventually iterate the tasks 3 through 8 to arrive at consistent results.
9. Integrate results for the calculations of the various failure modes for all units.

This proposed model only assumes that the APETs are built and run for individual units and the multi units effects and consequences are calculated separately by appropriate integration tools (EXCEL spread sheets can be useful).

Note that some inter-unit CCFs (the potential containment system CCFs, if the systems are not independent) are taken into account if the PDS characteristics are properly defined, because the failure of containment systems can be calculated before Level 2 through appropriate systems analysis (that can be taken from the existing Level 1 models). Iterations may be necessary only for sites with more than two units.

4 GUIDANCE ON THE VERIFICATION AND IMPROVMENT OF SAM STRATEGY WITH L2 PSA

4.1 INTRODUCTION

The severe accident management guidelines (SAMGs) are related to equipment and procedures that should be applied in case of a severe accident. A L2 PSA should model all fuel damage accidents identified by L1 PSA (fuel in reactor cores or storages) and analyze their progression taking into account:

- the equipment availability and resilience in severe accident conditions,
- the actions (both correct and erroneous) by the response teams in relation with the existing guidance (SAMG),
- the actions specified for the local emergency team as well as the actions specified for national emergency teams,
- the impact of severe accident strategies on physical phenomena, containment failure modes and radioactive release (different options can be studied and their impact on L2 PSA results can be compared).

SAM strategies and accordingly L2 PSA should cover the broadest possible scope of severe accidental situations (internal and external initiators, spent fuel pool, multi-units…). SAM strategies make use of components or systems under severe accident conditions. Note that for any NPP most equipment is not designed for severe accident conditions and that this has to be considered in PSA. Therefore, especially for Generation 2 plants (for which severe accident conditions are beyond design), the effectiveness of SAMG measures shall be evaluated critically in the L2 PSA.

The ASAMPSA2 guidelines [4], [5] discuss in detail how to introduce these issues in a L2 PSA or how to present the results of L2 PSA. Verification and improvement of SAM strategies is closely related to the issue of L2 PSA risk metrics to be applied for this task. Since mitigative SAM aim at reducing radioactive release to the environment, a risk measure should be selected which characterizes the radioactive impact outside the plant. Deliverable D30.7vol3 [30] of the ASAMPSA_E project addresses several potential risk measures which might be suitable for this purpose.

The D40.7vol3 deliverable [28] is the key document for L2 PSA SAM issues within ASAMPSA_E. It presents experience in eleven countries, addressing:

- L2 PSA regulatory framework,
- Role of L2 PSA,
- SAM objectives.

The main body of D40.5 deals with the identification of SAM strategies and with technical features of a L2 PSA for SAM strategies verification. The following sections summarize these issues.

4.2 IDENTIFICATION OF SAM STRATEGIES

The main objective of severe accident management (SAM) is to mitigate the consequence of a severe accident and to achieve a long term safe stable state. For successful and efficient SAM first the endangering processes and their likelihood must be recognized. Then SAM strategies have to developed, taking into account their potential benefit and their requirement in terms of human and system resources. It is trivial to ask for such a balanced approach, but it is more than difficult to realize it.

From a theoretical point of view it is very desirable to identify and define SAM based on a well-structured approach, applying full scope PSA models. Such an approach is certainly feasible for the implementation of SAM which involve no too difficult human action, or which consider plant states which can largely be represented by the existing analysis models. Examples for such SAM with limited complexity are passive autocatalytic recombiners (PARs) to cope with hydrogen challenge, or containment venting procedures.

However, it has to be recalled that L2 PSA deals per definition with plant conditions which are so severely disturbed that it has not been possible to avoid core melt – although preventing core melt is assured by probably the most sophisticated systems and procedures which exist in the history of industry. Therefore, dealing with SAM under core melt conditions has to acknowledge a difficult, probably chaotic and dangerous environment. Staff which has to take action carries the burden that a catastrophic technical or human failure has occurred, and that a disaster is imminent where their health or life is at risk. Evaluating system availability or human actions under such conditions obviously is extremely challenging. In addition, still considerable uncertainties exist in the accident simulation codes, so that the related results are not always a sound basis for judging SAM.

In particular after the Fukushima Dai-ichi disaster there was direct need for rational installment of additional safeguards against extreme and unforeseen circumstances. For almost all plants additional hardware and/or SAM procedures have been or are being implemented. Unfortunately, L2 PSA has only rarely been used as guidance in the decision process. This may be partly due to the difficulty of the issue as mentioned above, and partly to the pressing time constraints which called for urgent action without time available for extensive analyses. A third momentum may be the fact that in some cases the cost for performing detailed analyses may be comparable to the cost of a SAM procedure under consideration.

Having said that, it remains to be stressed that there is unanimous agreement provisions should be made for efficient SAM under severe accident conditions. Furthermore, the selection and design of SAM should be as reasonable as possible. Adequate PSA certainly is a very good basis for decision making. After the hasty activities in the wake of the Fukushima Dai-ichi events, it is advisable to apply PSA now for checking the benefit or possible improvements of the updates made.

Within the issue of applying PSA for the implementation of SAM there are – among others - the following remarkable challenges:

- Safety grade equipment and also operational equipment should be taken into account.
- Is the SAM analysis restricted to the plant operating staff, or is a crisis team (internal or external to the plant) part of the PSA modelling?
- How to address adverse environmental conditions due to external hazards?
- How to model multi-unit issues (mutual support and/or spread of negative impact from an affected plant to the next one(s))?
- How to model the decision process when there is a conflict of interest (e.g. limited amount of water is available, but two SAM actions require water)?
- How to deal with opposing requirements (a classical issue is venting the containment: it leads to immediate environmental releases, but prevents later catastrophic release)?

The tables 3 and 4 below present the main risk issues and objectives in case of severe accident phenomenon for PWR and BWR respectively, and some corresponding SAM strategies able to avoid or to limit radiological releases. These tables are just a set of examples and do not represent a complete list. In each plant specific PSA pertinent screening is needed for potential SAMs, followed by an assessment of their impact on the accident evolution.

Table 3 Main SAM issues and objectives in case of severe accidents – PWR

Risk/Objectives	SAM Strategies or design provisions
In-vessel phase	
Confirm entry in SAM	Criteria depending on reactor status (e.g. full power, shutdown state, SBO). Change priority : containment function instead of core integrity.
Get efficient emergency teams	Emergency team activation (local, national, utility, public bodies, …) Communication, radioprotection, data transmission… Strategy to keep control room, emergency control, crisis centers habitability (radiation protection, team rotation …)
Activate / repair any system which might be useful	Identify systems which are operable and systems which have failed ore are not operable, or could be brought back to operation. Identify systems which are strictly necessary to manage the severe accident
Decrease RPV pressure	Reduce the RPV pressure to support use of low pressure systems Reduce pressure to lower than 0.5 MPa (value depending on the NPP design) to avoid DCH during vessel rupture
Prevent Induced Steam Generator tube rupture	RCS depressurization Limit SG depressurization. Feed SG with water
Prevent Containment isolation failure	Check the containment isolation Close the containment if needed (specific procedures depending on initial reactor state - full power, shutdown states …)
Prevent gaseous release through ventilation	Control the ventilation device (filtration) and limit non filtered release
Control contaminated liquid release in auxiliary building or in environment	Reinjection of contaminated water in the containment Isolate leakage Use circuit with intermediate heat exchangers to avoid direct contamination of the environment Limit the circulation of contaminated water outside of the reactor containment
Control of flammable gases (H2)	PARs, igniters, containment inertisation strategy … Control of in-vessel water injection Control of containment spray system activation
Control the containment pressurization	CHRS, FCVS, …
Prevent large releases	SG isolation, ventilation control, spray the containment, depressurize the containment, flood the containment
Prevent vessel rupture	In-vessel water injection External flooding of RPV (IVR) Containment flooding
Confirm plant status	Instrumentation use to identify core melt, containment status, radioactive contamination and release

Risk/Objectives	SAM Strategies or design provisions
Vessel rupture phase	
Prevent containment failure due to DCH at RPV failure or vessel uplift	RCS depressurization Control in-vessel water injection Containment design (containment design pressure, geometry of internal structures to limit corium dispersion, geometry of the cavity to limit vessel uplift … (for new design)
Prevent containment failure due to ex-vessel steam explosion	Prevent vessel rupture (IVR) Limit water in reactor cavity Geometry of cavity (large cavity and small flow paths limit risks) (for new design)
Confirm plant status	Instrumentation use (RCS pressure during core melt, vessel rupture)
Ex-vessel phase	
Get efficient emergency teams	Activate additional support for plant-external SAM and related decisions (e.g. venting strategy)
Prevent basemat failure due to MCCI	Prevent vessel rupture (IVR) Optimize geometrical features: large area for corium spreading, large width of the basemat (core-catcher for new design, upgrade for existing NPPs) Suppress containment bypass in the basemat (e.g. close pipes) Control water injection: for corium cooling, to allow corium spreading, to quench the corium after the vessel failure ….
Control of flammable gases (H2, CO)	PARs, igniters, containment inertisation strategy … Monitor containment atmosphere conditions Control containment spray system activation
Control the containment pressurization	CHRS, FCVS, … Apply containment venting system Apply containment heat removal circuits able to withstand severe accident conditions
Prevent gaseous release through ventilation	Control the ventilation device (filtration) and limit non filtered release
Control contaminated liquid release in auxiliary building or environment	Reinjection of contaminated water into the containment Isolate leakage Use circuit with intermediate heat exchangers to avoid direct contamination of the environment Limit the circulation of contaminated water outside of the reactor containment
Prevent large releases	Control the pH in the sump, SG isolation, ventilation control, spray the containment, depressurize the containment, flood the containment, protect containment venting filter
Confirm plant status	Instrumentation use to identify containment status and radioactive contamination and release

Risk/Objectives	SAM Strategies or design provisions
Other	
Mitigate SFP accidents	Strategy if the SFP is inside the reactor containment Strategy if the SFP is outside the reactor containment

Table 4 Main SAM issues and objectives in case of severe accidents – BWR

Phase	Objectives	SAM Strategies or design provisions
In-vessel	Strategy change	Change focus from protecting the core to limiting releases to the environment
	Keep high pressure in RPV	If steam driven systems are used - Secure that the pressure in the vessel will be high enough for long time
	Lower RPV pressure	If no need exists for steam driven systems, a strategy shall be implemented to reduce the RPV pressure to support use of low pressure systems - e.g. fire system pumps Reduce pressure to pressure lower than 0.5 MPa to avoid DCH during vessel rupture
	Prepare for vessel penetration	Transfer water to be available under the RPV - need several hours to be performed
	Avoid critical gas mixes	For inerted containments avoid air intrusion to the containment
	Keep the PS-function	Support (keep available) as long as possible the pressure suppression function of the wet-well
	Feed and bleed	Establish a feed and bleed – status in which water is feed in to the vessel and bleed out through SRVs or a pipe break - with the aim to be independent of water level measurement Alternative establish the best possible knowledge about water level in RPV and control cooling water according to the measurement
	Containment status	If the containment has been open for handling scenarios before core damages (e.g. direct venting, filtered venting) it is of importance to secure the closure of the containment after the core damages occurs
Vessel rupture	Ex-vessel steam explosion	The best strategy for a plant with large water under the RPV (lower part of containment) is to keep the water level as high as possible (i.e. with short distance between the vessel bottom and the water level to reduce the loads from steam explosions)
Ex-vessel	Water filling in containment	Follow a strategy related to fill the containment with water which secure that steam production is low and gas phase in the containment is large enough to avoid drastic pressure increases. Avoid filling above the bottom of the RPV-level. Fill water slowly
	Containment status	Measure /Control leakages through the containment. Use available methods to control any increases of leakages through the containment
	Venting trough filtered venting systems	Follow procedures for open and closing valves to the filtered venting containment system
	Cooling of water in containment	Initiate any available functions including mobile functions to cool the water in the containment. As soon as possible, the cooling shall support to close the FVCS if open.

4.3 TECHNICAL FEATURES OF A L2 PSA FOR SAM STRATEGIES VERIFICATION AND IMPROVEMENT

Lessons for L2 PSA and SAM strategies improvement are given in the report, according to the plant design (PWR or BWR, SFP location …). The following recommendations have been highlighted.

Emergency team activation, rooms habitability, instrumentation, …	
Emergency Team	L2 PSA shall be able to identify scenarios: • where the emergency teams can fail to manage the severe accident due to context factors like time constraints, , extreme conditions …, • where no human action is possible.
SAMG entry	L2 PSA shall be able to identify scenarios where operators can miss the SAMG entry due to context factors like time constraints or hazards.
Room habitability	Functionality of Control Room shall be evaluated for several events (flooding, fires, earthquake …) and in case of radioactivity contamination (containment venting, containment leakage through the auxiliary buildings or directly outside).
Communications	This issue should be considered using post-Fukushima reinforcement of communication means.
Instrumentation	Instrumentation is needed to get a correct view of the plant status even during a severe accident and help emergency teams to take appropriate decisions. A precise modelling of the plant status is needed in L2 PSA for any application. The importance of instrumentation on L2 PSA results depends on its real use in procedures (for instance when SAMG entry is based on physical measures: core temperature, dose rate ...).
Training	L2 PSA results shall be used to assist staff trainings to emphasize the importance and positive impacts of certain human actions.

Human actions	
Modelled actions	Actions specified in the EOP/SAMG shall be modelled with their respective conditional success probabilities in the L2 PSA. Actions not specified or imprecisely specified in the EOP/SAMG shall not be credited at all.
Crucial actions	HRA shall be a relevant tool for safety improvement. Thus, identification of the crucial actions (that can lead to a significant effect on L2 PSA results), has to be performed periodically, i.e. during regular safety reassessment. This identification can be used as input data to improve actions operability by optimisation of related EOPs and SAMG. These crucial actions should also be taken into account by their inclusion in crew trainings, in consistency with WENRA RL.
Actions dependencies	The following issues can be analyzed with L2 PSA: • the dependencies between human errors before and after severe accident entry, • the impact of context factor on human errors. This can be developed for the extended PSA approach (internal and external hazards …).

Human actions	
Environmental conditions for actions	For each L2 PSA scenario, support studies should be used to verify conditions of intervention: time available, pressure, temperature….
Time dependent action modelling	New approaches can be investigated to combine a dynamic model of crew behavior with a dynamic model of the plant systems and physical processes (e.g. dynamic reliability analysis).

Feeding steam generators with water	
Priority level of SG water feeding action	There is unanimous understanding that feeding the SG has very high priority for several reasons.
Positive/negative impact of SG water feeding action	WOG SAMG (used in Belgium and Slovenian case) requires that before the injection into SGs is started, to identify and evaluate any negative impacts and to determine consequences of not feeding the SGs. Therefore, L2 PSA shall be also able to model both positive and negative effects of filling the SG. For example, negative aspects may be caused by injection of cold water into hot dry SG (thermal shock of SG), or by increasing the secondary pressure inside the SG (possibly leading to contaminated releases through the SG valves), or by depressurization of SG (creep rupture of SG tubes). The modelling shall distinguish SGTR cases, in particular related to sequences with and without SG isolation.

Corium cooling / water injection strategy	
In-vessel water injection	A precise link must be done between accident evolution and L2 PSA assumptions. A dialog between L2 PSA teams and researchers/experts in using severe accident codes (e.g. MAAP/MELCOR/ASTEC) is needed, in order to know how reflooding a melting core can be modelled with such codes. L2 PSA shall be able to model both positive impact of water injection (i.e. core cooling, in-vessel retention) and negative impact (i.e. hydrogen production and its combustion, vessel over-pressurization, in-vessel steam explosion, DCH, containment over-pressurization). The modelling shall be supported by specific analyses with severe accident codes. L2 PSA shall take into account repair of components which would lead to injection into a previously damaged core. L2 PSA shall be used to identify available timeframes and injection flow rates needed. Some studies could be done to improve the water management to prioritize the different sources of injection based on the different accident phases for the cooling success and for the combustions that could lead to losses of the systems. L2 PSA can be used to understand where the issues are (which scenario, which timeframe …). There are organizations which have concerns about the injection of too little water into the core because this might enhance hydrogen generation rather than improve coolability. Other organizations opt for an injection in any case, whatever the circumstances. Within the present compilation it was not possible to judge the reasoning for these positions. However, these discrepancies strongly suggest that the issue should be covered in a L2 PSA as precisely as

Corium cooling / water injection strategy	
	possible and finally provide advice for the SAM to be applied.
External flooding of RPV	There are plant designs where flooding the cavity is not possible – in this case the issue is not relevant. But there are also plants where SAM foresees flooding the cavity in order to prevent RPV failure and / or to cool debris below the RPV in case of its failure. Since there are also disadvantages involved in a flooded cavity, L2 PSA, associated to relevant analysis in support of phenomena like steam explosion, structural behavior or fuel debris quenching, is an indispensable tool for providing advice on the SAM to be selected.
Ex-vessel water injection	L2 PSAs shall include detailed analysis of the corium concrete interaction with and without late flooding. All issues shall be considered: corium cooling, gas production, containment pressurization, impact of late spray system activation, loss of instrumentation and equipment. The aim of these analyses is to assess all effects of water injection onto a molten pool, positive or negative. It seems that, for some Gen II power reactors, the main effect of water addition onto a molten pool would be to enhance steam production, without much success probability for stopping the core concrete interaction and avoiding a reactor containment failure. For some other Gen II power reactors, the basemat width, its concrete composition and the area available for the corium spreading provide high chance of success for the corium stabilization. L2 PSA can be used to evaluate the ex-vessel core debris cooling strategy of a given unit, for instance to decide whether to inject water into the cavity before or after vessel failure.
Risk analysis	Strategy for corium stabilization needs obviously a multi-criteria risk analysis. L2 PSA should be used to determine an optimal strategy able to : • Reduce as far as possible occurrence probability of energetic phenomena (hydrogen and carbon monoxide explosion, steam explosion, HPME and DCH) able to threat the confinement of radioactivity, • Reduce as far as possible the risks of containment bypass, • Reduce as far as possible the risk of over-pressurization (from steam and gas production), • Maximize the conditional probability of corium stabilization after severe accident entry. Global risk metrics for L2 PSA (see ASAMPSA_E deliverable D30.5) can be used to demonstrate the optimization of the strategy.
Importance of research	It is crucial in that area that teams in charge of L2 PSA development are supported by researchers in severe accident progression. All assumptions in L2 PSA, which influence the risks results, shall be appropriately justified. No undue conservatisms shall be applied in the L2 PSA assumptions because it can discourage decision of NPP reinforcements. The role of researchers for L2 PSA development is to provide consolidated opinions on knowledge, quality of modelling, uncertainties, … so that L2 PSA risk analysis is meaningful.

RCS depressurization	
Action benefit	RCS depressurization is a SAM strategy which is universally implemented. Since there are almost no doubts that depressurization is safety enhancing, L2 PSA is not exploring the benefits or drawbacks of SAM, rather than potential reasons for failure of this SAM procedure and the related consequences. A distinction has to be made between L2 PSA (during core melt depressurization is always beneficial) and L1 PSA (before core melt depressurization would stop steam driven pumps, and reduce the remaining coolant level in the RPV).
Conditional failure probability	L2 PSA shall consider failure of depressurization considering the human failure and system failure (safety valves, portable equipment). If the qualification and reliability of the safety valves for SA conditions is guaranteed, human failure becomes the main contributor to a failure in RCS depressurization. However, even when active depressurization fails, there are mechanisms which could reduce the pressure: high temperature failure of hot leg, surge line or steam generator tubes; and failure of safety valves in stuck open position. These (partly beneficial) failure modes should be considered in L2 PSA as realistically as possible. If such failure modes can be demonstrated as likely, efficient and not detrimental, the impact of the depressurization SAM procedure (and its failure) becomes less significant. Such demonstration may be difficult if the initial design is not intended for such events.
Scenarios	L2 PSA can help to check that the safety valves depressurization capacity is sufficient for a large panel of scenarios (e.g. electrical losses) and conditions (i.e. severe accident conditions, external hazards).
Specific risks in case of late RCS depressurization during core melt	L2 PSA can be used to identify scenarios with late depressurization and associated risks (e.g. fast hydrogen release into the containment: hydrogen in the primary circuit + hydrogen produced by the impact of accumulator water discharge).

RCS depressurization	
Long term management of RCS pressure.	During in-vessel accident progression (with an objective of in-vessel corium stabilization), the primary pressure may have to be controlled by the RCS safety valves for a long period of time. L2 PSA can be used to analyze the possibility of late SRV closure, for example : • closed in a SBO situation by depletion of batteries, • closed manually by the operators (error or simply because the situation seems to improve - e.g. after RCS flooding) Several issues are of interest for L2 PSA : • the RPV re-pressurization can cause the loss of the core coolability supported with low pressure injection systems, • if the SRV are closed in a RCS full of water, there is no steam inside to control pressure and the primary circuit can easily be at overpressure, • during the late phase of accident, the conditions can be beyond the qualification of the SRV ; the capacity of the SRV to be operable can be questioned, • the coupling between containment heat removal, RCS pressure and water injection possibility can be of crucial importance : o for example, this has conducted to the loss of steam driven water injection pumps for the Fukushima unit 2 and 3, o the containment pressure increase has a direct impact on the RCS pressure and may make some low head pumps unavailable. It is recommended that L2 PSA teams concentrate not only on the short term efficiency of in vessel water injection to stop the corium progression but check that the accident can be managed safely for the long term. The Fukushima accident shall be used as a lesson to demonstrate that even after 24 or 48 h, a reactor may not be stabilized. For BWR: L2 PSA shall assess the EOP/SAMGs procedures related to water level control in the RPV during the complete sequence from core damages to the final end-state. It will be of importance to understand: • existence of a clear preferred water level at each time of the scenario, • identification of systems needed for controlling the water level to the preferred level-measuring systems, process systems, power supply systems and other supporting systems, • failure modes that will be developed if the preferred mitigating systems fails. Such assessment will need to include assessment scenarios with a fixed water level inside the RPV as well as scenarios where the vessel is flooded (above the steam line) and bleed through the relief valves into the containment.

Control of flammable gas	
Objective of L2 PSA	It is expected that a L2 PSA demonstrates that in case of a severe accident, SAM strategies are able to reduce at a low level the conditional probability of containment failure induced by flammable gas burning. In general, the efficiency of the flammable gas management system (recombiners, igniters, …) is demonstrated by a limited number of calculations of postulated accident; associated to conservative assumptions (deterministic approach). The role of L2 PSA is to verify by a number of additional scenarios the efficiency of the system and of the human or automatic actions (if any). Uncertainties shall be considered in the assumptions of the L2 PSA APET. If some specific situations can lead to containment damage, then, depending on their frequency, improvement of SA strategies shall be considered.

Control of flammable gas	
Scenarios to be considered	The scenarios to be considered are defined with the list of PDS coming from L1 PSA. The systems activations or reconfiguration during severe accident progression have to be considered. The approach is plant dependent: • simple approach may be practical for NPPs with significant safety margin against flammable gas combustion effects, • more complex or precise approach is needed if the safety margins are low (typically for some Gen II reactors) (see below).
Modelling the time dependent phenomena in the event trees	For the risk quantification by L2 PSA, if the safety margins against effect of hydrogen combustion are limited, it may be needed to model some coupled phenomena with dynamics modeling : • kinetics of hydrogen or carbon monoxide release in the containment, • time of spray system activation and kinetics of steam condensation, • time of in- or ex-vessel water injection and effect on flammable gas release in the containment, • radiolysis and recombination in air and in water. Such analysis is obviously difficult to perform but is useful to assess how appropriate are the SAMG strategies for hydrogen system management (water injection, spray system activation, containment inertisation …). As described by some organizations, it may be needed to apply more sophisticated event trees (so called dynamic PSA technics) to take into account time dependent effects and dependencies between phenomena and SAM.
Source of uncertainties to be considered	The L2 PSA assumption shall take into account existing uncertainties, for example: • on hydrogen production during core degradation with or without reflooding, • on PARs efficiency, • on PARs ignition effect, • analyzing atmospheric processes with lumped parameter codes and coarse nodalization, • on time of combustion (stochastic phenomena except if a controlled igniter is used), • on radiolysis process if PARs are not available.
Effects of flammable gas burning to be considered in L2 PSA	L2 PSA event trees shall model all effects of a flammable gas burning, for example: • load (pressure and temperature peak) on the containment walls and impact on their integrity, • local load (pressure and temperature peak) on some key equipment for the SAM, • effect of hydrogen leakage into auxiliary buildings or secondary containment, • effect of hydrogen for the containment venting system. Comment: recombiners or partial combustion reduce the amplitude of the pressure peaks in the containment; nevertheless, approximately the same energy is released in the containment and the impact temperature increase shall be considered carefully.
Spray system activation criteria	L2 PSA approach can help verifying the spray system activation criteria of SAMG while taking into account the benefit of reducing the containment pressure and the drawback of de-inerting the containment for number of situations.
Design options	L2 PSA approach can be used to determine, for a given NNP design, which type of system (passive, active, recombiners, igniters, inertisation, …) is the most efficient to reduce the risks induces by flammable gas.

Control of flammable gas	
Limits	All respondents confirm that L2 PSA is being used for assessment of the combustion issue. However, as can be seen from one of the contributions, for some NPP design, it is a real challenge to deliver a technically and scientifically satisfactory assessment.

Containment function (isolation, ventilation/filtration of auxiliary buildings …)	
L2 PSA applications	L2 PSA can be used: • to identify all scenarios leading to a containment isolation default (typically SBO situations) and check if appropriate measures are in place (typically efficient procedures to close manually some valves, DC electrical supply for some valves or additional AC DG), • to understand which functions are lost or degraded while the containment is filled with water, • to identify the measures specified for reducing the effects of having the core at the containment floor, • to identify, on the base of finite elements codes evaluations, the maximum allowable pressure and temperatures before the leak rate of the containment will increase drastically, • to check that procedures are in place to close the containment for all reactor configurations in shutdown states (considering the human reliability based on time availability and the complexity of the action and taken into account the environmental conditions), • to check that guidelines are in place to ensure the availability of the ventilation/filtration system of auxiliary buildings both in operating state and in shutdown states, • to understand when leakages increase or decrease and to examine consequences of any leakage (transfer of contaminated gas or liquid into auxiliary buildings, transfer of combustible gas …). It will be of importance to assess if the available strategies are qualified and valid for all kind of external conditions as: hot air, cold air, strong wind, heavy rains and fire inside and outside the plant.

Containment pressure control	
FCVS – Design	The L2 PSA provides information on: • scenario leading to containment venting, • safety improvement due to filtered containment venting, • causes of FCVS failure, • need, feasibility and inconvenience of multiple FCVS open/close cycles during severe accident phase. FCVS can be opened without any power supply in some plants. This may reinforce the independence between level 2 and level of defense in depth. L2 PSA can be used for argumentation. At design phase, L2 PSA can be used to support decision in FCVS (or alternative solution) construction and define functional requirements.

Containment pressure control	
FCVS – Additional use for severe accident prevention	PSA can be used to assess the benefit/inconvenient of early containment venting before the severe accident phase.
FCVS – Risks associated to flammable gas release	A critical issue for venting systems is the release of hydrogen. L2 PSA should be used to investigate the probability for challenging conditions, and for potential failures, including combustion in the stack where applicable. Considerable uncertainty is related to ignition probabilities. As shown by the Fukushima examples, ignition seems to be an almost random event. In a L2 PSA performed for a PWR many years before Fukushima [71], a significant probability for hydrogen burns in the venting system and associated ventilation systems has been identified.
FCVS and external hazards.	Most venting systems discharge through the stack, and they have a piping system leading to the stacks. Some filtering devices are heavy and need provision against earthquake. External hazards could be a significant threat to these components. L2 PSA should consider related sequences and consequences. Development of external hazards L2 PSA may give opportunities to check that the robustness of FCVS against external hazards is sufficient. The importance will be to understand when the external hazard gives loads that are in excess of the design of the safety systems and buildings.
Other systems: internal spray, heat exchangers, external containment spray …	For all types of solutions to remove heat from the containment, L2 PSA shall provide valuable information on : • the system availability for accidents coming from L1 PSA, • robustness for the defense-in-depth concept (are such systems still available if the situation has already conducted to fuel melt), • risks associated to flammable gas : how is the containment atmosphere flammability controlled (typically effect of the steam condensation by spray, by external wall spray …) ? • risk associated to the leakage on the circuits (typically the risk of sump contaminated water release). After having explored all situations, L2 PSA shall help obtaining guidance for a safe operation of these SAM.

Radioactive release issues	
Source term assessment	L2 PSAs, if they include both frequencies and amplitude of source terms, can be used to take into account SAM strategies for minimizing releases. For example: • reliability of the sodium hydroxide injection system, • pH control with passive system, • pH control during the latest phase of the accident. These issues need a source term modelling in L2 PSA.
Risk ranking using both source term assessment and frequency of accident.	Modelling the details of chemical processes (in particular related to Iodine) is still a very uncertain issue nevertheless, there is a consensus on some mitigation solutions or some order of magnitude : • reactor containment leakage rate shall be as low as possible, • any containment bypass or failure would lead to a cliff edge on accident consequences and must be prevented : this is the purpose of SAM strategies, • high pH reduces gaseous iodine, • particles deposition in the containment has a key influence on the amplitude of accident consequences (there may be several decades between the consequences of an early containment failure (during fission products release from the fuel) and a late containment failure (after particles deposition)), • containment heat removal with no containment venting is the best solution to limit accident consequences, • containment venting system shall include filtration device or limit cliff edge effects. Even if some details of chemical processes are uncertain, the L2 PSA results, including order of magnitude of the amplitude of radioactive release, can be useful for the risk ranking and the identification of SAM improvement. For example, L2 PSA can justify implementation of SAM strategies for some low frequency accident but large scale consequences.

SAM strategies for SFP	
Significance of Risk assessment	It seems that the probability for SFP meltdown as determined by L1 PSA is in general significantly lower than core meltdown. If this is confirmed, an in-depth L2 PSA for SFP may not be needed. Having said that, the following comments are due: If the SPF is located inside the containment (e.g. Germen PWR design), the containment function must be compromised before relevant releases can occur, and pertinent analyses are required, including SAM for protecting the containment. If the SFP is located outside the containment (e.g. French PWR design) pertinent analyses and potential SAM are less complicated.
Status of existing L2 PSA	In most countries L2 PSA for SFP has not yet been performed. Therefore, also the risk contribution by SFP and the risk reduction due to the recently implemented SAM cannot be evaluated.
Deterministic analysis	Deterministic analyses partly have been done and are still in progress to describe accidents in spent fuel pool taking into account the plant specificities (SFP inside or outside the containment, …) and the operation mode (normal operation, shutdown states, …).
SFP outside the containment	Several plant designs have SFP outside of the containment. Obviously such an arrangement tends to produce very high releases in case of an accident. L2 PSA and related SAM seem to be particularly justified under such circumstances.

Equipment qualification	
Risk assessment and assumptions in L2 PSA	L2 PSA developers shall consider all available information on structures and equipment survivability in severe accident conditions (qualification, design basis, beyond design studies …) and define justified assumptions. Uncertainties shall be considered, especially if the data come only from simulation tools (no experimental evidence).
L2 PSA application	L2 PSA can be applied to discuss which SAM strategies are less demanding (safer) for structures and equipment regarding environmental conditions during severe accident progression.

4.4 LINKS WITH EXTERNAL HAZARD

In the Nordic project FRIPP [43], a detailed assessment of recovery actions have been assessed and evaluated. The project was ended in the early 1990s with clear recommendations on how to handle a BWR with core damages during a period of 5 years or more.

Moreover, the aspects of explosions or fires from a beyond design basis event (e.g., aircraft impact) has been thoroughly addressed. Namely, following the events of September 11, 2001, the U.S. NRC issued a rule 10CFR50.54(hh)(2) requiring that “Each licensee shall develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire ….” [44].

4.4.1 EXTENSIVE DAMAGE MITIGATING GUIDELINES (EDMG)

The industry response included acquisition and staging of additional equipment, and development of mitigation guidance documents, called Extensive Damage Mitigating Guidelines (EDMGs). EDMGs were developed to contain predetermined strategies for dealing with more extreme damage states than those previously considered in EOPs and SAMGs. It was recognized from their conception that EDMGs could also be beneficial in mitigating “traditional” severe accidents (e.g., prolonged station blackout).

Two types of EDMGs were considered [45]: Initial Response EDMGs and Technical Support Center (TSC) Response EDMGS. The scope of these Initial Response EDMGs would include:

- An assessment of on-site and off-site communication in light of potential damage to normal methods available to the emergency response organization (ERO);
- Methods for notifications of the utility ERO/ERO activation to mobilize additional resources to the site in a timely manner;
- Basic initial response actions needed to potentially stabilize the situation or delay event degradation, including key mitigation strategies to help manage critical safety functions in the near term;
- Initial damage assessment to provide the ERO with information on plant damage conditions and status, as feasible.

In US all licensees developed plant-specific EDMGs which are intended to be utilized by licensed operators and technical staff. Unlike SAMGs, the guidelines and strategies contained in EDMGs are regulatory and subject to NRC inspection

4.4.2 DIVERSE AND FLEXIBLE COPING STRATEGIES (FLEX)

The NEI 12-06 guide [46] states that one of the primary lessons learned from the accident at Fukushima Dai-ichi was the significance of the challenge presented by a loss of safety related systems following the occurrence of a beyond-design-basis external event. The design basis for U.S. nuclear plants includes bounding analyses with margin for external events expected at each site. Extreme external events (e.g., seismic events, external flooding, etc.) beyond those accounted for in the design basis are highly unlikely but could present challenges to nuclear power plants. In order to address these challenges, the NEI 12-06 guide [46] outlines the process to be used by licensees, Construction Permit holders, and Combined License holders to define and deploy strategies that will enhance their ability to cope with conditions resulting from beyond-design basis external events. The objective of diverse and flexible coping strategies (FLEX) is to establish an indefinite coping capability to prevent damage to the fuel in the reactor and spent fuel pools and to maintain the containment function by using installed equipment, on-site portable equipment, and pre-staged off-site resources (three-phase approach). This capability will address both an extended loss of alternating current power (i.e., loss of off-site power, emergency diesel generators and any alternate ac source but not the loss of ac power to buses fed by station batteries through inverters) and a loss of normal access to the ultimate heat sink which could arise following external events that are within the existing design basis with additional failures and conditions that could arise from a beyond-design-basis external event. The FLEX strategies are focused on maintaining or restoring key plant safety functions and are not tied to any specific **damage** state or mechanistic assessment of external events.

The hazards have been grouped into five classes: seismic events; external flooding; storms such as hurricanes, high winds, and tornadoes; snow and ice storms, and extreme cold; and extreme heat. Each plant will evaluate the applicability of these hazards and, where applicable, address the implementation considerations associated with each. These considerations include protection of FLEX equipment, deployment of FLEX equipment, procedural interfaces and utilization of off-site resources. FLEX Support Guidelines (FSGs) would be similar in intent as the current 10CFR 50.54(hh)(2) rules. The future EDMG may rely upon FSGs. In the FLEX three-phase approach the installed plant equipment is used first, then transition from installed plant equipment to on-site FLEX equipment is made and finally additional capability and redundancy from off-site equipment is obtained. Plant-specific analyses will determine the duration of each phase.

In Europe, Spain has implemented FLEX. Related information can be found in [ASAMPSA_E D40.7vol3] [28].

4.4.3 RELIABILITY OF OPERATOR ACTIONS

The reliability of operator actions following an external initiating event is also a topic that has increased importance following the 2011 seismic-induced tsunami at the Fukushima Dai-ichi site in Japan. The study [47] summarizes the development of the current external events human reliability analysis (HRA) methods and guidance, and summarizes recent insights from applying this approach to seismic PSAs and briefly presents the EPRI report 1025294 [48]. The purpose of EPRI report 1025294 is to provide methods and guidance for the human reliability analysis of external events PSAs based on the current state-of-the-art in both PSA and in HRA modeling.

External events response actions are new post-initiating event operator actions used to mitigate the effects of an external event. Response actions consist of the following types of actions: terminating the impact of the external initiating event, mitigation of external initiating event consequences using the affected SSC, mitigation of external initiating event consequences using alternate components. Regardless of how the operator action is identified, the corresponding HFE must be defined for use in the external events PSA. The feasibly assessment of HFE needs to consider the following, at a minimum: timing, manpower, cues, procedures and training, accessible location & environmental factors and tools and equipment operability.

If the operator action is feasible, the analyst can proceed to perform either a screening or a detailed quantification. If the analyst finds the screening to be too conservative or limiting, the analyst is encouraged to apply the detailed HRA method. Once the HEPs have been quantified at the appropriate level, the operator actions and associated HEPs must be appropriately integrated into the PSA model.

5 COMPLEMENT OF EXISTING ASAMPSA2 GUIDANCE FOR SHUTDOWN STATES OF REACTORS, SFP, AND RECENT R&D RESULTS

The deliverable D30.7vol3 [29] is dedicated to complete the existing ASAMPSA2 guidance for L2 PSA and according to the latest state of the art for:

- shutdown states of reactor,
- spent fuel pool,
- other topics identified on the basis of the recent R&D.

The following chapters describe how these issues have been addressed in [29].

5.1 COMPLEMENT OF EXISTING GUIDANCE FOR SHUTDOWN STATES

5.1.1 INTRODUCTION

Traditionally, the risk associated with the nuclear power plants are assumed to be dominated by the full power operation, however as the safety significant events are increasing during shutdown states, the risk associated with shutdown states are assumed to be comparable to the full power operation. The overall plant status in shutdown mode may be very different from the full power mode. The containment hatch or the containment head (BWR) could be open, several systems might be offline, alarms and set points are different, activities in the plant could increase fire risk or cause power disturbances in the electric systems, redundancies in safety systems might be unavailable. All these issues tend to decrease the plant’s ability to cope with unforeseen challenges, which in some sense are compensated by the lower decay heat in shutdown states. The incidents during shutdown states could lead to substantial loss of reactor coolant through draining events, or to loss of heat removal. The performance of PSA for shutdown states can support the enhancement of the safety during plant outage, and may contribute to reduction of the outage duration.

ASAMPSA2 guidelines [4] provided summary on specific issues related to shutdown states, for instances, the structural barriers normally used to ensure nuclear safety is challenged by the maintenance and refueling activities, open containment and open RPV head during refueling, unavailability of the systems and equipment’s, success criteria for phenomena mitigation, presence of additional personnel, presence of additional heavy loads and flammable materials. For BWRs and VVER-440/213 plants, shutdown states present difficulties as part of the containment barrier (in Swedish BWR design it is ‘containment lid’) may be removed and the containment integrity cannot be easily recovered if an accident occurs. Although the decay heat level is low in shutdown states but it can still be substantial, at least in the beginning of the outage period.

The purpose of shutdown PSA is normally to analyze an outage period with maintenance activities and refueling; and calculate the risk of radionuclide release from potential sources such as (for light water reactors):

- Reactor core
- **Spent fuel storages (e.g. SFP)** (normally not included in PSA for nominal power or low power, more emphasis after Fukushima though)
- **Spent fuel handling facilities and pathways** (except for heavy lifts this is normally not included in Shutdown PSA)
- **Waste facilities** (normally not included).

Typically a NPP experiences various types of outages, for instance short unplanned (forced outage) for repair or “adjustment” and regular planned for refueling and maintenance. In principle each outage is unique with respect to plant conditions, plant configuration, time and transitions between different operational modes. In order to not having to analyze an “infinite” number of initiating events for each type of outage and configuration it is practice to use screening, classification & grouping of initiating events and plant configuration which is often an iterative process. By defining a limited number of plant operating states (POSs) where plant status and configuration are clearly distinctive or representative, the problem of performing a Shutdown PSA becomes manageable. Each POS has a defined set of ‘boundary’ conditions within which there would be no changes in major characteristics which are important for PSA modelling. A typical number of POSs considered in shutdown PSA varies from 10 to 20.

5.1.2 INTERFACE BETWEEN L1 AND L2 PSA

The interface between L1 PSA and L2 PSA is accomplished through the plant damage states (PDS). The PDS defines the plant state at the beginning of the core damage and the conditions necessary for conducting severe accident progression analysis. The general overview of the development of a typical L2 PSA is given in IAEA SSG – 4 [8]. If the status of containment system is not addressed in the L1 PSA, it needs to be considered by means of so-called ‘bridge trees’ (also called extended L1 event trees) of the interface between L1 and L2 PSA or as the first step of the L2 PSA. The extended L1 event trees must also consider all system conditions that are necessary in order to analyse the future accident progression. For example, L1 PSA event trees do not distinguish between RCPB high pressure and RCPB low pressure core damage, although RCPB pressure is important for determination of future accident progression.

Additional PDSs are considered for L2 shutdown states, which are based on following characteristics:

- Location of the fuel (core or spent fuel pool)
- Containment/SFP building integrity/isolation
- Type of initiating event
- Time when fuel damage occurs (related to the IE)
- Amount of water surrounding the fuel
- Status of the containment protection and mitigation systems
- Recovery of fuel cooling
- Amount of water in Refuelling Water Storage Tank (RWST) (PWR specific)

- Amount of water in the condensation pool (BWR specific)
- Primary system pressure boundary integrity, e.g.
    - Primary system intact
    - Primary system open but RPV head still mounted
    - RPV head dismounted
- Primary system pressure
- Status of high pressure and low pressure safety injection system.

The PDSs are grouped based on the POSs of the plant at power operation and during refueling outage, e.g.

- Group 0 – Full power operation
- Group 1 – POSs similar to full power operation. Both the RCS and the containment are normally closed.
- Group 2 – POSs in which the RCS is closed but the containment is open.
- Group 3 – POSs in which both the RCS and containment are open. The fuel is located in the reactor vessel.
- Group 4 – POSs which is a special case because the fuel is relocated to the SFP.

For each plant configuration the boundary conditions are not perfectly constant. They have to be defined as realistically as possible, or if this assessment can be made, conservatively. Obviously, the reduction of the decay heat with time slows down the degradation processes into the L2 phase, increasing the effectiveness of late mitigation processes and also modifying the source term activity composition to be released (Table 5). So, it is recommended to add new sub-states on the previous ones for a more realistic treatment of L2 PSA if they were not implemented during the L1 PSA (i.e. separating the POSs before and after refueling for some of the subgroups defined before, see Table 6). A list of source terms for a 900 MWe PWR expressed as percentage of the initial activity of the radioactive substances present in the reactor core is given in [49].

Table 5 Decay power fraction distribution per fission product groups for different times from scram (an example from Spain)

Time since reactor scram (h)	Decay power fraction	Distribution per fission product groups (%)						
		Nobles gases (Kr, Xe)	Main volatiles (Cs, Rb, I)	Metalloids (Te, Sb)	Noble metals (Mo, Tc, Rh, Ru)	Rare earth metals (La, Pr, Nd, Sm, Y, Zr, Nb, Am, Cm)	Alkaline earth (Ba, Sr)	Others (Ce, Np, U, Pu,..)
0	1	7	17.1	10.3	10.8	31.9	9.6	13.3
2	0.1	3.8	21.5	6.9	6.4	37	7.2	17.2
4	0.01	3.4	19.1	4.8	7	38.9	6.6	20.2
8	0.008	2.9	17.9	4.3	7	39.3	5.7	22.9
15	0.007	2.4	17.1	4.1	7.9	39.3	5.2	24
30	0.006	1.9	15.7	4	8.6	40.4	5	24.4
60	0.005	1.4	13.8	3.5	9.5	44.4	5.4	22

Table 6 Total decay power for different stages

(Values of a generic PWR 1000 MWe for a generic 25-days outage)

Stages	Initial time since reactor scram (days)	Decay power (MW)
	before / after refueling	before / after refueling
RPV closed - Hot shutdown	0.5 / 23	22 / 4
RPV closed - Cold shutdown (RPV filled)	0.75 / 21	20 / 4.2
RPV closed - Cold shutdown (middle loop)	1.25 / 20	15 / 4.3
RPV open – Maintenance works	1.5 / 18	14.5 / 4.5
RPV open - Refueling	4 / 13	12 / 7

In L2 PSA, one question is how the external hazard impacts the core melt process and the related plant response. In deliverable D40.7vol2 [27], it is stated for full power scenarios that the accident progression after core damage does not depend much on the external initiating event. This is true also for shutdown states. The only and obvious particular issue to be addressed additionally is the status of SSCs (e.g. containment structure, venting system and other systems that are important to mitigate radioactive release) after impact of the external hazard. An example may be mobile equipment, diesel generators, system for filling the containment with water and other SAMG measures.

5.1.3 ACCIDENT SEQUENCES WITH RPC CLOSED

When the RPV is closed, core melt accident phenomena are very similar to the sequences going on in full power mode. Therefore, the large body of guidance which is available for full power mode is largely applicable to shutdown mode with RPV closed as well. However, the decay heat level is lower compared to full power mode, and additionally in some states the coolant level in the RPV is reduced. The reduced coolant level in the RPV may reduce the amount of time until core uncovery; however core degradation and the further accident progression will progress more slowly than for power operation. Therefore, the core degradation does not require additional methods for analysis or modified methodology in general. There is no need for specific guidance from the L2 PSA point of view. Already the existing frameworks take into account e.g. loss of the containment ventilation isolation, or the failure of dedicated safety systems. The probably higher likelihood of such detrimental issues does not imply that additional or modified guidance is needed. It is simply required that the evaluation of such plant conditions and plant responses is correctly adapted to the shutdown state.

For example for generic 25-days outage, the RPV closed phase may represent the 25% of the total outage period. From this percentage, for example 5% is before refuelling and 20% is after refuelling, where the decay heat power is significantly lower. Therefore in these conditions, the higher risk would be assigned to the cold shutdown stage before refuelling due to unavailability of the high pressure mitigation systems and the reduction of water inventory inside the RPV in combination with the higher decay heat power. The severe accident phenomenology should be dominated by low pressure degradation processes, but probably without relevant impact on the L2 PSA risk as long as the containment is tight and maintained.

Therefore, in this report shutdown states with closed RPV are mentioned for completeness, but it will probably be sufficient to recommend proper application and adaptation (e.g. due to different decay heat levels) of the existing L2 PSA guidance to these plant conditions, and to draw the attention to the possibly difficult plant conditions impacting mainly on L1 PSA. At transient states when RPV is closed, but drivers of main control rods are unsealed, the total area for potential release of coolant and fission products from the reactor is in the order of 100 cm^2. In some PSAs, these transient conditions with closed but unsealed RPV are classified as states with open reactor. Also, special attention shall be devoted to the following issues:

- availability or recovery of safety systems (e.g. spray pumps, high pressure emergency core cooling systems) which can be under maintenance;
- the state of the containment i.e. it is opened and questionable to be closed (an additional question may be introduced to the containment event tree reflecting this issue);
- accident management systems.

5.1.4 ACCIDENT SEQUENCES WITH RPV OPEN

When the RPV is open, some of the L2 PSA issues become irrelevant compared to full power mode, while others come into existence. The following specific aspects shall be considered for shutdown L2 PSA for states with open RPV head:

- the list of the Initiating Events (IEs) is different in comparison with tight RPV (less possibilities for Loss of Coolant Accidents (LOCAs) due to pipe breaks, but drain of coolant due to human error could occur);
- containment state (usually containment is opened, and probability for closing to be assessed);
- containment or the reactor building status (e.g. VVER-440: in case of open reactor or SFP accident, the steam, hydrogen and fission products release into the reactor hall, which is outside the containment. The reactor hall is not a hermetic building, but the fission products can be settled in it. The status of the reactor hall (intact, failed, filtered vented) should be calculated in case of external event);
- availability and efficiency of safety related systems may be reduced;
- low decay heat power leads to increased available time before core damage;
- some phenomena could not occur (e.g. Direct Containment Heating (DCH), alpha mode failure, etc.);
- new IEs (specific for open RPV) shall be considered (e.g. heavy load drops, man-induced LOCA, etc.);
- different procedures for personnel, human errors of different extent/types/more relaxed attitude on one side (e.g. performance shaping factors), but more stress from the point of view of pressure to keep

deadlines for shutdown and to start in planned time (economic reasons), therefore work performed in parallel, frequently disturbing/causing errors of one group of personnel to other group;

- limited amount of instrumentation available (due to maintenance of power supplies, disconnection of sensors - e.g. water level, temperature etc.).

For most shutdown states with open RPV head, reactor vessel and SFP are connected by a large water pool in some reactor designs. L1 PSA as well as L2 PSA for shutdown states should consider interconnection between RPV and SFP (possibility to use common safety systems, common SAMG strategies, etc.).

The following issues obviously are less significant as compared to closed RPV head:

- high pressure core melt sequences with the large number of associated complications;
- retention of radionuclides inside the reactor coolant loop; and
- restoration of heat removal system.

Table 7 contains the list of specific issues for open RPV, together with remarks how they are addressed in the pertinent ASAMPSA_E guidance document D40.7vol4 [29].

Table **7 Specific L2 PSA issues for open RPV and associated guidance suggestions**

Specific L2 PSA issue for open RPV	Present status of guidance	Suggestion for improvement of guidance in ASAMPSA_E document D40.7vol4 sections
Fission product release from core melt in open reactor into containment or other building (e.g. reactor hall), including different chemical environment (air versus steam) of core degradation	No specific guidance exists for open RPV	Application of state-of-the-art integral codes with focus on flow paths above RPV in order to calculate potential air ingress. If air enters RPV, discuss impact on Zr oxidation and Ru release.
Heat load from the core melt in the open RPV to structures above (e.g. to the containment roof)	No specific guidance exists for open RPV	Application of state-of-the-art integral codes with focus on flow paths above RPV in order to calculate convection and thermal radiation to containment structures.
Influence of modified containment and plant status (e.g. open containment, mitigating systems not available, ventilation operation modified etc.)	Present status of guidance covers such issues.	Existing guidance must be properly applied or adapted, e.g.: open containment could be represented by previous analyses with containment isolation failure. Practical containment analysis proposal in appendix of D40.7vol4 [27].
Influence of an accident progression in open reactor on spent fuel pool (including accident management actions).	No specific guidance exists for coupled accident in RPV and spent fuel pool.	See Section 5.1.5 of D40.7vol4 [27]: 1. Suggest conditions (e.g. relevant probabilities, high consequences) which require analysis of simultaneous accident in RPV and spent fuel pool. 2. Suggest analysis method for simultaneous accidents in RPV and spent fuel pool (state-of-the-art integral codes cannot model two melting volumes).

Severe accident analyses showed several MELCOR analyses with open RPV, for PWR and for BWR as well. The analyses showed that steam evaporating from the core replaces the air from the atmosphere above the RPV to a very large extent. Secondly, the containment atmosphere hardly moves downward towards the hot core. Therefore, when the core melts, the atmosphere above the RPV is almost pure steam. Consequently, almost no air-driven oxidation of zirconium has been observed, and the amount of hydrogen produced with open RPV is similar to that with closed RPV. However, this finding is based on a few calculations, and has been made with a traditional nodalization of volumes above the core. It is recommended that each extended L2 PSA for accidents with open RPV performs several pertinent analyses with integral codes.

If there is a RPV bottom leak (e.g. at circulation pumps in a BWR) in parallel to the open RPV head or an open steam generator manhole, natural draft and air ingress into the melting core is possible. The presence of air can lead to accelerated oxidation of the zircaloy cladding compared to that in steam because it has a faster kinetic and 85% higher heat of reaction. The combined effects can give rise to an increased rate of core degradation. In addition, under oxygen-starved conditions, nitriding of the metals can occur, the resulting zirconium nitride is highly flammable and indeed can detonate on re-introduction of oxygen or steam as can occur during reflood [50].

Air ingress and its contact with fuel can result in significant releases of some fission products. This is especially the case for ruthenium which has the same radiotoxicity as iodine in short term through ^{103}Ru isotope and as caesium in medium term through ^{106}Ru isotope. Globally, the ruthenium release from the core may be 10 to 50 times higher than with steam only and the ruthenium tetra-oxide might represent a problem comparable with that of iodine. The safety impacts of such air ingress was analyzed in an AECL test [51] and more recently in an AEKI RUSET test [52] and also discussed at the PHEBUS Air Ingress Working Group.

Convection and thermal radiation from core melt in an open RPV may generate significant thermal loads to structures above the RPV, in particular to the containment itself. This is different from a closed RPV where the massive RPV head obstructs any direct impact from the melting core. The heat tends to accumulate at the containment top and its integrity may be threatened. The magnitude of this effect for severe accident sequences with RPV open has been addressed by a few analyses only. It is recommended that extended L2 PSA for sequences with open RPV carefully evaluate temperature evolutions above the RPV. Typical integral accident simulation codes may be applied for this purpose; however care has to be exercised in the nodalization of the flow paths above the RPV.

5.1.5 SIMULTANEOUS ACCIDENT PROGRESSION IN REACTOR AND SPENT FUEL POOL

The analysis of simultaneous accidents in the core and in the spent fuel pool is rather straightforward for sequences with station black out (SBO), which may probably be the highest contributor to such simultaneous melting. However, when power is (at least partly) available, human response in utilizing resources needs to be modelled. It may be reasonable to assume that all resources will be dedicated to that source (core or spent fuel) which tends to melt first. If this rescue attempt fails for the leading source, there is probably no resource left for the other source which melts later. However, no good practice can be identified for performing PSA under such conditions.

The following remarks address simultaneous accident progression in the reactor and in a spent fuel pool which is located inside the containment. For reactors with a spent fuel pool melting outside of the containment there may be dependencies on a system level in the field of L1 PSA (e.g. in availability of power or heat sink or human resources), but not related to containment issues.

The following considerations assume an existing containment event tree analysis for core melt sequences in the reactor core. The following generic considerations apply when a melt process inside the spent fuel pool (which is located inside the containment) has to be added to the analysis. The accident progression is structured into four phases:

- Before boiling starts in the SFP: no effect of the SFP on the accident evolution in the RPV.
- After boiling started in the SFP and before fuel damage in the SFP: steam from the SFP adds to temperature and pressure and also increases inertisation by steam.
- After fuel damage in the SFP begins and before MCCI in the SFP: the hydrogen generation in the SFP adds to the hydrogen from the core. Radionuclides from the SFP add to the radiological threat.
- After MCCI in the SFP begins: the generation of various gases influences the atmosphere. Radionuclides from the SFP add to the radiological threat.

These generic considerations apply to the full power state as well as for shutdown, for open and closed RPV.

The practical realization of these analysis principles proves to be difficult because none of the available accident simulation codes is capable of simulating more than one melting fuel entity. Therefore, at present it will be necessary to combine accident analyses from the core and from the SFP with the help of expertise. The task may become less complicated when considering that, e.g. in the most cases the fuel degradation in the SFP is expected to begin much later than the reactor core.

5.1.6 SUMMARY FOR L2 PSA IN SHUTDOWN STATES

For L2 PSA in shutdown states, two plant conditions are to be distinguished:

- Accident sequences with RPV head closed,
- Accident sequences with RPV head open.

When the RPV head is closed, core melt accident phenomena are very similar to the sequences going on in full power mode. Therefore, the large body of guidance which is available for full power mode is largely applicable to shutdown mode with RPV closed as well.

MAAP4 can be used to perform calculations; however, the assessment of open reactor cases is limited. For example, heat radiation and convection above the RPV, the air inlet into the RPV cannot be assessed appropriately using MAAP4. MAAP5 can assess SFP severe accidents and it can perform assessments for open reactor cases too. MELCOR has been applied by several organisations in the shutdown regime, also with open RPV head. Apart from a few cautionary warnings regarding heat radiation and convection above the RPV, MELCOR is applicable for such analyses.

When the RPV is open, some of the L2 PSA issues become irrelevant compared to full power mode, while others come into existence. The following issues obviously are less significant as compared to closed RPV head:

- high pressure core melt sequences with the large number of associated complications;
- retention of radionuclides inside the reactor coolant loop; and
- restoration of heat removal system.

The situation is different for aspects which do not exist or which are less pronounced in sequences with RPV closed. The following summarize such issues, such as:

- fission product release out of the RPV,
- containment issues.

Fission product release out of the RPV

In case of a core melt accident with the RPV open, two cases can be identified. The first case is the RPV bottom closed (always the case for PWR, not always for BWR accident scenarios). In this case, core uncovery can only occur due to coolant boiling. The second case is a RPV bottom leak (e.g. at circulation pumps in a BWR), which leaves the RPV open at top and bottom.

In both cases it can be imagined that air contacts the melting core, generating different conditions and releases compared to the pure steam atmosphere which is present in a closed RPV. However, present analyses do not indicate significant differences. This may be due to the fact that the air in the atmosphere near the RPV top and bottom is almost completely replaced by steam. This statement cannot be considered as a general rule, and pertinent analyses are recommended for such scenarios in a PSA.

Release fractions for closed RPV cannot be transferred to open RPV sequences. It is justified to assume that all fission products which are released from the degrading core will be transferred to the containment atmosphere. Moreover, in BWRs with closed RPV, the release in most accident sequences passes through the wetwell, thereby scrubbing large fractions of the radionuclides. This significant mitigating feature also does not exist when the RPV is open.

Containment issues

It can be considered likely that hatches and airlocks are or will be closed when critical conditions in the containment begin. However, since the consequences of an open containment are very severe, a PSA should quantify the probability for an open containment. The flow path through the reactor building and auxiliary building or turbine hall or ventilation systems – whatever is applicable – to the environment has to be considered for an open containment. Hydrogen threats in the release path and deposition of fission products are the most relevant aspects in this regard. However, a detailed analysis of such buildings and flow paths and systems may be beyond the possibilities of most PSA. If there is no deviating evidence, it seems to be prudent to assume that severe hydrogen combustion occurs inside the buildings - see the Fukushima experience – and that a large release path to the environment will be opened.

In the context of an extended PSA also internal and external hazards should be taken into account which may affect the possibility to close the containment.

It is recommended that extended L2 PSA for sequences with open RPV carefully evaluate temperature evolutions in structures above the RPV. Heat radiation as well as convection out of the open RPV shall be considered. Typical integral accident simulation codes may be applied for this purpose; however care has to be exercised in the nodalization of the flow paths above the RPV.

5.2 COMPLEMENT OF EXISTING GUIDANCE FOR SPENT FUEL DAMAGE (SFD)

5.2.1 INTRODUCTION

For this section, the heading “spent fuel damage” (SFD) has been chosen, in addition to the more common “spent fuel pool”. The expression is motivated by the fact that apart from the RPV not only a spent fuel pool filled with coolant may experience fuel damage, but also dry storage or fuel handling systems. In the latter, a prominent event occurred in the Paks plant in Hungary, where a unique fuel cleaning system failed to properly cool the fuel, causing severe damage to several fuel elements. However, to limit the scope of discussion, this section will be more focused on fuel damage in spent fuel pool (SFP) only.

According to definition, L2 PSA deals with fuel degradation, considering all issues which occur before fuel degradation belong to the L1 PSA. Therefore such important items such as the vulnerability of the spent fuel pool against external events or the possibility of emergency measures to recover cooling before degradation are not discussed here.

The SFP storages used nuclear fuel from the nuclear reactor. The pool is typically situated near the reactor either in the containment or in the reactor building, or in a nearby building. During the refueling outage, part of the fuel, or in some cases even all fuel, is offloaded to the SFP. There can be more fuel in the SFP than in the reactor core, so that more hydrogen and more long-lived radionuclides can be released, it will take longer time though. Also, the fragility analysis of SFP should cover the 'likelihood' that cooling of the fuel is affected by the amount of fuel in SFP in different POSs and the amount of fuel increases over time.

In the past, the SFP has not been considered a high safety risk for operating plants. Studies generally showed that the frequency for an accident involving the SFP was low compared to the contribution of the core to the fuel damage frequency. It could be considered that this is again demonstrated in Fukushima Dai-chi where three cores melted, but no damage in a SFP occurred. Nevertheless, the anxiety during the Fukushima Dai-chi accident for the SFP in block N°4 was extremely high and the SFP have only been stabilized thanks to emergency recovery actions.

In recent years, it has been concluded that there is a need to better understand the risks associated with the SFPs. EPRI presented their development and pilot application of a generic framework and methodology for conducting PSA for SFPs at BWR plants with Mark I or II containment designs [53]. A similar methodology is now being developed for PWRs and the results are presented in an EPRI report [54], however there are still more guidance needed for L2 PSA for SFP, e.g. on phenomenology for fuel melting in air environment.
The European Utility Requirements (EUR) requirements regarding SFP are somewhat more general, for instance, EUR Chapter 2.17 [55] states that:
"C: The PSA shall check that potential radioactive releases from the spent fuel storage pool, from the spent fuel handling facilities and from the radioactive waste storage tanks can be reasonably neglected, due to their comparatively low magnitude and to their low frequency."

The following section addresses those issues which are specific for SFD events, and which need consideration in guidance for an extended PSA.

5.2.2 ISSUES RELATED TO SFD WITHIN EXTENDED L2 PSA

Inventory of SFP

In contrast to the reactor core which has a very well defined configuration, the SFP may have very different inventories during the lifetime of a plant. It could go from almost zero inventories in new plants to an inventory at the design limit for old plants or during core unloading in shutdown modes. L2 PSA in SFP needs guidance how to define the initial loading, residual heat generation and radionuclide inventory inside the SFP.

Criticality in SFP

Depending on the SFP design and its inventory, it may be imagined that criticality occurs during an accident sequence. Guidance is needed whether and how to address this issue in L2 PSA.
Different initial conditions in core and SFP

Reactor-SFP interactions

When considering core and SFP, one of the two components may be in a degrading condition (pertaining to the realm of L2 PSA), while the other component is still undamaged (pertaining to the realm of L1 PSA). This is the traditional approach in L2 PSA, where core damage is investigated assuming undisturbed conditions in the SFP. However, both components may be linked by systems (e.g. cooling systems – the most obvious example is SBO which affects both components) and by boundary conditions (e.g. containment atmosphere). Accident progression or successful SAM in one of the components can affect the other component in one way or another. Guidance is needed how to address this “mixed” L1/L2 PSA level.

When both core and SFP are degrading, this is clearly an issue to be dealt with both in L1 and in L2 PSA. It seems that the increased risk associated with interactions between the reactor and containment systems and the SFP should be treated in an integrated way.

Containment-SFP interactions

When the SFP is located inside the containment, the events during SFP degradation will threaten the containment. Most existing L2 PSAs are limited to core damage accidents, and to the related containment threats (e.g. due to hydrogen, pressurization, temperature). An important reason for this limitation is related to mission time. However, the Fukushima events demonstrated that this argument may not be convincing.

Melting in a SFP will cause different threats - an example is the heat load from the melting pool to structures above the pool. Guidance is needed how to take these different threats into account in extended L2 PSA.

Moreover, the influence of containment phenomenological effects on SFP risk should be addressed. There are a number of postulated effects related to severe accident progression and consequential containment challenges that can influence the risk evaluation of the SFP. Effects of reactor accident progression on SFP accident mitigation include phenomena, accident characteristics and containment failure, e.g. un-isolated break outside containment or interfacing system LOCA during at-power operation state, transfer of contamination and hydrogen.

SAM

SAMs are discussed in deliverable D40.7vol3 of the ASAMPSA_E project [28]. Particularities for SAM in SFP shall be mentioned there (e.g. limited accessibility to SFP due to high radiation when water level gets low or in case or leakage from the reactor building).

Other fuel locations than SFP

Depending on the plant design, apart from the SFP there may be other locations where fuel is present e.g. cleaning loops, fuel handling systems, dry storage, and transport casks. It is plant specific whether events in these locations can lead to fuel damage in the related system, or whether an event in these systems can trigger other failures and fuel damage in other locations, however this guidance is focused on fuel damage only in SFP.

Shared SFPs

Not each reactor is assigned to one dedicated SFP. For example, two reactors may share a single pool. Also, a single reactor may store fuel in more than one pool, or two reactors at the same site may move fuel between both pools located in common or separated buildings that may or may not be connected. Thus, guidance is needed to address the differences and potential interactions between shared SFPs in an integrated way.

Density of spent fuel racks

Some pools contain high-density spent fuel racks which allow multiplying the number of stored assemblies. In such a case the consequences of fuel damage may propagate too much larger populations of fuel assemblies. These racks also have much higher overall decay heat and larger fission product inventory. Therefore, the density of the fuel racks should be considered.

Spent fuel building ventilation

The spent fuel building ventilation flow rate is important in determining the overall building energy balance. Airflow through the building is an important heat removal mechanism. It also provides a source of oxygen for zirconium oxidation, and for diluting hydrogen. The ventilation system has an effect on the magnitude, timing and height of the fission product release. The ventilation system with aerosol and iodine filters can decrease the released mass, but the efficiency of the filters shall be checked with considerations to the special accidental circumstances.

Particular heat transfer mechanisms for spent fuel pools

There are several heat transfer mechanisms that can influence the cooling of spent fuel during various postulated severe accident scenarios. These include:

- convective cooling to the surrounding water,
- steam cooling from surrounding steam generated by boiling coolant,
- conduction through the ends of the fuel rods,
- radiation cooling.

The degree of success associated with different heat transfer mechanisms depends on the configuration of the SFP, rack/canister design (e.g., closed or open lattice), density of the fuel assemblies, arrangement of the hottest bundles within the SFP lattice and the SFP water level.

Structural integrity of fuel racks

For recently discharged fuel or for severely restricted air flow (e.g. high density spent fuel racks) the exothermic oxidation reaction is predicted to be very vigorous and failure of both the fuel rods and the fuel racks is expected. The steel racks may not be able to maintain structural integrity because of the sustained loads at high temperatures. Thus, a large fraction of fuel rods would be expected to fall to the bottom of the pool and will tend to heat the adjacent assemblies, which appears to be an additional mechanism for oxidation propagation.

5.2.3 IDENTIFICATION OF INITIATING EVENTS

The loss of pool cooling initiating event is probably the most likely initiating event. It can be caused by the failure of electrical power, of pumps or valves, piping failures leading to flow diversion, failure of heat exchangers, failures in cooling service water system etc.

Loss of SFP coolant inventory includes draindown events and structural failures. Draindown events can occur due to breaks or alignment errors on pipes connected to the SFP. In refuelling state, where the transfer canal is opened, they may also occur due to breaks or alignment errors on pipes connected to the reactor building pool. In addition, draindown events may lead to loss of SFP cooling if the SFP water level is lowered below the suction lines of the SFP residual heat removal system.
A loss of coolant inventory can also be caused by SFP structural failures following for example an earthquake. A seismic event may also lead to an initial limited loss of coolant inventory due to sloshing. Another type of events with the potential of causing structural failures is reactor-related phenomena.

Reactivity accidents of interest are any events where criticality can lead to insufficient fuel cooling and thereby fuel damage. Criticality is prevented by dispersal of the fuel assemblies and equipping the pool storage racks with neutron absorbers. The impact of a reduction of boron concentration in the SFP should be analysed. In addition, fuel handling accident such as a drop (of a fuel assembly) or incorrectly placed fuel should be evaluated as potential initiators.

5.2.4 ACCIDENT SEQUENCE ANALYSIS

The accident sequences analysis is performed in a similar way as in the PSA for the reactor core. The analysis should describe scenarios that can lead to the defined consequence. It should address system responses, operator actions, phenomena and also dependencies that can impact the availability of the mitigating systems.
Specific event trees should be developed for the SFP. End state in the SFP L1 PSA is fuel damage. In some SFP PSAs the frequency for boiling in the SFP is assessed separately. Boiling in the SFP would lead to a continuously decreasing water level. It could also affect the environment in the spent fuel pool and building and could for example make it impossible to perform necessary manual actions. The radioactive release should be categorized based on magnitude and timing to constitute appropriate L2 end states.
Since there are limited barriers to contain a radioactive release from the fuel in the SFP if the pool is not located in the containment, it might be possible to integrate the L1 and L2 event trees in this case.

Combustible gas deflagration

Hydrogen generated by spent fuel as a product of Zircaloy water reaction could accumulate in the Fuel Handling Building or Reactor Building in a combustible mixture. The subsequent combustion or deflagration may result in significant collateral damage such that mitigation equipment, sprinkler outlets, even structural integrity of the SFP may be compromised. In addition, potential generation of Carbon Monoxide (CO) may occur which has similar deflagration characteristics as hydrogen. Hydrogen management concepts developed for hydrogen release from a degrading core (e.g. autocatalytic recombiners, igniters) need to be checked for their efficiency in SFP.

Safety assessment of spent fuel pool during decommissioning

Spent fuel from the reactor vessel is removed at an early stage of decommissioning of the plant to SFP. Its timely removal from the installation simplifies monitoring and surveillance requirements on plant safety systems. For a defueled reactor in decommissioning state, public risk is predominantly from potential accidents involving spent fuel.

5.2.5 THERMAL HYDRAULIC CALCULATIONS AND SUCCESS CRITERIA

Thermal hydraulic calculation is needed to determine the accident progression parameters. These should be used to support realistic system success criteria, to provide timing to assess necessary operator actions and to provide the fission product release magnitude and timing. The calculations provide information on the following:

- Time to boiling;
- Time to fuel uncovery;
- Time to fuel damage;
- Time to SFP structure breach;
- Time to penetration of concrete around SFP;
- Source term magnitude and timing.

Success criteria should be defined for different configurations and different decay heat loads. Calculations should be performed based on the amount of fuel that normally is replaced during a refuelling outage and should also be performed for a full core offload if this will be put into practice.
Calculation can for example be performed with MAAP5, which includes a spent fuel pool model capable of modelling severe accidents in the SFP, or MELCOR.

5.2.6 HUMAN RELIABILITY ANALYSIS

No change in Human Reliability Analysis (HRA) method compared to the PSA for the reactor vessel is required, but a number of additional operator actions will need to be analysed in connection with the SFP PSA. These actions include:

- Handling of fuel

During a refuelling outage fuel is being transferred to and from the SFP. Identified fuel handling accidents that could cause for example criticality should be analysed.

- Heavy load operations

Analysis of dropped heavy load, for example a fuel cask, should be performed. It could lead to a structural failure of the SFP or cause damage to fuel already in the pool. A dropped object could result in closer spacing of fuel assemblies which could create the potential for criticality.

- Manual alignment of possible cooling and make-up systems

Non-automated cooling and make-up systems available for the SFP should be analysed. There might also be systems not originally intended for SFP cooling or make-up that can be used for this purpose. In these cases operator instructions might be missing. Also, since the operators in the most cases will have long time for their actions it can be questioned if the HRA is necessary for those long-term scenarios.

Typical for the manual actions associated with the SFP is that they may occur over long time frames and that they may need to be performed during harsh environmental conditions. Various calculations have been performed regarding the radiation level during severe accident events in a SFP [56]. In the following example, only the direct gamma radiation from fuel is accounted for. Calculated radiation levels from a drained SFP one meter above the level of the floor results in 14,000 rem/hr. Even out of direct sight of the spent fuel, the radiation dose rates from gamma rays scattered by the air, roof and walls are over a hundred rems/hr [53].

5.2.7 FUEL DEGRADATION PROCESS IN SPENT FUEL POOLS

At first sight, it seems reasonable to assume that air could be present when melting occurs in the open spent fuel pool, in contrast to the closed RPV where no air access is possible. If air were present instead of steam the chemistry of the degradation process would change: Zr would be oxidized by Oxygen from air instead of by Oxygen from water. The thermal output of Zr-air oxidation is higher, but on the other hand less or no hydrogen would be produced. Volatile Ruthenium oxide could be produced by air impact, which is very relevant in terms of radiological effect. However, analyses performed with MELCOR under various conditions for loss of heat sink show that the previous evaporation of the large amount of water from the SFP would almost completely generate a steam atmosphere with little air having access to the degrading fuel. There are only two potential scenarios which may lead to significant oxidation by air: A rather fast loss of coolant from the SFP (can be practically excluded in some SFP designs), or an extremely low evaporation from the SFP with most of the steam being condensed before fuel degradation. However, the latter sequence may last for weeks, and have such a low energetic level that even without water the SFP may not heat up to the threshold for chemical reaction.

During fuel degradation in the SFP (before Molten Corium Concrete Interaction (MCCI) begins) the temperatures in some of the sequences are lower than in RPV accidents during normal operation. Therefore less radionuclides are released from fuel. However, after MCCI has started the release fractions from fuel reach levels which are known from accidents in the RPV.

There are no specific accident simulation codes available for spent fuel pool degradation. Therefore, the codes for reactor core degradation need to be applied. The models provided by the codes need to be adequately modified in order to achieve meaningful results. Some experience by ASAMPSA_E partners exists with the application of the code MELCOR, and the related issues are as follows:

- Modelling of the spent fuel can be done straight forward using the available models for representing the core. Of course the number of fuel elements, their decay heat level and fission product inventory have to be adapted. If the geometry of the fuel element array is significantly different from a rectangle or cylinder, this will introduce uncertainties.

- If there are specific supporting structures inside the spent fuel storage, their representation may be difficult to achieve.
- The RPV which does not exist in the spent fuel pool has successfully been represented by a very thin metal sheet which in reality is the metal liner on the spent fuel pool bottom and walls.
- There is concern that air ingress into the pool might change some aspects of the events. However, in loss of heat sink accidents the evaporation of the spent fuel pool water will create so much steam and replace the air that such concern is not relevant. (Leakage accidents with a fast loss of coolant accidents have not been simulated by this partner).
- Core-concrete interaction and the destruction of structures below the spent fuel pool bottom could be calculated similar to core melt accidents.
- Heat readiation from the degrading fuel to structures above needed particular additional modelling.
- The modelling of fission product release is certainly not perfect. However, there is an inherent mechanism which stabilizes the results: If the initial fuel degradation provides little release, more nuclides remain and are relocated to the core concrete interaction phase. They will then be released there – and vice versa. Therefore, it is expected that the uncertainty in the total released amount is limited and acceptable.

Apart from this experience by an ASAMPSA_E partner, at the time of drafting the present report an international benchmark on this issue is in its final phase (http://s538600174.onlinehome.fr/nugenia/portfolio/air-sfp/). A final report should be available very soon. Several codes have been applied by different partners in order to calculate loss of cooling and loss of coolant accidents in a spent fuel pool. It seems that the differences among the results are significant – however, the analyses did not cover the full scenario, and fission product release was not discussed. In summary, this benchmark demonstrates that most of the available codes can be applied in principle, but that the lack of experience and precision is significant.

5.2.8 HYDROGEN ISSUES IN SPENT FUEL POOL MELTING

Several analyses performed with MELCOR for loss of heat removal scenarios show that the evaporation of the large amount of water from the SFP would almost completely generate a steam atmosphere with little air having access to the degrading fuel. Consequently, in such scenarios hydrogen generation by steam in a melting SFP is an issue. In addition, large amounts of hydrogen will be generated when concrete erosion occurs.

Further discussion of the issue requires to distinguish SFP which are inside the containment (e.g. German PWRs), and SFP which are outside the containment (e.g. French PWRs). Almost all plants worldwide have SAM and/or specific systems to cope with RPV core melt accidents, including the associated hydrogen issues. Therefore, hydrogen generated in a SFP inside the containment is in principle covered by these arrangements. For example, Passive Autocatalytic Recombiner (PARs) installed in German PWRs recombine the hydrogen produced by a SFP accident until all the oxygen is used up. Later, when still more hydrogen is generated without oxygen available for recombination, the hydrogen accumulates inside the containment and becomes a threat when it is released from the containment – either by purpose through the venting system, or accidentally through leaks.

The situation is different if the SFP is located outside the containment in the reactor building or in specific buildings (e.g. French PWRs). There, in general no provisions for hydrogen challenge are available. Consequently, it has to be assumed that a significant risk of deflagration or even detonation exists. Furthermore, the barriers between the SFP and the environment are less reliable than the containment. Altogether, there is a high probability for catastrophic releases if a SFP outside the containment begins to melt.

5.2.9 HEAT LOAD DUE TO SPENT FUEL POOL MELTING

Several analyses performed with MELCOR under different conditions show that the heat load from the SFP upwards to structures above (containment dome, or roof of reactor hall) is significant. Depending on assumptions about heat radiation, nodalization, and accident sequence maximum temperatures of up to 1000 K have been calculated pessimistically in the upper atmosphere and in the containment structure. This is by far beyond design temperature.

Based on these analyses, the following comments are due:

- Heat transfer from melting SFP (convection and radiation) seriously affects the temperatures of structures above the SFP.
- The models for thermal radiation from a melting SFP to the surrounding structures need validation and probably improvement.
- Calculating thermal convection upwards from a melting SFP is a challenge for state-of-the art lumped parameter codes. Coarse nodalization could, in principle, miss local plumes of hot gas.
- Very high temperatures will be experienced not only by the upper structures, but also by the upper atmosphere and by several components and systems in the vicinity (e.g. crane, refuelling machine, penetrations, doors, venting system, building ventilation, roof, isolation valves, cables etc.). There seems to be a significant probability that everything which is located above the melting SFP will fail.
- Only for low decay heat inside SFP, where uncovering of the fuel assemblies is terminated before their heat-up, air oxidation can occur after steam concentration has been depleted.
- It might be helpful to initiate filtered containment venting early in case of severe accident inside SFP in order to prevent high containment loads and high venting temperatures later. In any case, it is very likely that severe accident sequences run into venting of building where SFP is located.
- During fuel degradation in the SFP (before MCCI begins) the temperatures are lower than in RPV accidents during normal operation → less release of radionuclides from fuel. After MCCI has started, the release fractions from fuel reach levels which are known from accidents in the RPV.
- With full loading of the SFP, the fuel melt layer thickness (including material of the racks) at the bottom of the SFP is in the order of 1 m. Such a thick melt layer would probably develop heat transfer mechanisms (convection, steel layer on top) which enhance lateral erosion. Depending on the NPP design, this may lead to different sequences than vertical erosion. In case of the German PWR design, radial melt-through of the

containment may be possible. If, on the other hand, corium penetrates through the bottom of the SFP into the sump region, MCCI could be stopped because of the large amount of water in the sump, and because the melt spreads on bigger areas.

- For normal loading of the SFP (i.e. in normal operation with RPV fully loaded) the accident evolution in the SFP is much slower than in the RPV.

5.2.10 RELEASE PATHWAYS TO THE ENVIRONMENT IN CASE OF SPENT FUEL POOL MELTING

The mechanisms which influence the transport of mobile radioactive species from the spent fuel pool through building volumes to the environment are, in principle, the same as those which command the transport of material from the core. Therefore, the codes which are used for release after core melt accidents can be used also for spent fuel pool accidents. Of course, the usual care has to be applied when doing the analysis and when interpreting the results, because the codes still have deficiencies, and the users must be well qualified. But there are no particular phenomena involved compared to core melt accidents.

Obviously, release paths from the SFP to the environment are different depending on the location of the SFP i.e.:

1. the SFP is located inside the containment,
2. the SFP is located outside the containment.

<u>If the SFP is located inside the containment</u>, the potential release paths to the environment are almost the same as for core melt accidents in the RPV. Depending on the specific design an additional release path may be possible as follows: After penetrating the concrete wall or bottom of the SFP, the molten debris may come into contact with the containment wall and penetrate it. This would lead to a unique containment failure mode. However, from a general perspective this is just another type of late containment failure.

<u>A BWR is taken as example for a SFP which is located outside of the containment</u> in the upper floors of the reactor building. BWR reactor buildings have a predefined release path in case of loss of coolant which is directed into the turbine hall. Also on top of the turbine hall flaps are provided for release of steam into the atmosphere. In case of a melting SFP in the reactor building, this is the preferred release path as well. Since the volumes of reactor building and turbine hall are very large, significant deposition of aerosols will occur there, mitigating the environmental impact. A severe additional concern exists with regard to hydrogen generation from the melting SFP. This hydrogen will enter the reactor building atmosphere, and it is very likely that hydrogen combustion occurs inside the building. Depending on the building design (e.g. concrete or light construction like Fukushima Dai-chi) and on specific issues like ventilation ducts or doors, a more direct release path to the atmosphere may open up.

Another example of the SFP located outside the containment is the pressurized water reactor VVER 440. Some interesting outcomes were obtained from analyses of three different types of severe accident scenarios (Heavy load drops, SFP leakage, Loss of SFP cooling system) in the SFP for VVER 440. A very important question is, if any decontamination factor for released fission products can be considered. In case of VVER 440 reactors, the fission products are released directly into a reactor hall, if

1) ventilation flow above the SFP is turned off and

2) a cover of the SFP is removed for fuel handling.

According to the SAMG in case of SFP accident, the reactor hall should be closed and the filtered vent system of the hall should be used. The human errors or system failures should be taken into account, and the fission product behavior in the large non-hermetic compartment should be calculated by a lumped parameter code. The nodalization of this code should be defined according to 3D code calculations. Significant deposition of aerosols can occur in the reactor hall due to gravitational settling.

5.2.11 CORRELATIONS BETWEEN ACCIDENT PROGRESSION IN SPENT FUEL POOL AND IN THE REACTOR VESSEL

Most experience in L2 PSA exists for analysis of accidents in the RPV in normal operation, not taking into account any correlations between reactor core and SFP. In an extended PSA, such potential correlations should be explored, according to the following reasons:

- Core melt occurs only if the plant status is in severe disorder. It seems difficult to prove that the SFP systems would not be affected by such disorder. This is especially the case for external hazards. For such scenarios, it should be considered that subsequent SFP melting may significantly increase the source term.
- Core melt phenomena will threaten the containment. This is evaluated in most PSA, and in general there is a satisfactory reliability of the containment for mitigating the consequences. However, additional loadings due to SFP steam generation and melting processes will add an additional challenge. Therefore, it is conceivable that containment and its systems (e.g. venting system) would be able to manage a core melt accident, but not a combination of core melt and SFP accident. This could be considered as a cliff-edge effect.
- Depending on the plant design, it is conceivable that melt-through of the SFP structure could affect systems and components which are important for safety. This is, for example conceivable in some PWRs where radial melt-through of the SFP could damage the containment. For reactors where the SFP is outside the containment but inside the reactor building, melt-through of the SFP could lead to fuel melt impact onto the containment outside, or onto safety systems in the bottom of the building.
- MCCI in the SFP could induce an accident inside the RPV if, for example, the SFP is located inside the containment, and melt from the SFP gets into the containment sump. This might damage ECCS components (e.g. blocking of filters), leading to failure of core cooling.

5.2.12 CORE CONCRETE INTERACTIONS FOR SPENT FUEL POOL ACCIDENTS

Depending on the amount of spent fuel and rack material, the melt level in the SFP can become significant. Such a thick melt layer would probably develop convection patterns which predominantly transfer the heat to the upper edge of the melt. In addition, a metal layer could float on top of the melt and also create local high lateral heat fluxes. On the other hand, vigorous bubbling due to fuel-concrete interaction would tend to equalize heat fluxes. In summary, it has to be taken into account that local peak heat fluxes at the upper edge of the melt pool in the SFP can exist.

Therefore, when considering consequences of MCCI in SFP melt accidents, melt breakthrough has to be assumed in various positions. Depending on the plant design, different consequences can develop, like damage to the containment, or damage to systems in the vicinity. If circumstances are unfavorable, an accident in the SFP could induce an accident inside the RPV as well. This could occur, for example, if the SFP is located inside the containment, and melt from the SFP gets into the containment sump. This might damage ECCS components (e.g. blocking of filters), leading to failure of core cooling.

Obviously, when the SFP is located away from RPV and containment in a separate building, such dependencies as mentioned above can probably be excluded.

5.2.13 CRITICALITY IN SPENT FUEL POOLS

10 CFR 50.68: part (b)(2) states that "...(k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level."

The US NRC report [57] identified the potential scenarios that could lead to criticality in decommissioned SFPs, which are discussed as below:

- A compression or buckling of the stored assemblies due to heavy load drop (e.g. fuel cask) could result in closer spacing (geometry) in SFP and could lead to potential for criticality. However, this scenario is mitigated by using fixed neutron absorber plates in high density PWR or BWR racks and soluble boron in low density PWR racks. But compression of a low density BWR rack could lead to a criticality since BWR racks contain neither soluble nor solid neutron absorbing material. The reason is low density BWR fuel racks use only geometry and fuel spacing to maintain subcriticality and high density racks utilise both fixed neutron absorbers and geometry to control reactivity.
- For BWR SFPs, if the stored assemblies are separated by neutron absorber plates (e.g. Boral or Boraflex), loss of these plates could result in a potential for criticality. But for PWR SFPs, soluble boron is sufficient to maintain subcriticality and absorber plates are generally enclosed by cover plates (stainless steel or aluminium alloy).

In the USA NPPs, boraflex has been found to degrade in SFPs because of gamma radiation and exposure to the wet pool environment. Therefore many licensees replaced the boraflex racks in their SFPs or reanalysed the criticality aspects, assuming no reactivity credit for boraflex.

From the neutronics point of view, SFPs are designed to be subcritical systems [58]. The amount of fissile material contained in an SFP, as well as its geometrical configuration, varies from unit to unit; special care in the arrangement design is therefore always taken in order to maintain a given subcriticality margin which guarantees criticality safety under both operational and accident conditions for the entire lifetime of the SFP itself [59].

5.2.14 SUMMARY FOR L2 PSA FOR SPENT FUEL POOLS

The ASAMPSA2 [4], [5], [6] guidelines provide the best practice guidelines for the performance and application of L2 PSA development for the Gen II PWR, Gen II BWR L2 PSAs and extension to Gen III and Gen IV reactors, however discussion on SFP guidance is not included in the scope of ASAMPSA2, so the SFP L2 PSA discussion is complemented in this present report.

In the past, the SFP has not been considered with a high safety risk for operating plants. Studies generally showed that the frequency for an accident involving the SFP was low compared to the contribution of the core to the fuel damage frequency.

Nevertheless, the anxiety during the Fukushima Dai-chi accident for the SFP in block N°4 was extremely high and has increased the interest of the nuclear safety community for the SFP issues.

There are some challenges in considering SFP PSA, for instance reactor-SFP interactions, radioactive and hydrogen release, shared support system between reactor and SFP, maintaining SFP cooling and human actions/responses in these scenarios.

Table 8 contains a list of the issues which have been developed within deliverable D40.7vol4 [28].

Table 8 Specific L2 PSA issues for spent fuel pool and associated guidance suggestions

#	Specific L2 PSA issue for spent fuel pool	Suggestion for improvement of guidance in ASAMPSA_E as explained in D40.7vol4 [28]
1.	Fuel degradation process, including energy and fission product release from melting spent fuel into containment	There is concern about the impact of air on the fuel degradation process. However, this may not be relevant for loss of heat removal scenarios. Several analyses performed with MELCOR show that in such scenarios the previous evaporation of the large amount of water from the SFP would almost completely generate a steam atmosphere with little air having access to the degrading fuel. It is recommended to further substantiate this statement by performing additional analyses. During fuel degradation in the SFP (before MCCI begins) the fuel temperatures in some of the sequences are lower than in RPV accidents during normal operation. Therefore less radionuclide are released initially. However, after MCCI has started, the release fractions from fuel reach levels which are known from accidents in the RPV.

#	Specific L2 PSA issue for spent fuel pool	Suggestion for improvement of guidance in ASAMPSA_E as explained in D40.7vol4 [28]
2.	Hydrogen generation in spent fuel pool and its distribution in containment	As mentioned earlier for loss of heat removal sequences, above the SFP there is a steam atmosphere with little air having access to the degrading fuel. Consequently, hydrogen generation by steam in a melting SFP is an issue. In addition, large amounts of hydrogen will be generated when concrete erosion occurs. Hydrogen generated in a SFP inside the containment is in principle covered by the arrangements foreseen for core melt accidents. If the SFP is located outside the containment in the reactor building or in specific buildings, in general no provisions against hydrogen challenge are available. Consequently, a significant risk of deflagration or even detonation exists. Furthermore, there are less reliable barriers between the SFP and the environment. Altogether, there is a high probability for catastrophic releases if a SFP outside the containment begins to melt.
3.	Heat load from the melting spent fuel to structures above (e.g. to the containment roof)	Several analyses show that the heat load from the SFP upwards to structures above (containment dome, or roof of reactor hall) is significant. Analytical models should include thermal radiation and apply a suitable nodalization to model convection. Consequences of the high thermal load should be considered (e.g. reduction of containment pressure bearing capacity, impact of hot gas on venting system, induced fires).
4.	Release pathway for radionuclides from degrading spent fuel to environment	If the SFP is located inside the containment, the potential release paths to the environment are almost the same as for core melt accidents in the RPV. If the SFP is located outside the containment, the potential release paths to the environment depend very much on plant specific properties, e.g. ventilation systems, building doors, roof under thermal impact, size of rooms on the path etc. In any case the impact of very hot gas and of hydrogen has to be considered.
5.	Concurrent accident progression in spent fuel pool and reactor system	Fuel melt occurs only if the plant status is in severe disorder. It seems difficult to prove that not both the reactor and the SFP would be affected by such disorder. This is especially the case for external hazards. There are a large number of analyses for various containments to cope with the consequences of core melt accidents. However, additional loadings due to SFP steam generation and melting processes will add an additional challenge for containments which house the SPF. This could be considered as a cliff-edge effect for containment performance. It is conceivable that melt-through of the SFP bottom or wall could affect systems and components which are important for reactor safety, e.g. molten material from the SFP could enter the sump and damage ECCS components.
6.	Core concrete interactions for spent fuel pool accidents	The melt level in the SFP can become rather thick. Such a thick melt layer would probably develop convection patterns which predominantly transfer the heat to the upper edge of the melt. In addition, a metal layer could float on top of the melt and also create local high lateral heat fluxes. On the other hand, vigorous bubbling due to fuel-concrete interaction would tend to equalize heat fluxes. In summary, it has to be taken into account that local peak heat fluxes at the upper edge of the melt pool in the SFP can exist.
7.	Criticality	A qualitative analysis can be performed to demonstrate that SFP criticality is not likely in case of PWR spent fuel pool as it has sufficient fixed neutron absorber plates to mitigate any reactivity increase.
8.	Safety assessment of spent fuel pool during decommissioning	All phenomena of SFP accidents which are relevant in operating reactors are relevant for the decommissioning phase as well. An interesting additional issue still to be solved is whether after a certain extended time the decay heat is so low that even without water no significant fuel damage and radioactive release would occur.

5.3 COMPLEMENT OF EXISITING GUIDANCE BASED ON RECENT R&D ON CORE MELT ISSUES IN GENERAL

5.3.1 RECENT R&D ON ACCIDENTS IN REACTOR SHUTDOWN STATES

The accident analyses available so far, in principle, did not reveal unexpected phenomena or evolutions. Of course the timing of events is different from full power accidents, and specific issues occur with open RPV. Therefore, it can be concluded that existing guidance to perform L2 PSA for full power mode can be applied, in principle, for shutdown sequences in the RPV as well.

In assessment of SA scenarios progression and consequences the analysts shall take into account the results of SA computational analysis are characterized by significant uncertainties which are associated with limited code validation basis, assumptions/simplifications applied during input model (deck) development, as well as with initial and boundary conditions selected for particular analysis. As an example, recent benchmark studies for the spent fuel pool loss of cooling and loss of coolant SA scenarios performed under NUGENIA+ Air-SFP project demonstrate that significant differences in the results obtained with different and even with the same computer codes can be observed regardless of the fact that initial and boundary conditions are well-defined and fixed. These differences were related to different approaches applied for SFP modeling (e.g., advanced vs simple SFP models) and assumptions of boundary conditions for the calculations (position/orientation and hydraulic losses of the leakage flow path, conditions for oxidation start, etc.). To account these uncertainties in SA analysis it is recommended to perform case studies of key SA scenarios with different codes, model assumptions, and variation of initial and boundary conditions.

5.3.2 ANALYSIS OF THE COMPLEXITY OF SEVERE ACCIDENT PHENOMENOLOGY BY CODE SIMULATION (ASTEC AND MELCOR)

ASAMPSA_E document D40.7vol4 highlights the recent modelling improvements since ASAMPSA2's end (thus from 2014 to 2016) by using ASTEC and MELCOR code. Here is a summary of the most recent achievements:

The ASTEC integral severe accident code, jointly developed by IRSN and GRS since 1996, has multiple applications, including:

- evaluation of possible releases of radioactivity outside the containment;
- PSA2 studies, including determination of uncertainties;
- accident management studies, with emphasis on measures for prevention and mitigation of severe accident consequences;
- phenomenological analyses of scenarios to improve understanding of physical phenomena, as part of the support for experimental programs.

Consistently with severe accident R&D priorities, key model improvements have already been identified for the next code versions, in particular in-vessel and ex-vessel corium coolability. In accordance, the main ongoing modelling efforts are spent in priority on the reflooding of severely damaged cores, on pool-scrubbing phenomena in the containment, on MCCI (in particular on the coolability aspects) as well as on kinetics of iodine and ruthenium chemistry in the circuits, and in lower priority on DCH. In addition, though first calculations of the Fukushima-Daiichi accidents were successfully performed with the current V2.0 version, developments are underway to more properly account for the specifics of BWR cores.

In the framework of 2013-2015 MELCOR development [60] different tasks have been completed, i.e. mechanistic fan cooler model, new debris cooling models in the CAV package (water-ingression and melt eruption through crust). Other model developments are in progress, i.e. CONTAIN/LMR model for liquid metals reactors and multiple fuel rod types in a COR cell.

5.3.3 INVESTIGATION OF IN-VESSEL MELT RETENTION STRATEGY

In-vessel melt retention strategy through external cooling of the reactor vessel is a promising SAM measure. The aim is to terminate the progress of a core melt accident and to ensure the final coolability of the reactor pressure vessel. IVMR strategy is a potential solution to avoid or mitigate reactor vessel failure and further fission products release to the containment and to the environment outside.

The European H2020 project IVMR (In-Vessel Melt Retention Severe Accident Management Strategy for Existing and Future NPPs), leaded by IRSN, is aimed at analyzing the applicability and technical feasibility of the IVMR strategy to high power reactors, both for existing ones (e.g. VVER 1000 type 320 units) as well as for future reactors of different types (PWR or BWR). In this regards, the specific project objectives, are:

- Review, from an analytical point of view, the possibility to retain the corium inside the vessel by external cooling, for several kinds of reactors in Europe (existing or under project), following the standard methodology already applied to some existing VVER-440 (Loviisa and Paks) and to new concepts like AP-600, AP-1000 and APR-1400;
- Provide new experimental results to assess the models used in the methodology, in particular to cover all possible configurations of corium in the lower plenum and all geometries of lower head (VVER-1000 and BWR geometries were less studied in the past).
- Investigate several options to improve the IVMR methodology by reducing the degree of conservatism in order to derive more realistic safety margins, which is necessary when considering in-vessel melt retention in high power reactors.
- Elaborate an updated and harmonized methodology for the analysis of IVMR that will be used for various types of reactors and implemented in various codes used in Europe.

The main outcomes of the project will be relevant assumptions and scenarios to estimate the maximum heat load on the vessel wall, improved numerical tools for the analysis of IVMR issues and a harmonized methodology on the IVMR. To this end, in the frame of the project will be done:

- Making a comparative assessment of the existing results, assumptions and models that are applicable to evaluate the safety margins of various types of existing reactors, including high power reactors (1000 MWe or above) for which the safety demonstration is more difficult because the margins are low.
- Providing new experimental results that will allow to make less conservative assumptions in the models used to evaluate heat transfers from the molten corium to the vessel wall. Experiments with real materials will help to understand the transient evolutions of material layers in the molten pool and the effects of the presence of crusts. Experiments with simulant materials will help to understand the heat transfers associated to transient evolutions of material layers.
- Providing new experimental results for external cooling of the vessel, including innovative technologies such as porous coating, spray cooling or optimization of baffle shape for semi-elliptical vessels.
- Establishing a new methodology using new (less conservative) assumptions and new models based on the new data obtained. The methodology will consider several reactor designs (including Gen-II and Gen-III) and will consider complementary accident management options to optimize IVMR, such as the combined in-vessel reflooding. The methodology will also include uncertainty evaluation.

The INRNE has done some investigations on the applicability of the In-Vessel Melt Retention (IVMR) strategy with external vessel water cooling to the reactors of VVER-1000/v320 type. IVMR strategy is one of the feasible solutions to mitigate reactor vessel failure and further fission products release to the containment and to the environment outside. The reference power plant for this investigation is VVER-1000/v320 reactor sited at Unit 5 and 6 of Kozloduy NPP. The ASTECv2.0r3 severe accident computer code was used to simulate Large Break LOCA (2×850 mm) with full Station blackout (SBO) in VVER1000/v320 reactor model. In the calculation external water cooling of the vessel lower head was simulated and the model boundary conditions for the vessel/water heat exchange are applied. According to this study, external water cooling can be a successful strategy for severe accident management. In the frame of the European project IVMR (662157) for the future activities it is planned to assess the applicability IVMR SAM strategy for VVER 1000 reactor type based on results from experimental test facilities. This will consist in performing calculations with state of the art computer codes used for Severe Accident analyses like ASTEC computer code.

5.3.4 STATUS OF SOURCE TERM RESEARCH AND PERSPECTIVES FOR L2 PSA

Source Term (ST) research remains of high priority for evaluation and reduction of radioactive releases during accidents in NPP. Despite the recent achievement of major experimental programs, see for instance [61], and significant advances in understanding of ST issues, as reported in [62], additional research is still required as recently reviewed in an international workshop [63] for the consolidation of ST and radiological consequences analyses. A short synthesis of acquired knowledge and remaining gaps is provided in ASAMPSA_E document D40.7vol4 and summarized as follows:

- FISSION PRODUCTS RELEASE FROM FUEL UNDER ACCIDENTAL CONDITIONS

The existing large experimental database on Fission Product (FP) release from fuel [64], [65], [66] in accidental conditions highlights that volatile FPs (I, Cs, Te) are nearly completely released in core meltdown accidents involving significant fuel degradation, while release of semi-volatile FPs (Mo, Ba, Ru) is strongly dependent on fuel oxidation and oxygen potential in the coolant flow. Mo and Ru release tend to be large in oxidant conditions while Ba release tends to be larger in reducing conditions.

Semi-volatile FP-release understanding and modelling have still to be improved since presently FP release models do not capture well the effects of fuel oxidation and of oxygen potential in the coolant flow. With respect to ST assessment and radiological consequences analyses, semi-volatile FP contributions are and will be reassessed based on research results, more particularly that of ruthenium which, through the gaseous RuO_4 species, may contribute significantly to short and long term consequences in accidents involving oxidant conditions.

- FP TRANSPORT IN REACTOR COOLANT SYSTEM FOCUSING ON IODINE AND RUTHENIUM

Much progress was made on understanding and modelling of gas-phase iodine chemistry in the Reactor Coolant System (RCS) based on PHEBUS FP and CHIP tests results. The experimental and kinetic database is currently being extended at IRSN to treat Ag, In and Cd effect on iodine transport and chemistry (CHIP+ program). All these developments aim at providing better predictions of the gaseous iodine fractions at the RCS break during a severe accident.
Some progress in understanding Ru transport was obtained from experimentation [67]. However, experiments with more representative deposition surfaces are necessary to provide data for the development of applicable models.

A remaining important issue is the development of a pragmatic research approach to tackle complex heterogeneous processes (interactions of gas species with surfaces and aerosols in the RCS) and assess the effect of re-suspension/re-volatilization/decomposition of deposits resulting from mechanical, thermal and dose loadings. Performing reactor-relevant experiments and developing models still appear challenging due to the complexity of the involved processes and the importance of using representative surface states and deposits.

- FP BEHAVIOUR IN THE CONTAINMENT FOCUSING ON IODINE AND RUTHENIUM

The research was recently conducted in the International Source Term Program (ISTP) conducted by IRSN and CEA, the OECD/NEA BIP-1 and BIP-2 [70] conducted by CNL, the OECD/NEA THAI-1 and THAI-2 conducted by Becker Technologies and OECD/NEA STEM project conducted by IRSN. Gas-phase processes are reasonably well covered by past, on-going and planned research (within the OECD/NEA BIP-3, STEM-2, THAI-3 follow-up projects which are being launched in 2016) with a focus on inorganic gaseous iodine species, I_xO_y, Org-I and gaseous ruthenium tetroxide (RuO_4) behavior. Estimates of remaining uncertainties in ST evaluations were examined. Such studies were also helpful in identifying main sources of remaining uncertainties and these research programs are well targeted for their reduction.

Less work was recently performed on containment aqueous-phase chemistry in SA as the main source of volatile iodine was considered to be in the gas phase. The effects of evolving hydrodynamic and chemical conditions on FP pool scrubbing efficiency in suppression pools and FCVS during a severe accident were, are or will be partly investigated. There are presently only limited concerted research actions in the field, with the notable exception of the EU-PASSAM[1] project covering some aspects, and a larger collaboration is currently being built to progress on scrubbing modelling.

- FUTURE MILESTONES IN FP BEHAVIOUR

In terms of research in the ST area, the next identified major milestones will correspond to the achievement of main on-going research programs (STEM2, BIP3, THAI3, and PASSAM) and the implementation of their outcomes in SA codes; i.e. in 2020.

The final objectives of the ST research are to contribute to the consolidation of reference ST calculations used, notably, for design of population protection measures and of fast-running calculation tools used to support emergency response. Some on-going projects are dealing with this issue such as the on-going EU FASTNET project. One of the objectives of the project will be first to define main categories of accident scenarios in main types of operating reactors in Europe, including spent fuel pool accidents and to benchmark source term calculations for these “reference” categories of accidents.

- ACCIDENT PROGRESSION AND POSSIBLE OFF-SITE CONSEQUENCES

In the event of a nuclear power plant accident, protection of the public and environment from the potential release of radioactive materials would need efficient diagnosis and anticipated decision. Some organizations are developing more sophisticated approach using deterministic and probabilistic approaches. For example, Lloyd’s Register Consulting has developed the RASTEP (RApid Source TErm Prediction) tool alongside the Swedish Radiation Safety Authority (SSM). This product is a dynamic new type of software tool, supporting fast diagnosis and clear, informed decision-making.

RASTEP is based on Bayesian Belief Networks (BBNs). This is an established method of representing uncertain relations among random variables and capturing the probabilistic relationship between these variables (using Bayes’ theorem). The BBN approach is to take prior beliefs at the outset and, later on, when information on the progression of an event becomes available, modify and update those beliefs.

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[1] See public documents at https://gforge.irsn.fr/gf/project/passam/

Independently from Lloyd's Register Consulting and RASTEP, GRS in Germany has developed a tool which is based on the same principles and which has the same objectives. It is already implemented in most German NPPs in order to support the crisis teams for predicting source terms in case of an accident. The tool is being further developed, adding features for accidents in the spent fuel pool and in shutdown conditions.

It is interesting to note that two organizations independently of each other develop very similar solutions. This seems to be an indication that this approach is promising and that it may be recommended for general use.

5.3.5 RECENT R&D ON SPENT FUEL POOL ACCIDENTS

While ASAMPSA_E recognizes the importance of filling large knowledge gaps for accidents at SFPs, not much is done beyond the identification of areas to be further developed or investigated. Concerning thermal-hydraulics, phenomena related to de-watering, hydrogen generation, fuel degradation and possible release pathways, are only briefly mentioned.

- Concerning the treatment of criticality safety, fundamental aspects are dealt with very briefly for PWR types, and only marginally for BWRs that are characterized by a potentially higher risk level.
- While the SFP damage frequency may be lower than for reactor cores, the radiological consequences of severe accidents in SFPs may be higher.
- There is high uncertainty that remains about simulation codes. Particularly, when comparing codes in AIR-SFP, which is quite uncertain i.e. even little assumption by the code user lead to substantial differences in the results. Actually it is not a code issue, since the same code produces rather scattered results depending on the model built by the user.

As part of the CSNI activities motivated by the Fukushima Dai-ichi accident, WGAMA and WGFS have produced a "CSNI Status Report on Spent Fuel Pool under loss of cooling and loss of coolant accident conditions" [58]. The main objectives were:

(1) to produce a brief summary of the status of Spent Fuel Pool accident and mitigation strategies to better contribute to the post-Fukushima Daiichi NPP accident decision making process;

(2) to provide a brief assessment of current experimental and analytical knowledge about loss of cooling accidents in spent fuel pool and their associated mitigation strategies;

(3) to briefly describe the strengths and weaknesses of analytical methods used in codes to predict spent fuel pool accident evolution and assess the efficiency of different cooling mechanisms for mitigation of such accident; and

(4) to identify and list additional research activities required to address gaps in the understanding of relevant phenomenological processes, to identify where analytical tool deficiencies exist, and to reduce the uncertainties in this understanding.

Separate and integral effect tests have been conducted since the 1980s to better understand the fuel behaviour and degradation under severe accident conditions. The main objective of these tests was to provide data for model development and validation of computer codes used for reactor safety analysis. A number of the tests, while not originally developed for SFP accidents, provided valuable data and insights for application to SFP accident phenomenology. For example, the international PHEBUS Fission Product program, conducted in France, provided insights and data on the fission product release and late phase melt progression for LWRs. Most of the findings of these tests are directly applicable to accident progression in SFPs.

Another set of integral tests, suitable for model validation and application to SFP accident conditions, are QUENCH-10 and QUENCH-16, conducted in Germany. These tests not only provided an improved understanding of the oxidation phenomena, but also examined the phenomena associated with recovery and quenching of overheated fuel rods. Also the experiments and tests carried out to investigate the 2003 Paks cleaning tank incident have provided useful data.

The only integral tests specifically targeted for SFP loss of cooling accidents were conducted at Sandia National Laboratories, USA, partly within the OECD/NEA Sandia Fuel Project [68]. The main objective of the experimental work was to provide basic thermal-hydraulic data for completely uncovered and air cooled fuel assemblies for boiling and pressurized water reactors, and facilitate severe accident code validation and reduce modelling uncertainties. The accident conditions of interest for the SFP were simulated in a full-scale prototypic fashion.

5.3.6 CFD SIMULATION TOOLS

Simulation tools applied to SFP accidents include computer programs developed for analysis of thermal-hydraulics, nuclear criticality, fuel rod behavior and severe accidents. For the simulation of SFP thermal-hydraulics, CFD tools can be used in cases where 3D phenomena/regimes are important. They have the capacity to address problems at the local scale in 3D. However, SFP analyses are usually done at a larger scale, and the large simulation domain necessitates simplified modelling of the storage racks (porous medium approximation) and relatively coarse meshes in the CFD simulations. Thermal-hydraulics system codes are mostly applied for accident analysis at a large scale. System codes make use of 1D or 2D representations of the considered geometry, but they are being further developed into 3D tools.

Computational tools used for evaluation of the nuclear criticality safety of SFPs calculate the effective neutron multiplication factor of the SFP for any static configuration described in terms of geometry, material compositions, and extra information regarding cladding degradation, debris formation and physical state and level of the cooling water. These codes can in fact be used for both operational and accident conditions. Three types of calculation schemes are employed: a purely stochastic, a purely deterministic, and a hybrid scheme. A high level of accuracy in the results can typically be obtained by any of the schemes. The burnup dependent fuel composition can be provided by dedicated codes, which perform an in-core fuel depletion and fission products build-up analysis.

The fuel rod behavior during the early phase of a loss of cooling incident or accident, up to the loss of rod-like geometry, can be simulated with transient fuel behavior codes, which simulate the thermo-mechanical phenomena and the changes in fuel pellet and cladding in detail. However, they usually lack models for cladding high temperature oxidation in air-containing environments.

Severe accident codes have been developed for reactor applications by extending existing thermal-hydraulic codes with models for simulating phenomena in the reactor core during severe accidents. These codes are also used for analyses of SFP cooling accidents, because the major phenomena in severe reactor accidents are fundamentally the same as in severe SFP cooling accidents. However, the geometry and conditions expected in SFP accidents differ from those in reactor accidents, and the applicability of models in different severe accident codes is currently being verified for SFP conditions.

5.3.7 ABILITY OF REACTOR CORE SEVERE ACCIDENT CODES TO SIMULATE SFP SEVERE ACCIDENTS

The European Severe Accidents Research Network SARNET investigated the capabilities of severe accident codes to analyze SFP accidents [69]. This investigation comprised:

(1) the state of knowledge, especially with regard to phenomena related to oxidation in air of the fuel rod claddings,

(2) the state of code assessments on integral tests like QUENCH or PARAMETER; tests allowing to study accidental transients of oxidation in air of fuel rod claddings, ending by reflooding; and SFP tests allowing to study the behavior of one of several fuel assemblies for representative transients of loss of coolant SFP accident, inducing fuel claddings oxidation in air and burn propagation, and

(3) the assessment of different SFP accidents with different severe accident codes for different SFP geometries, different scenarios, and different levels and partition of the residual power on fuel assemblies.

The first two tasks clearly identified lacks in knowledge, and therefore on physical relevance of available models in severe accident codes; regarding the phenomena related to the oxidation in air or steam/air mixtures of the fuel claddings, especially the role of nitrogen in the acceleration mechanisms of cladding degradation and on the mechanical behavior of oxidized/nitrided claddings. Moreover, difficulties were revealed to model correctly the real 3D geometry and heterogeneity of fuel assemblies with the 2D cylindrical geometry usually applied by severe accident codes.

Concerning calculations of SFP transients, five different severe accident codes were used, namely: ASTEC, MELCOR, ATHLET-CD, ICARE/CATHARE, and RELAP/SCADPSIM. The calculations have shown the impact of modelling assumptions such as the number of nodes used to represent the fuel building, which can have strong impact on the gas flow between the different parts of the building. They also raise questions about the reliability of some results obtained with these severe accident codes, regarding in particular:

- the phenomena related to the cladding behavior in the presence of air or a steam / air mixture, such as oxidation, nitriding and embrittlement;
- the phenomena of natural convection and boiling in the fuel building. In fact, the conclusions on the coolability of fuel assemblies can be very different depending on the calculations; some studies show, for a loss of water transient (conducting to fast dewatering and air ingress in the fuel assemblies), that air flow is sufficient to remove the power, for other studies this conclusion depends on the air flow that could actually flow in the fuel assemblies;
- the conditions of air ingress in the assembly, according to the water depth, the assembly power, and the intensity of boiling; some studies show that for certain conditions, during the phase of fuel assembly dewatering, the air ingress flow through the top of the assembly (counter-current of steam flow) can cool down the upper part of the fuel assembly.
- the coolability of dewatered fuel assemblies with water injections.

5.3.8 ANALYSIS OF HEAVY LOAD DROPS IN THE SFP (UJV)

There are relatively large uncertainties connected to extent and type of fuel damage after the load drop. That is why it is necessary to perform a special deterministic analysis using an expert code dedicated for very fast non-stationary dynamic events like for example crash tests. Such analysis is necessary for definition of scope and type of damage for different loads.

The next step is analysis of selected scenarios (usually the most serious) using some of the integral codes for severe accident (MELCOR, MAAP, etc.). For Level 2 PSA, there are basically two most important factors:
1) degree of fuel damage and
2) location of the SFP (inside or outside containment).
Above described analyses were performed for a VVER-440 reactor at UJV Rez. The results proved that even for the worst case (fall of SFP cover with weight 6900 kg, speed 20 m/s) only a very limited number of fuel rods (26 = 14 in the central fuel assembly + 2 in each of the 6 neighboring assemblies) would be damaged directly. However, the fall of the SFP cover causes compression of the fuel rods together, so water cannot flow around the rods and the damaged rods heat up and melt. The melting process takes approximately 1 day and according to the analyses less than 1% of fuel is melted. The associated release of fission products into environment was assessed as late low release.

5.3.9 ONGOING R&D ACTIVITIES

- FRANCE

The DENOPI project, operated by IRSN and supported by the French government in the framework of post-Fukushima activities, is devoted to the experimental study of SFPs under loss of cooling and loss of coolant accident conditions. The project is divided into 3 parts:

- Two-phase convection phenomena in SFPs under loss of cooling conditions: The approach proposed in the DENOPI project is to conduct experiments on models of an SFP at reduced scale to contribute to the development and validation of two-phase flow convection models across the entire SFP.

- Physical phenomena at the scale of a fuel assembly under loss of coolant conditions: Experiments will be performed with partially uncovered fuel assemblies in order to study:

(1) the conditions for air penetration into the fuel assemblies;

(2) the void fraction in the fuel assemblies during boil-off, which is an important parameter in the evaluation of criticality issues; and

(3) the efficiency of a water spray to cool the fuel assemblies in case of a loss of coolant accident.

- Oxidation of zirconium by an air/steam mixture: Experiments on oxidation and nitriding of zirconium alloy fuel cladding will be performed in order to better estimate the margin to runaway of these exothermal reactions, leading to the destruction of the cladding.

- GERMANY

KIT is also planning to perform another semi-integral bundle test in the QUENCH facility, with special focus on SFP conditions, including steam-air mixtures. Such a test is expected to be conducted in the framework of the EC-sponsored Severe Accident Facilities for European Safety Targets (SAFEST) program.

The AIR_SFP project, launched recently in the framework of the European NUGENIA plate form, is dedicated to the application of accident codes to spent fuel pools, with three main objectives:

- improving severe accident code models to simulate air oxidation phenomena,
- defining recommendations to the use of severe accident codes for SFP accident applications,
- defining more precisely needs of R&D on different topics like large-scale flow convection, impact of partial dewatering or air flow on thermal runaway and fuel degradation.

GRS has been working on a research project financially supported by the German Federal Ministry of Economics and Technology (BMWi) regarding the extension of probabilistic analyses for spent fuel pools. Supporting deterministic analyses of the accident progression inside the SFP were a main part of the project. The accident progression has been analyzed for both PWR and BWR pools by using the integral code MELCOR 1.8.6.
From a R&D perspective, it is interesting to note that:

- MELCOR (and probably all other integral codes as well) cannot model melting in more than one “core”. This means that simultaneous melting in RPV and SFP cannot be calculated. Before melting begins, the water evaporation can be estimated by modelling the “first core” correctly, and assuming a certain heat load to the water in the “second core”.
- Heat transfer by radiation upwards from a melting SFP is not well represented by present integral simulation tools.

- JAPAN

NRA has been carrying out a spray test program for BWR spent fuel to obtain quantitative spray effects for accidental situations in SFP since 2014. The target scenarios are loss of coolant accidents (LOCAs) in SFP. Water spray is injected from a spray nozzle located above the fuel assemblies when spent fuel assemblies are uncovered fully or partially due to abnormal decrease in water level. In the tests, important knowledge of spray effects such as thermal hydraulic characteristics of liquid droplets atomization, counter-current flow and heat transfer between fuel rods and liquid droplets/liquid film will be obtained by measurements of fuel rod temperature, liquid velocity and void fraction inside/outside spent fuel assemblies. The tests start in 2016 after the test facility which consists of a storage tank, spent fuel assemblies (single bundle or multi bundles), storage racks and spray injection system is fabricated.

- OECD

In 2015 the Status Report on Spent Fuel Pools under Loss-of-Cooling and Loss-of-Coolant Accident Conditions (NEA/CSNI/R(2015)2 [58]) was issued. The report addresses number of topics including:

- phenomenology of SFP loss of cooling and loss of coolant accidents (criticality, thermal-hydraulic behavior) with an emphasis on severe accidents (fuel behavior, fuel assembly and rack degradation, fission product release and transport);
- integral tests and separate effect tests with relevance to SFP accidents;
- simulation tools used for analysis of SFP accidents.

In 2016 a Phenomena Identification and Ranking Table (PIRT) exercise on SFP under loss of cooling or coolant accidents conditions has been launched under the OECD/NEA/CSNI auspices. A particular emphasis will be placed on mitigation strategies.

- NUGENIA and Air-SFP Project

NEA/CSNI/R(2015)2 Status Report [58] indicated the necessity of benchmark activities to evaluate limitations associated with use of the codes originally developed for reactor applications in the SFP accident analyses. Recent activities in this area were performed under NUGENIA+ Air-SFP project and included evaluation of loss of cooling and loss of coolant SA scenarios for SFP geometry similar to Fukushima unit 4 spent fuel pool. Calculations were performed with 6 different computer codes, either developed for calculation of severe accidents in a reactor (ASTEC, ATHLET CD, MELCOR and SCDAPSIM) or for the calculation of thermal hydraulic problems (RELAP5). Evaluation of benchmark results identified that for the loss of cooling scenario the onset of fuel heat-up is rather well predicted. However, for the loss of coolant scenario the SFP draining velocities show a wide range of results which can be partly explained by differences in assumptions used for modelling of SFP leak flow path.

5.3.10 KNOWLEDGE GAPS AND FUTURE NEEDS

Even today's advanced L2 PSA and the related research encounter some important knowledge gaps. The following topics belong to this group where research is needed to improve the L2 PSA quality according to the opinion of the authors of D40.7vol4.

Releases into the waters and ground

The ASAMPSA_E May 2014 meeting participants noted that most of L2 PSA exclusively addresses releases into the atmosphere. Quantitative analyses of releases into water (river, lake, sea – see the Fukushima Dai-chi experience) were considered as missing. This is rooted in historic developments which concentrated on (immediate) health effects, and which seem to be less significant for water and ground releases. Nevertheless the consequences of such releases may be very significant.

Therefore, the related source term characteristics should be explored by L2 PSA. The WP40 partners note that relevant research and guidance in this field is missing.

Long-time effects inside plants

The ASAMPSA_E May 2014 meeting participants noted that long time effects – in particular related to the long term resilience of containments against fuel degradation accidents – should be addressed by L2 PSA. There may be some activities going on in this field, but the state of the art seems unfit for producing guidance.

Iodine and Ruthenium chemistry

In section 5.3.4 considerable R&D achievements and projects related to fission product chemistry are presented. What seems to be missing is a comprehensive and consistent application of such knowledge for severe accident analysis and for L2 PSA. While there are many references which address the chemical issues, there is hardly any L2 PSA which identifies the impact of chemical issues and the related uncertainties on L2 PSA results.

Combustible gas outside the containment

Hydrogen and carbon monoxide issues within the containment are routinely taken into account in PSA. However, related issues outside the containment seem to require more attention. The following topics belong to that field:

- distribution and transport of combustible gas in containment venting systems, in particular connected to steam condensation processes;
- leak of combustible gases out of the containment into adjacent rooms, and related distribution of these gases;
- distribution and transport in ventilation systems, taking into account the disturbed plant conditions after core melt;
- probabilities of ignition for potentially ignitable atmosphere in different parts of the disturbed plant.

Detailed CFD models or lumped-parameter containment models may in principle be available for precise evaluations, but given the multitude of potential accident sequences, their routine application in PSA is not practical. Additional guidance seems to be needed for adequately addressing these issues.

Treatment of uncertainties

Assessment of uncertainties should provide among other things a measure of the confidence that the results provided by PSA represent "real life" (what used to be called "robustness of results"). If the confidence is found to be low, the uncertainty analysis in L2 PSA shall provide information on the possible deviation in accident progression on the NPP and impact on the accident consequences.

Advances in this area have not been forthcoming since issuance of the ASAMPSA2 guidelines; hence ASAMPSA_E needs not address or repeat what has been already discussed at length in the past in these areas.

Since the Fukushima accidents sometimes doubts were raised whether PSA truly represent the accidental risk of NPPs. In this discussion it seems prudent to distinguish between L1 PSA issues and L2 PSA issues which are subject of the present document. With regard to L2 PSA the available experience in TMI, Chernobyl and Fukushima is not at all surprising. If L2 PSA had been performed based on the status of these NPPs at the onset of core damage, L2 PSA would have probably provided results not far from the actual experience.

5.3.11 SUMMARY CONSIDERING RECENT R&D FOR L2 PSA

The present section concentrates on fields of research which affect L2 PSA. In principle, the present section is updating the pertinent ASAMPSA2 documentation. Recent R&D and the ongoing research with relevance on extended L2 PSA are evaluated. The survey concentrated on ASAMPSA_E, SARNET (Severe Accident Research Network), SARNET-2, OECD and European projects (public results only), NUGENIA roadmap and ASAMPSA2.

The following Table 9 summarizes some R&D activities which are considered relevant for taking them into account in the discussion of further L2 PSA guidance. This list is preliminary, and since it is at least partly covering projects which are not yet finalized, no references are given. Where appropriate, the relevant leading organization(s) responsible for the topic is provided. Most of the issues mentioned below have links to one of the previous sections in the present document.

Table 9 Recent R&D related to L2 PSA and associated guidance suggestions for extended PSA

#	Recent R&D issues related to L2 PSA	Suggestion for improvement of guidance in ASAMPSA_E
1.	Analysis of fuel melt process in spent fuel pool with integral code (e.g. MELCOR)	There is no specific guidance for fuel melting in spent fuel pools; however existing guidance for core melting is largely applicable. Heat transfer from SFP to structures above and large scale MCCI need particular consideration, but no specific R&D or guidance is needed on this topic (see section 5.2.8).
2.	Deterministic and probabilistic analysis of accidents caused by external hazards in full power state	There is no specific guidance for accident analysis caused by external hazards; however existing guidance for internal initiators is generally applicable. According to D40.7voL2 [27] L2 PSA for external events needs no specific guidance.
3.	Conditional probability of containment loss of tightness after an earthquake	Seismic fragility of the Containment Vessel for overall structure failure and local failure can be evaluated based on the seismic design where response analyses for design basis earthquakes are conducted. In the context of an extended PSA also internal and external hazards could be taken into account which may affect the possibility to close the containment. Definition of mechanical criteria for loss of tightness is beyond the common practices of mechanical and seismic engineering practices. The definition of mechanical criteria for loss of containment tightness is not elaborated in this report. This issue shall be addressed in L1 PSA. Regarding L2 PSA, this issue is already discussed in D40.7vol3 [28].
4.	Source term assessment for L2	Source term research remains of high priority for evaluation and reduction of

#	Recent R&D issues related to L2 PSA	Suggestion for improvement of guidance in ASAMPSA_E
	PSA	radioactive releases during accidents in NPP. Despite the recent achievement of major experimental programs and significant advances in understanding of source term issues, additional research is still required. A short synthesis of acquired knowledge and remaining gaps, e.g. fission product release, its behaviour focusing on iodine and its chemistry is provided in this report. On-going research programs (STEM2, BIP3, THAI3, PASSAM) and the implementation of their outcomes in Severe Accident codes are expected by 2020.
5.	Analysis of the complexity of severe accident phenomenology by code simulation, ASTEC and MELCOR	Existing guidance for L2 PSA is generally applicable. In the framework of 2013-2015 MELCOR development different tasks, i.e. mechanistic fan cooler model, new debris cooling models in the CAV package (water-ingression and melt eruption through crust) is completed. Other model development is in progress, i.e. CONTAIN/LMR model for liquid metals reactors, multiple fuel rod types in a COR cell.
6.	Analysis of heavy load drops into the SFP	There is no specific guidance. Existing guidance for core melting is generally applicable.
7.	Investigation of the IVR by external cooling of reactor vessel for VVER-1000 type reactors	Guidance is under development The European project IVMR (662157) – H2020 is started with the main objective to review, from an analytical point of view, the possibility to retain the corium inside the vessel due to external cooling, for several kinds of reactors in Europe (existing or under project).
8.	Accident progression and possible off-site consequences	On-going research activities in FASTNET. Development of two tools is summarised in section 5.3.4. These tools (developed by LRC and GRS) facilitate diagnosis and decision-making by prognosis of source terms for nuclear emergency management.
9.	Spent Fuel Pool Rupture Characterization Based on Water Level Monitoring	There is no specific guidance on critically important instrumentation and measurements. In case of SFP LOCA the location and size of SFP rupture is critical. Therefore pertinent instrumentation availability deserves consideration in L2 PSA and related guidance.

6 LIST OF REFERENCES

[1] Synthesis of the initial survey related to PSAs End-Users needs. - Technical report ASAMPSA_E/WP10/D10.2/2014-05 IRSN PSN-RES/SAG/2014-00193

[2] Minutes and recommendations of the ASAMPSA_E Uppsala End-Users workshop (26-28/05/2014) - Reference ASAMPSA_E - Technical report ASAMPSA_E / WP10 / 2014-07 - Reference IRSN PSN-RES/SAG/2014-00335

[3] Minutes of the ASAMPSA_E WP10 WP 21 WP22 WP30 technical meetings 8th-12th September 2014. Hosted by Vienna University in Vienna, Austria WP5/2014-06 - Reference ASAMPSA_E: WP5/2014-06 - Reference IRSN: PSN/RES/SAG/2014-00318

[4] ASAMPSA2, Best-Practices Guidelines for L2 PSA Development and Applications, Volume 1 - General, Reference ASAMPSA2, Technical report ASAMPSA2/ WP2-3-4/D3.3/2013-35, IRSN-PSN/RES/SAG 2013-0177, dated 2013-04-30.

[5] ASAMPSA2, Best-Practices Guidelines for L2 PSA Development and Applications, Volume 2 - Best practices for the Gen II PWR, Gen II BWR L2 PSAs, Extension to Gen III reactors, Reference ASAMPSA2, Technical report ASAMPSA2/ WP2-3-4/D3.3/2013-35, IRSN-PSN/RES/SAG 2013-0177, dated 2013-04-30.

[6] ASAMPSA2, Best Practices guidelines for L2 PSA development and applications, Volume 3- Extension to Gen IV reactors, Technical report ASAMPSA2/ WP2-3-4/D3.3/2013-35, IRSN-PSN/RES/SAG 2013-0177, dated 2013-04-30.

[7] "Probabilistische Sicherheitsanalyse (PSA): Qualität und Umfang, Richtlinie für die schweizerischen Kernanlagen," ENSI A-05, Ausgabe Januar 2009.

[8] "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," Specific Safety Guide No. SSG-4, IAEA, May 2010.

[9] Safety Report Series No.52, Best estimate Safety Analysis for Nuclear Power Plants: Uncertainty evaluation; Vienna 2008, http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1306_web.pdf

[10] “Summary report of already published guidance on L2 PSA for external hazards, shutdown states, spent fuel storage, Technical report ASAMPSA_E / WP40 / D40.2 / 2014-08, 31/12/2014

[11] IAEA: INES: The International Nuclear and Radiological Event Scale User's Manual, 2008 Edition, Non-Serial publications, , IAEA-INES-2009, 206 pp.,, Vienna 2009

[12] J. Vitazkova: Methodology of common risk target assessment and quantification for severe accident of nuclear power plants based on INES scale, Slovak University of Technology, Bratislava, June 2014

[13] G. Cojazzi et al., Benchmark Exercise on Expert Judgment Techniques in PSA Level 2, Proceedings of FISA 99, EU Research in Reactor Safety, 29 November - 1 December 1999, Luxembourg

[14] Basic safety principles for nuclear power plants 75-INSAG-3 Rev. 1, INSAG-12, A report by the International Nuclear Safety Advisory Group, 1999

[15] IAEA Safety Standards for protecting people and the environment, Fundamental Safety Principles, Safety Fundamentals No. SF-1, IAEA, 2006, ISBN 92-0-110706-4, ISSN 1020-525X, Vienna

[16] U.S. Nuclear Regulatory Commission, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-informed Decision making”, draft report for comment, NUREG-1855, Rev. 1, March 2013

[17] https://forms.nrc.gov/reading-rm/doc-collections/commission/slides/2011/20110728/fleming-integrated-risk-paper.pdf

[18] V. Hassija et al, “Probabilistic safety assessment of multi-unit nuclear power plant sites – An integrated approach”, Journal of Loss Prevention in the Process Industry, Volume 32, November 2014, Pages 52–62)

[19] US Nuclear Regulatory Commission (USNRC), “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants”, NUREG-1150 (1990).

[20] Ajit Kumar Verma, Srividija Ajit, Durga Rao Karanki: Reliability and Safety Engineering Springer Series in Reliability Engineering, ISSN 1614-7839, ISBN 978-1-84996-231-5, DOI 10.1007/978-1-84996-231-2, Springer Verlag London Limited 2010.

[21] Report number: 2010:36 ISSN:2000-0456, by Jan-Erik Holmberg and Michael Knochenhauer VTT, P.O.Box 1000, FI-02044 VTT, Finland, and Scandpower AB, SE-172 25 Sundbyberg, Sweden, May 2011.

[22] IAEA TECDOC 1511, Determining the Quality of PSA results for applications in Nuclear Power Plants , IAEA, July 2006.

[23] http://www-pub.iaea.org/MTCD/publications/PDF/Pub1013e_web.pdf, INSAG10

[24] US NRC : http://www.nrc.gov/reading-rm/doc-collections/acrs/letters/1999/4621827.html, The role of defense in depth in a risk-informed regulatory system,

[25] J.Vitazkova, E.Cazzoli: The principle of Defence-in-Depth in the perspective of Probabilistic Safety Analyses in wake of Fukushima, Risk Analysis IX, Book series: WIT Press, 9th International Conference on Risk Analysis and Hazard Mitigation, June 2014.

[26] ASAMPSA_E, Lessons of the Fukushima Dai-ichi accident for PSA - Technical report ASAMPSA_E/ WP30 / D30.2 / 2017-32 - Reference IRSN PSN/RES/SAG/15-11

[27] ASAMPSA_E guidance for level 2 PSA. Volume 2. Implementing external Events modelling in Level 2 PSA- Technical report ASAMPSA_E WP40/D40.7/2017-39 vol 2. Reference IRSN PSN-RES/SAG- 2017-00021

[28] ASAMPSA_E guidance for level 2 PSA. Volume 3. Guidance on the verification and improvement of SAM strategies with L2 PSA- Technical report ASAMPSA_E WP40/D40.7/2017-39 vol 3. Reference IRSN PSN-RES/SAG-2017-00001, IRSN PSN-RES/SAG-2017-00026

[29] ASAMPSA_E guidance for level 2 PSA. Volume 4. Complement of existing ASAMPSA2 guidance for shutdown states of reactors, Spent Fuel Pool and recent R&D results, ASAMPSA_E/WP40/D40.7/2017-39 vol 4,

[30] ASAMPSA_E, “Risk Metrics and Measures for an Extended PSA”, Technical report ASAMPSA_E/WP30/D30.7/2017-31 vol 3 - Reference IRSN PSN/RES/SAG/2017-00018

[31] INSAG, “Basic Safety Principles for Nuclear Power Plants”; 75-INSAG-3 Rev. 1 - INSAG-12; 1999

[32] ASAMPSA_E, "The Link between the Defence-in-Depth Concept and Extended PSA”, Technical report ASAMPSA_E/WP30/D30.7/2017-31 vol5-Reference IRSN PSN/RES/SAG/2017-000019

[33] ASAMPSA_E, Guidance for Decision Making based on Extended PSA, ASAMPSA_E D30.7vol1, Technical report ASAMPSA_E/WP30/D30.7/2017-31vol1, IRSN PSN-RES/SAG/2017-00016

[34] ASAMPSA_E, “Synthesis report of the End-Users survey and review of ASAMPSA_E guidance, and final workshop conclusions. Identification of follow-up useful activities after ASAMPSA_E”, Technical report ASAMPSA_E/WP10/D10.5/2017-40, IRSN PSN-RES/SAG-2017-00026

[35] Vitazkova, J., Cazzoli, E. Common Risk Target for severe accidents of nuclear power plants based on IAEA INES scale. Nuclear Engineering and Design, Vol.262, Pages 106-125; ISSN 0029-5493, September 2013

[36] IAEA: INES: The International Nuclear and Radilogical Event Scale User's Manual, 2008 Edition, Non-Serial publications, IAEA-INES-2009, 206 pp., Vienna 2009

[37] ASAMPSA_E, Decker, K., H. Brinkman, “List of External Hazards to be Considered in Extended PSA”, ASAMPSA_E/WP21/D21.2/2017-41, IRSN PSN-RES/SAG-2017-00011

[38] Vitazkova J., Cazzoli E.: Probabilistic Safety Assessment KKM Shutdown Internal Floods, September 2010

[39] Cazzoli E., Vitazkova J.: Assessment of Impact of DIWANAS backfits on LERF, KKM LEVEL2 update, September 2013

[40] Cazzoli E., Vitazkova J.: Probability Safety Assessment, KKM LEVEL2 update, August 2013

[41] G. Georgescu and R. Georghe (nuclearsafety.gc.ca), “Summary of Session 3, Experience wim Multi-Unit PSA, Part 3”, from Proceedings of International Workshop on Multi-Units Probabilistic Safety Assessment (PSA), Ottawa, Canada, November 17-20, 2014

[42] K. N. Fleming, “A Technical Approach to Meeting Challenges in Multi-Unit PSA”, from Proceedings of International Workshop on Multi-Units Probabilistic Safety Assessment (PSA), Ottawa, Canada, November 17-20, 2014

[43] « SLUTRAPPORT för FRIPP för Forsmark ½ », Vattenfall – report PK-41/91 - 1991-07-17

[44] Pat Hiland, Station Blackout and Advanced Accident Mitigation (B.5.b) Overview, April 28, 2011, http://www.nrc.gov/reading-rm/doc-collections/commission/slides/2011/20110428/staff-slides-20110428.pdf.

[45] NEI, B.5.b Phase 2 & 3 Submittal Guideline, Report no. NEI 06-12, Rev. 3, September 2009.

[46] NEI, Diverse And Flexible Coping Strategies (FLEX) Implementation Guide, NEI 12-06, Rev. 0, August 2012.

[47] Jeffrey A. Julius, Jan Grobbelaar, & Kaydee Kohlhepp, Advancing Human Reliability Analysis Methods for External Events with a Focus on Seismic, Probabilistic Safety Assessment and Management PSAM 12, June 2014, Honolulu, Hawaii.

[48] EPRI, A Preliminary Approach to Human Reliability Analysis for External Events with a Focus on Seismic. EPRI, Palo Alto, CA: 2012. EPRI 1025294.

[49] D. Jacquemain et.al, Nuclear Power Reactor Core Melt Accidents, IRSN 2015.

[50] Vasilev, Alexander, Role of Zirconium Nitride formation under Zirconium-based claddings oxidation in air during NPP beyond-design-basis accidents, 22nd Int. conference on Nuclear Engineering, 2014, ISBN: 978-0-7918-4596-7.

[51] F. C. Iglesias, C. E. L. Hunt, F. Garisto and D. S. Cox, Measured Release Kinetics of Ruthenium from Uranium Oxides in Air, Proc. Int’l. Seminar on Fission Product Transport Processes During Reactor Accidents, J. T. Rogers, editor, Hemisphere Publishing Corp., Washington, D. C., 1990.

[52] Z. Hózer, L. Matus, P. Windberg, CODEX and RUSET air oxidation experiments, Proc. 5th Technical seminar on the Phebus FP programme, Aix-en-Provence, France, June 24-26, 2003.

[53] EPRI-3002000498 Spent Fuel Pool Risk Assessment Integration Framework (Mark-I and II BWR) and Pilot Plant Application, Palo Alto, CA 2013.

[54] EPRI-3002002691, PWR Spent Fuel Pool Risk Assessment Integration Framework and Pilot Plant Application, Palo Alto, CA June 2014.

[55] European Utility Requirements for LWR Nuclear Power Plants, Volume 2, Generic Nuclear Island Requirements, Chapter 17, PSA Methodology, Revision D, October 2012.

[56] Alvarez, R. et al, Reducing the hazards from stored spent power-reactor fuel in the United Stateş, Science and Global Security, Vol.11, pp.1-51, 2003.

[57] U.S. Nuclear Regulatory Commission, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," NUREG-1738, February 2001.

[58] OECD, Nuclear Safety NEA/CSNI/R(2015)2, Status report on Spent fuel pool under loss of cooling and loss of coolant accident conditions, Final report, May 2015.

[59] Grimm, P., et al. Parametric studies on the reactivity of spent fuel storage pools, 1984. In: Fifth International Meeting on Thermal Nuclear Reactor Safety, September 9-13, 1984, Karlsruhe, Germany: Nuclear Research Center Karlsruhe, KfK 3880/1, Vol. 1, pp. 626-633.

[60] Larry Humphries, MELCOR Code Development Status, 7th meeting of the "European MELCOR User Group", BEL V, Tractebel Engineering, Brussels, Belgium, march 17-18, 2015.

[61] B. Clément, R. Zeyen, The Objectives of the PHEBUS FP Experimental Program and Main Findings, Special Issue of Annals of Nuclear Energy, Vol. 61, pp. 4-10, 2013

[62] Herranz L.E., Haste T. & Karkela T., Major Advances in the Source Term Area within the European Severe Accident Research Network (SARNET2), Nuclear Engineering and Design, Vol. 288, pp. 56-74, 2015

[63] International Iodine Workshop, Marseille, March 2015, Full Proceedings, OECD/NEA/CSNI report, NEA/CSNI/R(2016)5, 2016

[64] Lewis B.J., Dickson R., Iglesias F.C., Ducros G. & Kudo T., Overview of experimental programs on core melt progression and fission product release behaviour, Journal of Nuclear Materials, Vol. 380, pp. 126–143, 2008

[65] Pontillon, Y., Ducros, G., Behaviour of fission products under severe PWR accident conditions: The VERCORS experimental programme - Part 2: Release and transport of fission gases and volatile fission products, Nuclear Engineering and Design, Vol. 240 (7), pp. 1853-1866, 2010

[66] Grégoire A.C. & Haste T., Material Release from the Bundle in Phebus FP, Special Issue of Annals of Nuclear Energy, Vol. 61, pp. 63-74, 2013

[67] Karkela T., Ver N., Haste T., Davidovich N., Pyykonen J. & Cantrel L., Transport of Ruthenium in Primary Circuit Conditions during a Severe NPP Accident, Annals of Nuclear Energy, Vol. 74, pp. 173-183, 2014

[68] Characterization of thermal-hydraulic and ignition phenomena in prototypic, full-length boiling water reactor spent fuel pool assemblies after a postulated complete loss-of-coolant accident Report NUREG/CR-7143, USNRC

[69] J. Fleurot et al., Synthesis of Spent Fuel Pool Accident Assessments using Severe Accident Codes. Annals of Nuclear Energy, Vol.74, pp. 58-71, 2014

[70] Glowa, G.A., Moore, C.J. & Ball, J.M., The main outcomes of the OECD Behaviour of Iodine (BIP) Project, Annals of Nuclear Energy, Vol. 61, pp. 179-189, 2013

[71] J. von Linden et al., Assessment of the Accidental Risk of Advanced Pressurized Water Reactors in Germany, GRS-184, April 2012

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