**"NUCLEAR FISSION"**

**Safety of Existing Nuclear Installations**

**Contract 605001**

# Guidance on the verification and improvement of SAM strategies with L2 PSA

- *This version of the report will be submitted to a peer rewiew*
- *The conclusions of the review will be discussed during the ASAMPSA_E workshop with PSA End-Users (12-14th Sept. 2016)*
- *The report will then be improved before the end of the project (31-12-2016)*

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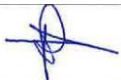
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Summary :

The report presents examples and comparison of severe accident management objectives and strategies from the experience of the project partners. It proposes an analysis of the use of L2 PSA for the verification and improvement of these strategies.

| Visa grid | | | |
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# SUMMARY

For each NPP, severe accident management (SAM) strategies shall make use of components or systems and human resources to limit as far as possible the consequences of any severe accident on-site and off-site. L2 PSA is one of the tools that can be used to verify and improve these strategies.

The present report (deliverable D40.5 of the project ASAMPSA_E) proposes a comparison of objectives in the different countries in terms of SAM strategies verification and improvement. The deliverable summarizes also experience of each partner (including potential deficiencies) involved in this activity, in order to derive some good practices and required progress, addressing:

- o SAM modeling in L2 PSA,
- o Positive and negative aspects in present SAM practice,
- o Discussion on possible criteria related to L2 PSA for verification and improvement: risk reduction (in relation with WP30 activities on risk metrics), reduction of uncertainties on the severe accident progression paths until NPP stabilization, reduction of human failure conditional probabilities (depending on the SAM strategy, the environmental conditions …),
- o Review with a perspective of verification and improvement on the main SAM strategies (corium cooling, RCS depressurization, control of flammable gases, reactivity control, containment function, containment pressure control, limitation of radioactive releases, …),
- o SAM strategies to be considered in context of an extended L2 PSA (as far possible, depending on existing experience), taking into account all operating modes, accidents also occurring in the SFPs and long term and multi-unit accident.

The deliverable D40.5 is developed from the partners' experience. Many of the topics described here are beyond the common practices of L2 PSA applications: in many countries, L2 PSA application can be limited to the calculations of frequencies of release categories with no formal requirement for SAM verification and improvement.

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*The following table provides the list of the ASAMPSA2 partners involved in the development of this document.*

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| 2 | Gesellschaft für Anlagen- und Reaktorsicherheit mbH | GRS | Germany |
| 4 | Ricerca sul Sistema Energetico | RSE S.p.A. | Italy |
| 5 | Lloyd's Register Consulting | LRC | Sweden |
| 6 | Nuclear Research Institute Rez pl | UJV | Czech |
| 8 | Cazzoli Consulting | CCA | Switzerland |
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| 12 | Electricité de France | EDF | France |
| 13 | Lietuvos energetikos institutas | LEI | Lithuania |
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# CONTENT

# GLOSSARY

| ALARP | As Low As Reasonably Practicable |
|-------|----------------------------------|
| APET | Accident progression event tree |
| CRT | Common risk target |
| CSIP | Complex (Integrated) Program of Ukrainian NPPs Safety Improvement |
| DCH | Direct containment heating |
| DiD | Defence in depth |
| EDMG | Extensive damage mitigating guidelines |
| EOP | Emergency operating procedure |
| FCVS | Filtered containment vent system |
| HRA | Human reliability assessment |
| IE | Initiating event |
| IVR | In-vessel retention |
| LERF | Large early release frequency |
| LOCA | Loss-of-coolant accident |
| L2 PSA | Level 2 probabilistic safety analysis / assessment |
| NPP | Nuclear power plant |
| PARs | Passive autocatalytic recombiners |
| PDS | Plant damage state |
| PFS | Performance Shaping Factor |
| PRISE | Primary to secondary leak |
| RCS | Reactor coolant system |
| RPV | reactor pressure vessel |
| SAM | Severe accident management |
| SAMG | Severe accident management guideline |
| SBO | Station Black Out |
| SFD | Spent fuel damage |
| SFP | Spent fuel pool |
| SG | Steam generator |
| SRV | Safety and Relief Valves |

# 1 INTRODUCTION

The severe accident management guidelines (SAMG) 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 role of the local and national emergency teams,
- the impact of severe accident strategies on physical phenomena, containment failure modes and radioactive release.

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…).

L2 PSAs help ranking NPPs weaknesses, by means of a suitable metric, in order to target priority efforts (modifications of design, optimization of procedures and guidelines …). They also help identifying some risks that can be reduced by mitigation measures or can be better understood through R&D efforts. An appropriate level of confidence in this ranking process through L2 PSA is definitely needed to be sure that efforts are directed on the relevant issues.

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.

For example, the question if the RPV has already failed or not may not be answerable based on the existing measurements. In addition, even if relevant measurements are available, cables in the containment or nearby buildings, which will experience a beyond design temperature and radiation during a severe accident, may not be available, so that such measurements cannot be used. In addition, many systems and signals have already failed; otherwise there would not be a severe accident. 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.

Generation 3 plants however, have features which allow the identification of plant state during a severe accident using dedicated equipment which is designed for the loads that occur during a severe accident.

It also needs to be mentioned that, whatever the L2 PSA method and objective, the results can be uncertain to a large extent. For example, can L2 PSA be used to identify which part of accident progression is very uncertain and whether some SAM strategies can reduce the uncertainties: the common sense for SAMG is to provide guidance on determining the plant state during a severe accident and to indicate measures to either delay or reduce the release of fission products into the environment.

Also, as mitigative measures may have both positive and negative effects, the goal of SAMG is to provide guidance to the plant staff comparing such risks.

The ASAMPSA2 guidelines [1], [2] discuss in detail how to introduce these issues in a L2 PSA or how to present the results of L2 PSA, but some practical views on SAM strategies verification and improvement based on L2 PSA conclusions are not provided. 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 D30.5 of the ASAMPSA_E project addresses several potential risk measures which might be suitable for this purpose.

The development of the D40.5 deliverable within ASAMPSA_E provides an opportunity to compare experience in the different countries and also objectives that can be associated to this activity.

A content of the D40.5 is given hereafter with the idea to collect the experience of each partner involved in this activity, including potential deficiencies, and then derive some good practices and required progress, addressing:

- o SAM modeling in L2 PSA,
- o Positive and negative aspects in present SAM practice,
- o Improvement of SAM strategies from L2 PSA experience,
- o Verifying the scope of SAMG (all types of accident shall be covered, whatever the initial plant status).

It has been decided during the 2015-05-26&27 WP40 technical meeting to not address the SAM verification and improvement for multi-units accident because this issue is considered not yet mature for L2 PSA.

# 2 HIGH LEVEL OBJECTIVES OF SAM STRATEGIES VERIFICATION AND IMPROVEMENT

This chapter compares the objectives formulated in each country in terms of SAM strategies verification and improvement. The objectives can be qualitative (for example, limited off-site countermeasures induced by a severe accident) or quantitative (e.g. frequency of releases exceeding $2x10^{15}$ Bq of $I_{131}$ below $10^{-6}$ /yr as used in Switzerland [4], [16]). They can also be associated to plant safety continuous improvement (e.g. L2 PSA use during periodic safety review) or legal requirement (e.g. demonstrate that strategies are as efficient as reasonable, ALARP concept …).

The objectives may be different for existing or future reactors.

The chapter includes a review of the situation for different NPPs / countries on the following issues:

- SAM verification and improvement concept (or equivalent);
- objectives to be reached (quantitative or qualitative);
- role of L2 PSA;
- link with PSRs and/or living PSA;
- link with legal requirement.

## 2.1 FRANCE

France has adopted since several years a continuous safety improvement approach for the PWRs in operation, in particular at the occasion of the periodic safety reviews of the nuclear installations, based on the analysis of past events and accidents, on operating experience or state of art progress. This includes SAM strategies verification and improvement. For EPR, an important objective was to achieve a significant reduction of potential radioactive releases due to all conceivable accidents, including core melt accidents.

### 2.1.1 L2 PSA REGULATORY FRAMEWORK

In France, the Nuclear Safety Authority (ASN) requests the development of a L2 PSA by the utility (EDF) to support the decennial periodic safety review of nuclear power plants series. In this context, IRSN, acting as TSO of the ASN, reviews the L2 PSAs developed by EDF. The objective of this review is to ensure that the study developed by the utility reflects as accurately as possible the risk associated with the NPPs. For some specific issues, this process can lead to additional actions (L2 PSAs or SAM strategies upgrade) by EDF or sometimes to additional requests by the safety authority.

By this way, L2 PSAs are associated to a progressive improvement of SAM strategies.

## 2.1.2 LINK WITH LEGAL REQUIREMENTS

Order of 7 February 2012 setting the general rules relative to basic nuclear installations:

- **Article 1.2** : […] *"in view of the state of knowledge, practices and the vulnerability of the environment, <u>enable the risks and drawbacks</u> mentioned in article L. 593-1 of the environment code t<u>o be brought to as low a level as possible under economically acceptable conditions</u>"* ;

- **Article 3.3**: *"The nuclear safety demonstration shall also include <u>probabilistic analyses of accidents and their consequences</u>, unless the licensee demonstrates that this is irrelevant. Unless otherwise specified by ASN, these analyses can be carried out in accordance with methods applied to the installations mentioned in article L. 512-1 of the environment code. They integrate the technical, organisational and human dimensions"*.

- **Article 8.1.2**: *"For any basic nuclear installation comprising one or more nuclear reactors, the probabilistic analyses mentioned in article 3.3 include <u>probabilistic safety studies associated with the risk of damaging the nuclear fuel and the risk of abnormal releases of radioactive substances</u>"*.

## 2.1.3 ROLE OF L2 PSA

For reactors in operation, the actual safety level and the impact of any improvement can be measured by accident frequencies and radiological consequences.

To achieve this, a L2 PSA is used to:

- provide a realistic determination of accident frequencies and consequences;
- take into account severe accident phenomenology depending on each scenario;
- estimate the benefits of accident management procedures for these scenarios;
- check if improvement of accident management procedures can be proposed for these scenarios;
- check if reactor design or operation modifications can be proposed for these scenarios;
- provide quantitative elements about advantages of any reactor design or operation modifications.

Uncertainties/sensitivity analysis is used for some issues.

For a new reactor (EPR FA3), L2 PSA is used in complement to the deterministic approach to demonstrate that the reactor safety objectives are achieved.

## 2.1.4 SAM OBJECTIVES TO BE REACHED

High level objective of SAM strategies is to preserve confinement of radionuclides assuming that preventing core melt was unsuccessful as it was targeted in accidental procedures before core melt. The aim is to:

- avoid or limit large radioactive release (release for which off-site protective measures are insufficient to protect people and the environment),
- avoid or limit early radioactive release in order to have off-site protective measures fully effective in due time.

Taking into account the highly degraded context in case of severe accident, SAM strategies should be pragmatic and robust, avoiding experts debate between emergency teams during the management of the accident.

Additionally some safety objectives are associated to a Long Term Operation (LTO) program (existing PWRs may be operated up to 60 years). These are:

- a significant reinforcement of water and power supply means (high impact expected on prevention of core melt accident due to hazards),
- for severe accident:
  - a reduction of radiological consequences in case of filtered containment venting,
  - solutions to extract power from the containment without containment venting,
  - solutions to prevent basemat failure in case of vessel rupture.

For EPR, the objectives of SAM strategies were defined in 2000 as follows:

- Accident situations with core melt which would lead to large early releases have to be "practically eliminated": if they cannot be considered as physically impossible, design provisions have to be taken to design them out. This objective applies notably to high pressure core melt sequences.
- Low pressure core melt sequences have to be dealt with so that the associated maximum conceivable releases would necessitate only very limited protective measures in area and in time for the public. This would be expressed by no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in consumption of food.

## 2.2 GERMANY

### 2.2.1 L2 PSA REGULATORY FRAMEWORK

Every ten years, a periodic safety review has to be performed by the licensees of NPPs in Germany. L1 PSA has been part of the periodic safety review for many years. A few L2 PSAs were performed prior to 2005, exploring L2 PSA methodology within R&D projects, but outside of the periodic safety review. In 2005 L2 PSA became part of the periodic safety review, and the licensees now have to submit a PSA (including Level 1 and Level 2) to the licensing authority. The scope of L1 PSA is normal operation and shutdown states, while L2 PSA has to be performed for normal operation only. A guideline (including L1 and L2 PSA) has been published by the Bundesamt für Strahlenschutz (BfS) on behalf of the federal ministry for environment, nature conservation and reactor safety (BMU). This guideline comprises a general introduction [5], a volume on methods [6] and a volume on data [7]. A working group has been installed which continuously monitors evolutions in PSA and proposes updating of the guidelines if needed. However, since shutting down of all German NPPs will occur until 2022, there is no strong incentive to further update the regulations.

### 2.2.2 ROLE OF L2 PSA

Performing and reviewing L2 PSA has become a routine task for German NPPs, but knowledge on production and review is not widespread. The production of L2 PSA is done by experienced companies on behalf of the utilities. The review is done by German TSOs (sometimes including additional experts) on behalf of the responsible licensing authority of the state where the plant is located.

Since no quantitative probabilistic safety criterion exists in Germany, and since frequencies of severe accidents are very low, and since L2 PSA issues are considered beyond design, the L2 PSA results mostly have no direct impact on plant improvements or decisions about plant safety.

However, results of severe accident analysis including L2 PSA are increasingly used for the assessment of existing and new SAM, and for the planning of plant-external emergency measures. Examples for such applications are the evaluation of the efficiency of passive autocatalytic recombiners, or the reliability of the containment venting system. A particular application is the implementation of L2 PSA data into a fast running code for the prediction of source terms during an accident.

## 2.2.3 SAM IN L2 PSA

SAM can be distinguished into preventive SAM (in order to prevent core melt – this is a L1 PSA issue) and mitigative SAM (in order to mitigate core melt consequences – this is a L2 PSA issue).

The modelling of SAM in present German L2 PSA is, in almost all cases, restricted to:

- Passive autocatalytic recombiners (PARs): In deterministic accident analysis (mostly done with MELCOR code), PARs and their action are modelled as realistic as possible. In probabilistic analysis, PARs are purely passive systems, and their reliability is not affected by accident-related phenomena, with the insignificant exemption of some mechanical damage due to mechanical impact of a large LOCA. As a consequence of the reliable PAR operation, hydrogen combustion contributes only vanishingly little to the containment threat.

- Filtered containment venting: The reliability of this system is assessed, taking into account system failures (e.g. blocked valves), and human error during operation. As a consequence, there is a probability of a few percent that venting is not activated when required. Furthermore, various deterministic analyses are performed in order to check that the venting system has sufficient capacity to depressurize the containment in different accident conditions. The capacity of the filter, in particular in presence of gaseous Iodine, is another topic which requires further assessment.

- Manual depressurization of the primary system: If a high pressure core melt scenario exists, there may be a delated activation of the depressurization system. In L2 PSA this is modelled in a rough manner, taking into account unsuccessful previous attempts before core melt (in L1 PSA) for depressurization.

- Retention of a molten core inside the RPV: If flooding of a molten core occurs (e.g. after depressurization of the RPV), L2 PSA estimates whether the core can be kept inside the RPV (TMI scenario) or whether the RPV will fail. The retention success depends on the point in time (to be assessed in PSA) when flooding begins and the flooding rate.

There may be much more mitigative SAM in a real accident, in particular when taking into account SAM upgrades and additional hardware after the Fukushima accident. However, these SAM additions have been decided and implemented without related PSA assessment.

## 2.3 SPAIN

### *2.3.1 L2 PSA REGULATORY FRAMEWORK*

In Spain, the Nuclear Safety Regulator (CSN) requires the utility to develop the L2 PSA by Safety Instruction IS-25 (June 2010) [26]. This instruction establishes that the scope of PSA must be Level 1 and Level 2 and must consider internal and external events in all modes of operation, considering also other radioactivity sources.

This instruction and Spanish Safety Guide 1.15 [27] are being updated and we know that the new requirements will be to submit:

- L2 PSA at full power for internal events every 5 years. Each plant must also submit a report every 9 months after the end of the refueling outage with the analysis and reasons to prove that it is not necessary to submit a new revision of the L2 PSA.
- L2 PSA at full power for internal flooding and fire events or L2 PSA at low power and shutdown for all events every 10 years.

Uncertainty/sensitivity analyses are also required for some issues.

This instruction includes applications of PSA related to modifications in design, specifications or procedures, including severe accident management guidelines, to prove that the risk of the plant remains within the acceptable level. The modification will be acceptable if it involves a risk reduction or if the rise of the risk is below the acceptable threshold indicated in the Spanish Safety Guide 1.14 [28]. The rise of the large early release frequency must be lower than $10^{-6}$ and the rise of the large releases frequency must be lower than $2.10^{-6}$. After the acceptance of an application, the PSA must be updated to represent the new situation.

On the other hand, Safety Instruction IS-36 (January 2015) [29] establishes the requirements that nuclear power plants must meet regarding the Emergency Procedure Guidelines (EPG) and Severe Accident Guidelines (SAG). This instruction requires the utility to develop SAG with the objective of mitigating the consequences of severe accidents in case that measures implemented by EPG did not have success in preventing core damage. This SAG must be developed for all modes of operation. For sites with more than one unit, SAG must be developed to be applied in case there was accident in several units simultaneously.

This instruction requires having instrumentation that could be used in severe accident conditions.

It also requires capabilities to protect containment for a selected group of accidents exceeding the design bases, in terms of:

- Containment isolation,
- Containment tightness,
- Containment pressure and temperature control,
- Combustible gases control,
- Overpressure protection,
- Reduction of high pressure melt ejection probability,
- Reduction of containment degradation by molten core-concrete interaction.

The selected group of accidents exceeding the design bases must consider deterministic and probabilistic analysis and expert judgement.

The utility must also have portable equipment for a long term SBO.

IS-36 also specifies that verification and validation of SAMG must be done in the same way they are going to be used. When a modification is done in SAMG, the scope of verification and/or validation must be done according to the importance.

It indicates that the utility must have a plan for maintaining and updating SAMG based on the current situation of the systems in the plant, the operating experience, the plants owners group considerations and the newest recommendations of the supplier of the equipment.

## 2.3.2 ROLE OF L2 PSA

The role of L2 PSA has evolved over time as their development has been consolidated. Initially, identifying vulnerabilities of the containment to the severe accident, determining a quantitative safety criterion and providing input into the development of the accident management guidance and strategies and finally, developing risk informed applications and decision making support incorporated into the rulemaking more recently. This new application field demands a quality and a higher level of development that guaranty an effective use on it:

- a sufficient level of detail, taking into account the most significant dependencies (operational and functional) and the procedures;
- the impact of the conservatism should be measurable;
- the detail of the models should permit changes related with the modifications to be analysed.

The Spanish regulation enables the use of the RG 1.200 to determine the L2 PSA quality for risk-informed application and decision making support.

## 2.3.3 SAM OBJECTIVES TO BE REACHED

For BWR, the operational emergency response is divided into Emergency Procedures Guidelines (EPGs) and Severe Accident Guidelines (SAGs). The EPGs define strategies for responding to emergencies and events that may degrade into emergencies up until it is determined that the core cannot be adequately cooled. The SAGs define strategies applicable after it is determined that the core cannot be adequately cooled. The SAGs strategies are grouped into two guidelines:

- RPV and Primary Containment Flooding: which objectives are to cool the core and core debris, shutdown the reactor and depressurize the RPV and prevent it from re-pressurizing
- Containment and Radiological Release Control: which objectives are to protect equipment in the primary and secondary containments, maintain primary and secondary containment integrity and limit radioactivity release into areas outside the primary and secondary containments.

## 2.3.4 SAM IN L2 PSA

Severe accident management strategies can be included into the L2 PSA if they are adequately treated into the operational guidelines and all the interfaces related with a successfully implementation are covered, i.e.

equipment and instrumentation survivability and impact of the environmental and stress conditions on the human actions. The uncertainty about all these aspects is significant higher; at any case the analysis is necessary not only to identify the relative importance but also to know the risk by indirect effects or due to an incorrect implementation or failure on it. In this sense, any SAM new strategy implemented in a NPP has to be analysed with the L2 PSA criteria for a risk evaluation.

The Stress Test evaluation on the Spanish NPPs, after Fukushima Dai-ichi accident, identified some SAM improvement to be implemented mainly based in the European experience and the state of the art: Passive Autocatalytic Recombiners and Filter Containment Vent System are the most significant new capacities installed on them. These systems are sized based on the severe accident specific impact for each plant, covering all potential phases of a severe accident progression and maximizing the risk to be mitigated. The mainly passive design of these systems reduces also a possible human error.

# 2.4 CZECH REP

SAM strategies verification and improvement received a new strong impulse after Fukushima Dai-ichi accident followed by stress tests performed around all Europe. Beside the stress tests, detailed revision of the emergency preparedness, event and accident feedback, system of procedure preparation (including SAM strategies) is a part of periodic safety review of the nuclear installations.

## *2.4.1 L2 PSA REGULATORY FRAMEWORK*

Till August 2015 there was no explicit legal requirement to conduct PSA in the Czech Republic and therefore there are also no regulatory probabilistic safety criteria required to be met by the operator. Nevertheless, a new "Atomic Law" which includes also PSA requirements should be approved by the Czech Parliament and released by the end of 2016. Currently, there are only recommendations for conducting L1 PSA (Safety Guidance BN-JB-1.6: L1 PSA). This guide is going to be extended so that also issues of L2 PSA will be taken into consideration. Considering qualitative safety goals, there is only general regulatory body recommendation to comply with IAEA probabilistic safety criteria [18].

## *2.4.2 ROLE OF L2 PSA*

PSA activities are mainly initiated by utility based on concrete NPP needs, experience of other countries and consideration of regulatory recommendations. The PSA activities are conducted to enhance the safety level of the plant operation in the frame of existing safety culture environment. The long-term (10 years) operation licence includes the requirement regarding Living PSA and risk monitoring to be performed.

The aims of PSA studies and models have been changed over time and the main present role is to support risk informed applications and decisions making in adequate ways, which are used as complementary to deterministic ones.

More details related to the topical status of PSA in the Czech Republic (and other countries) may be found in [19].

## 2.4.3 SAM OBJECTIVES TO BE REACHED

SAM strategies for Czech NPPs are based on Westinghouse SAM strategies, which may be divided into two main categories:

- Primary objectives,
- Secondary objectives.

According to Westinghouse philosophy, there are three primary objectives:

1. Return the core to a controlled and stable state,
2. Return or maintain containment to a controlled and stable state,
3. Terminate fission product releases into environment.

Besides the primary objectives, there are two general secondary objectives that should be achieved:

1. Minimize fission product releases,
2. Maximize equipment and monitoring capabilities.

The general principles stated above may be divided into 8 more specific principle actions:

- Inject into the Steam Generators,
- Depressurize the RCS,
- Inject into the RCS,
- Inject into Containment,
- Reduce Fission Product Releases,
- Control Containment Conditions,
- Reduce Containment Hydrogen,
- Spent Fuel Pool.

## 2.4.4 SAM IN L2 PSA

There are two different types of reactors in the Czech Republic – VVER-440 and VVER-1000. For the VVER-1000, finalization of L2 PSA for all modes and SFP is planned for 2017 (till now, only full power modes are completed). By contrast, L2 PSA for the VVER-440 was finished for all modes (incl. SFP) in 2015. This analysis was performed in EVNTRE code with separate models for full power modes, low power and shut down modes, and for SFP. The models include approximately 100 questions (each) which are based on (1) considered phenomena and (2) human actions described in SAM guidelines. In other words, the PSA models contain only such human actions, which are considered in SAMGs. Quantification of the human errors (required by SAMGs) was performed by using established HRA methods (THERP, ASEP, Decision Trees), while more conservative values of related PSFs (level of stress, type of step ...) was usually selected (compared to L1 PSA).

## 2.5 SWITZERLAND

In Switzerland, the issue of SAM has been resolved for all plants more than 10 years ago and SAM guidance was approved thereafter (or even before), hence all PSAs (Level 1 and 2) must include the SAM provisions in the models, including uncertainties in the implementation, where it is possible to quantify them.

The process of assessing the impact of SAM, and the redaction of SAMG for each of the installations in Switzerland (five units of four different reactor types, vendors and designs) has gone through successive iterations (in some cases lengthy and painful), following performance of Level 1 and 2 PSAs (initial and at least two periodic revisions for each plant) by operators and complete regulatory re-assessments. Most of the burden has been placed on the operators. Since the Swiss installations are so diverse, the results of the process cannot be generalized. Nevertheless, the efforts (and general considerations) have concentrated on the following SAM procedures:

- Depressurization of the primary system (or vessel). It is contemplated for both PWRs and BWRs, however for BWRs it may be a PREVENTIVE (of core damage) measure that should be implemented before core degradation accompanied by injection of low pressure systems. As such it is largely beyond Level 2 considerations.

- Injection of firewater (or any other potential source of non-borated water). This is valid for PWRs and BWRs. However, as for depressurization, for BWRs this may be used as a preventive measure.

- Implementation of containment venting through a Filtered Containment Vent System (FCVS). The implementation depends on the individual designs, and again results cannot be generalized. As for the previous measures, in BWRs (especially the smaller and older GE BWR-4), containment venting may be used as a PREVENTIVE measure (as a long term decay heat removal mechanism).

- Manual actuation of containment sprays (for BWR4). The measure was not normally contemplated due to the small capacity of the system, and it is currently considered in the base case analyses and sensitivity analyses.

- Addition of water in the secondary side for PWRs. This measure has been shown effective to reduce releases in specific calculations performed for one of the installations *[note: report is proprietory and confidential, cannot be listed]*.

- Implementation of hydrogen control in containment (for PWRs). The results of early PSAs had shown that implementation of containment venting was very much dependent on control of non-condensable gases in the containment. If the venting system were to be actuated at the "wrong" time, depressurization of the containment might have triggered a massive hydrogen combustion. Given that measurements of hydrogen concentration could not be relied upon with sufficient confidence, it was finally decided that Passive Autocatalytic recombiners should be installed in the containments.

Note that for the BWRs most of the "SAM" measures are codified in the Emergency Operating Procedures (EOP). As such, the operator interventions and systems are modeled in Level 1 (as well as in Level 2) with the same requirements (including HRA models) used for all Level 1 interventions.

Given that the measures are very much plant dependent, the results cannot be easily generalized either. For instance, one of the most recent analyses ([38], for a BWR-4) has shown the following (in Italics the original texts here and later are changed to make them better understandable in the context of this document):

> "The sensitivity analyses show that the presence of both the *Firewater Containment Sprays* and *FCVS* systems are very important in view of risks (and therefore the plant operator should ensure

that the mitigative measures associated with these systems are properly implemented). With availability of both systems, the "average" severe accident at this plant would result only in minor offsite consequences, while without both systems essentially all CDF would result in large releases."

On the other hand, earlier regulatory analyses for another BWR ([39], a BWR-6) had shown that, due to uncertainties in accident progression (and modeling assumptions) and makeup of severe accidents:

"*The utility's Level 2 PSA has shown that* either core damage is prevented or arrested prior to vessel breach, or containment failure <u>prior to core damage</u> is assured (*i.e., no SAM measure can help*)……*However, in the regulatory assessment* the FCVS has been found to be important in reducing the risk of activity of aerosol (*i.e., all released activity minus Noble Gases*) by 67%...."

Note that in both cases the risk is measured as integral of releases times frequencies.

In summary, the only common things about the results (including PWRs) is that the most effective SAM measures for (radioactive release) risk reduction seem to be those which could be considered as "preventive", i.e. early depressurization followed by injection of emergency water (firewater or other source of water), despite the uncertainties in the probability of core melt arrest and the uncertainties in hydrogen generation. A complete assessment of the effectiveness and potential for risk reduction however can be performed only on an installation-by-installation basis (at least for Switzerland).

PSA is currently used ONLY for the purposes delineated in the guideline of the Swiss Federal Nuclear Safety Inspectorate [16]. The required range of PSA applications is there defined as a minimum, which shall be carried out, in Chapter 6:

- 6.1 - Probabilistic evaluation of the safety level
- 6.2 - Evaluation of the balance of the risk contributors
- 6.3 - Probabilistic evaluation of the Technical Specifications
- 6.4 - Probabilistic evaluation of changes to structures and systems
- 6.5 - Risk significance of components
- 6.6 - Probabilistic evaluation of operational experience

Explanation of issue 6.1

For evaluation of the safety level in full-power operation a mean value limit for CDF $10^{-5}$ per year is used and for LERF $10^{-6}$ per year. The limit mean CDF $10^{-5}$ per year for FDF is defined for non-full-power operation. In case the given values are exceeded, measures to reduce risk shall be identified and - to the extent appropriate - implemented. Preference is to be given to measures that not only reduce LERF but also reduce CDF. LERF is defined as the sum of the frequencies of <u>all</u> accidents where releases of $I_{131}$ exceed $2\times10^{15}$ Bq within 10 hours after CD (the definition of LERF for the Swiss authority is in ([4], pg. 53). Note that, the concept of LERF is normally tied with the possible implementation of immediate offsite interventions, specifically timely early evacuation. In Switzerland (until present) immediate evacuation is not contemplated. Sheltering in secure locations (e.g. bunkered underground cellars) is required until the radioactive "cloud" has passed (this in respect to the 10 hours release in the definition of LERF). Eventual local relocation may be implemented later if measurements of radioactivity on the ground show that the affected area is not safe for long term habitation. Therefore in the

definition of LERF the component "E" or Early refers more to "early" health effects that, given the expected composition of releases from an NPP, are tied mostly to inhalation (and immersion in the passing radioactive cloud) of Iodine and especially of $I_{131}$, due to its relative abundance, high toxicity by inhalation and relatively longer half-life.

To complement the requirements on safety, a second regulatory limit is defined, i.e. LRF (Large Release Frequency), defined as the sum of the frequency of all accidents where the expected release of $Cs_{137}$ exceeds $2x10^{14}$ Bq ([4], pg. 53). This requirement is more closely tied with the potential for long term relocation needed in some areas due to deposition of radioactivity, and the regulatory limit is also $10^{-6}$ per year.


Explanation of issue 6.2

The balance among the risk contributions from accident sequences, components and human actions shall be evaluated. If any of the accident sequences, components or human actions are found by PSA to have a remarkably high contribution, measures to reduce risk shall be identified and - to the extent appropriate - implemented.

If an initiating event category contributes more than 60% to the mean CDF and its contribution is more than $6x10^{-6}$ per year, measures to reduce risk shall be identified and - to the extent appropriate - implemented.

If the ratio of the mean CDF to the $CDF_{Baseline}$ is greater than 1.2, measures to reduce risk due to planned or unplanned maintenance shall be identified and - to the extent appropriate - implemented.


Explanation of issue 6.3

In this part of [16], probabilistic evaluation of the completeness and the balance of the allowed outage times, component maintenance during Full power operation and changes to Technical Specifications are evaluated, where different parameters and limits related to CDF, FDF and LERF are defined to be compared with.


Explanation of issue 6.4

This part [16] requires assessment of the impact of structural and system-related plant modifications on the risk. In this case the analysis of the impact of modifications on CDF, FDF and LERF is required. And, even if the impact can be considered as insignificant and CDF calculated considering the modification remains below $10^{-5}$ per year, measures shall be identified and - to the extent appropriate - implemented in order to compensate for or to minimize the risk increase resulting from the plant modification.


Explanation of issue 6.5

According to [16] a component is regarded as significant to safety from the PSA point of view if Fussel-Vesely (FV) or Risk Achievement Worth (RAW) are as follows:

FV ≥ $10^{-3}$ or RAW ≥ 2


Explanation of issue 6.6

Different parameters, as for instance maximum annual risk peak, incremental cumulative core damage probability, and the trend of these safety indicators are evaluated. The contributors shall be reported in terms of four categories of "maintenance", "repair", "test" and "reactor trip" and dominant contributors shall be identified and

evaluated for both events and susceptibility to component or system failure. The probabilistic rating of events shall be established in relationship between incremental cumulative core damage probability ICCDP and the INES scale.

It should be mentioned, that the requirements and limits from issues 6.1 through 6.4 comply actually with IAEA 10 safety principles defined in [17]. These requirements also cover the relationship between PSA and INES similarly as the Common Risk Target developed by Jirina Vitazkova and Erik Cazzoli (CCA (see [15]). Nevertheless, none of these PSA applications mentioned above are used any longer for SAM verification, but actually for optimization of strategies only.

# 2.6 HUNGARY

Nuclear power plants with VVER-440/213 type reactors have specific design features; therefore these plants should have plant-specific AM strategy and SAMG. Selection of the possible severe accident management strategy was based on the results of a L2 PSA study (2000-2003).

Results of L2 PSA analyses were used for the development of a severe accident management strategy with the identification of the accident sequences that result in core damage, containment failure (or containment by-pass) and the release of fission products into the environment. The SAMG in Paks follow the generic Westinghouse SAMG approach with special adaptations for Paks VVER 440/213 units. The main elements of the strategy are as follows:

- depressurization of the primary system,
- water injection into primary system and in steam generators,
- in-vessel corium retention,
- decreasing fission product release,
- preventing containment failure (preventing overpressure due to steam production or hydrogen burn and preventing failure due to excessive vacuum),
- water injection into spent fuel pool.

For these strategies the following hardware changes have been implemented in the power plant:

- construction of reactor cavity flooding system,
- severe accident diesels for autonomous electrical supply for other SAM equipment,
- installation of passive hydrogen recombiners,
- bleed from ruptured steam generator to the containment,
- reinforcement of cooling circuit of spent fuel pool,
- modification of containment vent system to be used for filtered venting,
- installation of severe accident measurement system.

These modifications and also the SAMG were implemented until the end of 2014 at all units of Paks NPP. Some specific details of the Hungarian approach can be found in section 4. The containment long term cooling system has not been implemented yet. This independent system will condense the produced steam from in vessel melt retention more than 1.5 day after a cavity flooded. This system is under design now.

## 2.6.1 LEGAL REQUIREMENTS

Annex 3 of Govt. Decree No. 118/2011 [25] contains the legal requirements. The main points which are connected to SAM and PSA are:

"*3.2.2.4300. Implementation of specific design solutions or preventive accident management capabilities shall ensure that the occurrence frequency of the following accidents with catastrophic energy release in the reactor vessel or within the containment is infinitesimal:*

- *reactivity accidents with prompt criticality, including heterogeneous boric acid dilution,*

- *steam explosion, and*

- *hydrogen detonation.*

*3.2.2.4610. The required means of accident management shall be designed and accident management guidelines shall be devised for the efficient mitigation of the consequences of beyond design basis events analysed in detail, including severe accident processes resulting in a complete fuel meltdown, in such a way that any hazard posed to the environment and the population remains below a predefined, manageable level if the procedures and means of accident management work successfully.*

*3.2.3.0600. During the whole lifetime of the nuclear power plant the suitability of all interventions or modifications of nuclear safety related systems, structures and components that deviate from the authorized conditions shall be demonstrated with deterministic safety analysis or a combination of deterministic and probabilistic safety analyses.*

*3.2.3.1500. To define the exact risk of the nuclear power plant, to verify the fulfilment of relevant acceptance criteria, to evaluate the consistency and coherence of the design as well as to determine the suitability of the extended design basis a probabilistic safety analysis shall be performed.*

*3.2.3.1600. For the design of a nuclear power plant unit design level 1 and level 2 probabilistic safety analysis shall be developed, which considers all possible operating conditions, system configurations and all of the postulated initiating events for which it cannot be demonstrated by any other method that their contribution to the risk is insignificant.*

*3.2.4.0900. For all initial operating conditions and effects, excluding sabotage and earthquake, the collective frequency of severe accident event sequences resulting in large or early large releases shall not exceed $10^{-5}$/year, but with every reasonable modification and intervention $10^{-6}$/year shall be targeted. The fulfilment of criteria shall be demonstrated by Level 2 probabilistic safety analyses.*"

## 2.6.2 ROLE OF L2 PSA

The main role of L2 PSA before and after the modification is to demonstrate that the frequency of the large release decreased due to the changes. The next table and the explanation of the differences are a way to show the role of L2 PSA. The effects of modification can be seen in the change of frequency of different severity categories. The most severe is the first category and release can be large release in categories I-IV.

| Severity category | Cs release (A) [TBq] | Release category | Frequency [1/year] 2003 | Frequency [1/year] 2014 |
|---|---|---|---|---|
| I | $A > 10^5$ | 1 | $6.83 \cdot 10^{-8}$ | $2.52 \cdot 10^{-8}$ |
| II | $10^5 > A > 10^4$ | 2,3 | $3.88 \cdot 10^{-6}$ | $2.21 \cdot 10^{-7}$ |
| III | $10^4 > A > 10^3$ | 7 | $1.01 \cdot 10^{-6}$ | $1.15 \cdot 10^{-8}$ |
| IV | $10^3 > A > 10^2$ | 4,5,6,10 ,11 | $1.50 \cdot 10^{-5}$ | $7.55 \cdot 10^{-7}$ |
| V | $10^2 > A$ | 12,13 (8,9) | $1.52 \cdot 10^{-5}$ | $1.30 \cdot 10^{-5}$ |

Significant difference can be seen in category II, III and IV. Category IV was decreased by more than an order of magnitude due to the reactor vessel cooling. Category II was decreased due to the elimination of PRISE by bleed of the broken steam generator into the containment and elimination of containment early failure by hydrogen burn. Before installation of hydrogen recombiners, the recovery of the spray system is expected to cause hydrogen burn and containment early failure. The SAM and SAMG decreases the probability of category III (early containment failure with spray), because the SAMG prohibits the use of the spray system and recombiners decreases the hydrogen concentration in case of high steam concentration. The slight decrease of category I is thanks to the SAMG, the guideline of the primary system depressurization.

## 2.7 UKRAINE

The requirements and rules in the area of nuclear and radiation safety in Ukraine represent a hierarchical structure consisting of 4 tiers with the Laws of Ukraine at the highest tier, followed by the Government decrees (2nd tier), nuclear regulatory authority requirements (3rd tier), and guidance and recommendation documents (including the rules, guides and procedures developed by the Utility) at the lowest tier.

The fundamental principles of radiation safety, namely justification, limitation and optimization (ALARA), and the maximal allowed radiation doses for public and NPPs personnel are defined in the Laws of Ukraine "On the usage of nuclear energy and radiation safety", "On Human Protection from Ionizing Radiation". These fundamental principles and limitations are detailed and supplemented with other radiation safety criteria in Ukrainian Radiation Safety Rules (NRBU).

The quantitative safety criteria on core damage frequency and a frequency of large radioactive release to be met by Ukrainian NPPs are defined in the regulatory document "The General Provisions of NPP Safety" (NP 306.2.141-2008). In particular, for NPP units currently operating in Ukraine the large release[1] frequency

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[1] According to "The General Provisions of NPP Safety" (NP 306.2.141-2008), large release is radioactive release

estimate shall not exceed $10^{-5}$ per reactor-year and the target value is $10^{-6}$ per reactor-year[2]. Correspondent values for new NPP units are of one order of magnitude lower than for existing units. To demonstrate compliance of each unit safety to regulatory requirements the Utility is obligated to develop safety analysis reports (SAR). The assessment results are reviewed and renewed by the Utility periodically (over 10 years period) to account for changes in regulatory requirements, plant configuration, equipment characteristics, etc., and documented in periodic safety re-assessment (PSR) reports. Each SARs and PSR reports are the subject of regulatory review by the State Nuclear Regulatory Inspectorate of Ukraine and its technical support organization (the state enterprise "Scientific and Technical Center for Nuclear and Radiation Safety").

High level requirements of the General Provisions related to the safety assessment are further detailed in regulatory document NP 306.2.162-2010 "Requirements to the Safety Assessment" that defines the objectives and scope of the assessment, types of events (i.e., internal initiators, internal and external hazards) to be evaluated, operational states to be covered, basic requirements SAR and PSR reports contents, and in NP 306.2.099-2004 "General Requirements to NPP Units Long Term Operation Based on Periodic Safety Reassessment Results". More detailed guidance on SAR and PSR reports format and contents are provided in:

- RD-95 "Requirements to the Safety Analysis Report Contents for Operating NPP Units with VVER-type Reactors",
- KND 306.301-96 "Requirements to the Safety Analysis Report Contents for NPP Units with VVER-type Reactors at the Construction Licensing Stage",
- KND 306.302-96 "Requirements to the Safety Analysis Report Contents for NPP Units with VVER-type Reactors at the Commissioning Stage", and
- Utility guidance document "Requirements to PSR Reports for Operating NPPs".

## 2.7.1 LINK WITH LEGAL REQUIREMENTS

As described in previous section the Utility has to confirm compliance with the regulatory requirements documenting the results of safety assessment in SARs and PSR reports. Both the General Provisions (6.4.3 of NP 306.2.141-2008) and Requirements to Safety Assessment (4.2 of NP 306.2.162-2010) stipulate that the safety assessment shall comprise of deterministic and probabilistic analyses, including L1 and L2 PSA to justify that quantitative criteria on core damage and large release frequencies are met. Beside demonstration of compliance to the regulatory requirements the assessment results shall also be used to develop safety upgrade measures as well as to improve the emergency operating procedures (EOPs) and severe accident management guidelines (2.6 of NP 306.2.162-2010).

The safety upgrade measures and analytical evaluations are reflected and prioritized according to their safety significance (estimated based on PSA results whenever possible) in Complex (Integrated) Program of Ukrainian

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resulting in the conditions at the sanitary protection area border that require evacuation of public.

[2] In Ukraine there are several multi-unit sites. According to recent discussion between regulatory authority and utility, it was agreed that the LRF value of each individual unit will be compared with safety criterion separately.

NPPs Safety Improvement (CSIP). The Program is approved by the Government thus becoming the requirement of the 2nd tier of Ukrainian rules and regulations in the area of nuclear and radiation safety. CSIP measures directly associated with extension of PSA and development of SAMGs include:

- extension of existing SARs, in particular, with L2 PSA;
- improvement of existing PSA to cover extended set of internal and external initiating events (IEs) for all plant operation states, including IEs at spent fuel pools;
- development of Living PSA model;
- SA analysis, development of SAMGs.

Regulatory requirements related to SAMGs verification and validation are specified in the General Provisions. In particular article 10.9.3 of NP 306.2.141-2008) states that both EOPs and SAMGs require analytical justification and verification, and shall be validated using full scope plant simulator. However, in practice the last requirement is hard to implement nowadays for SAMGs validation due to the limitations of simulator analytical capabilities. Therefore the simulators are used for validation of operator actions and training till the transfer conditions from EOPs to SAMGs are reached, while SAMGs validation is performed using the tabletop method. Some activities to extend plant simulator capabilities for SA simulation have been initiated by the Utility recently.

The Utility guidance document on SAMGs Verification and Validation Guideline defines two major objectives of SAMGs validation:

- to confirm that SAMGs can be understood and implemented correctly by the operators, and provide sufficient details on strategy actions and conditions;
- to confirm technical correctness of SAMGs.

The Guideline requires to utilize the knowledge and expertise of operators and analysts to identify the areas of SAMGs improvement including an identification of alternative ways and means of SAM strategy implementation. The Guideline also defines a set of qualitative criteria to be assessed in the process of SAMGs validation.

## 2.7.2 ROLE OF L2 PSA

Currently L2 PSA for power operation state is completed for all Ukrainian NPPs and its extension to other operational states continues. In the framework of L2 PSA the severe accident progression scenarios were identified and their consequences were evaluated. These results were further used in developing the SAMGs and their analytical justification. In particular the scenarios considered in L2 PSA are used to define conditions of severe accident analyses to justify SAM strategies, and the frequency estimates allowed to identify and prioritize safety upgrade measures needed to mitigate SA consequences.

Examples of CSIP measures originated from existing L2 PSAs are:

- prevention of early containment failure[3] at the ex-vessel phase of SA;
- containment hydrogen control during beyond design basis accidents (including SA);

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[3] Here containment integrity loss caused by radial ablation of reactor cavity wall at the location of ionizing chambers channels is assumed.

- filtered containment venting.

It is worth mentioning that the number of safety upgrade measures for station blackout conditions (e.g., alternative means of SG feed) were identified based on PSA results and implemented at several units in Ukraine even before Fukushima-Daiichi accident occurred. This confirms effectiveness of PSA as a tool for identification of potential safety issues, development and prioritization of safety upgrade measures.

To cover uncertainties associated with SA progression a number of assumptions was introduced and utilized in L2 PSA, SAMGs and their analytical justification. Therefore based on the regulatory review results the areas of further improvements and additional study needs have been identified including:

- in-depth evaluation of SA phenomena;
- evaluation and implementation of additional measures for SAM (e.g., in-vessel melt retention for VVER-440 type reactors);
- qualification of equipment for SA conditions.

Correspondent activities have been scheduled in CSIP. In-depth Evaluation of SA Phenomena Program developed by the Utility (under regulatory review at the moment) envisages:

- collection and analysis of up-to-date results of SA phenomena studies including available experimental data;
- evaluation of MELCOR code capabilities to simulate phenomena, formulation of recommendations on input decks improvements or usage of specialized codes;
- re-evaluation of SA progression scenarios, SAM strategies and SAMGs improvement (on as-needed basis).

## 2.7.3 SAM OBJECTIVES TO BE REACHED

Ukrainian SAMGs utilize symptom-based approach and define specific actions aimed in:

- limiting the extent of core damage;
- avoiding or limiting radioactive releases to the environment;
- avoiding or delaying potential loss of containment integrity to extend time for Emergency Response Plans implementation;
- bring the unit back to a controllable state.

Corium cooling and retention inside the reactor pressure vessel is a key objective to be reached in order to regain control over the situation.

SAM strategies are selected based on the results of vulnerability assessment which provides essential information for understanding unit response to beyond design basis accidents (in particular, accidents involving severe core damage). Other valuable insights for developing SAM strategies were obtained from:

- evaluation of the international experience in the analysis of severe accidents and development of SAMG,
- probabilistic safety analysis,
- analysis of severe accident phenomenology,
- generic studies and analyses performed for the same type of similar NPP designs,
- overview of the existing procedures, identification of their limitations,

- analysis of the availability of instrumentation under severe accident conditions and of limitations in SA identification and monitoring,

- identification of existing means for SAM,

- evaluation of plant-specific and generic operational experience applicable to NPP unit being analyzed.

The following strategies were selected for Ukrainian NPPs:

- reactor coolant system (RCS) depressurization,

- safety injection to RCS,

- SFP feeding,

- secondary circuit depressurization,

- steam generators feed,

- containment spray,

- hydrogen concentration control in the containment,

- containment venting,

- ex-vessel water injection.

According to the General Provisions (5.3.4 of NP 306.2.141-2008, level 4 of the defense-in-depth concept), any available means shall be utilized to cope with beyond design basis accidents and return the plant into controllable state. To support BDBA and SA strategies implementation a set of plant safety upgrade measures are implemented or scheduled including additional water and electrical power supplies, hydrogen recombiners, etc. Completion of these measures and of correspondent justification will in turn require updating of L2 PSA results and SAMGs.

# 2.8 BELGIUM

## 2.8.1 REGULATORY FRAMEWORK AND ROLE OF L2 PSA

In the nineties of previous century, the first L2 PSA was performed for certain Belgian NPPs but it was limited to the analysis of containment response with the aim of investigating dominant containment failure modes. There was no source term analysis and it considered full power operational state only.

The previous L2 PSA has supported the implementation of Passive Autocatalytic Recombiners in all Belgian NPPs to reduce the risk of containment failure due to $H_2$ burn. Sensitivity studies considering some severe accident management actions have shown their beneficial impact on containment failure probabilities.

In the framework of the first common Periodic Safety Review of the Belgian NPPs and considering the WENRA Reference Levels, L2 PSA has been updated in Belgium.

The WENRA Reference Levels issued in 2008 have been implemented into the Belgian regulations. The WENRA Belgian action plan was established in 2007 and was including L2 PSA related actions. The L2 PSA update has taken into consideration most of these actions. Accordingly, L2 PSA has been performed for all Belgian representative NPPs and it has included the source term analysis and the shutdown states.

The main objectives of the L2 PSA update were the following ones:

- Identification of containment failure modes;

- At power and shutdown states;

- Source term assessment;

- Assessment of Severe Accident Management Guidance.

To answer to these objectives, the L2 PSA update consisted of the extension of the previously developed Accident Progression Event Tree (APET): the APET is generic for all Belgian NPP (specificities of all units are included), considers the implemented Severe Accident Management Guidance and is extended for source term analysis. It has been based on the NUREG-1150 large event tree approach. The containment fragility curves were established for every representative unit. The supporting calculations were performed with MELCOR 1.8.6. Methodology for basic event quantification were developed with detailed sections on the use of expert judgement (based on NUREG-1150) and HRA methodology (based on level 1 HRA methodology, THERP and SPAR-H methodologies). Homemade tools to help the quantification process were also developed (regarding hydrogen risk analysis for example).

In L2 PSA, the impact of the SAM is evaluated through sensitivity studies. Generally, it is demonstrated that the SAM helps to decrease:

- the probability of vessel failure of about 40 to 60%;
- the probability of structural containment failure of about 20 to 50%;
- the probability of BMMT[4] of about 50 to 70%;
- and the probability of very important FP releases after vessel failure of about 20 to 50%.

Furthermore, it is shown by sensitivity studies that, among all SAM actions, the RCS injection and recirculation related actions are the most impacting ones on the L2 PSA results.

Presently, the L2 PSA is extended to consider internal hazards such as internal fire and flooding.

## 2.8.2 SAM OBJECTIVES

The SAM objectives in Belgium are the generic WOG SAMG objectives.

The WOG SAMG has three primary goals:

- Terminate fission products releases from the plant,
- Maintain or return the containment to a controlled, stable state and,
- Return the core to a controlled, stable state.

It has also interim and secondary goals:

- Minimize fission products releases while achieving primary goals,
- Maximize equipment and monitoring capabilities while achieving the primary goals.

## 2.9 SLOVENIA

The requirements for mandatory performance of probabilistic safety analyses were set in 2009 by regulation JV9 (entitled Rules on operational safety of radiation or nuclear facilities). The scope, quality and way of PSA have been defined. Finally, mandatory use of PSA is required in working processes relevant to radiation or nuclear safety, to identify needs for modifications to the facility and written procedures for its operation, including the needs for severe accident management measures and in assessing risks involved in the facility operation.

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[4] BaseMat Melt Through.

## 2.9.1 SAM VERIFICATION AND IMPROVEMENT PROCESS

In the frame of Krško NPP, L2 PSAs have been performed in the 1990's. They analysed also selected beyond design basis accident conditions. The analyses include conditions with core damage and containment failure, known as severe accidents. These analyses were the basis for preparation of plant specific severe accident management guidelines (SAMG). Thereafter a full scope simulator was built in 2000 and plant specific SAMG was prepared. (based on generic Westinghouse Owner Group (WOG) SAMG, which have been validated by WOG). Krško plant specific SAMG were validated on the full scope simulator, capable of simulating also severe accident. Krško SAMG validation has been also performed during exercises on emergency preparedness. Each exercise gave opportunity for improvement of SAMG. In 2011 the safety upgrade program was requested on modernization of safety solutions for prevention of severe accidents and mitigation of their consequences.

## 2.9.2 SAM OBJECTIVES TO BE REACHED

The primary goals of the WOG SAMG are to terminate fission product releases from the plant, to prevent failure of intact fission product boundaries and to return the plant to a controlled stable state. A controlled, stable core state is defined as core conditions under which no significant short term or long term physical or chemical changes (i.e., severe accident phenomena) would be expected to occur. A controlled, stable containment state is defined as containment conditions under which no significant short term or long term physical or chemical changes would be expected to occur.

## 2.9.3 ROLE OF L2 PSA

The Krško NPP has in place L2 PSA analysis for full power modes, including all external hazards. A full scope PSA (including Level 2) for low power and shutdown events shall be implemented by the end of 2015.

The level 2 study is an integral part of the Krško Individual Plant Examination (IPE) and it supports the overall objectives of the IPE for the plant:

- to develop an appreciation of severe accident behavior,
- to understand the most likely severe accident sequences that could occur at its plant,
- to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and
- if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The regulation JV9, Article 40 defines application of PSA. By use of PSA, the facility operator of a nuclear facility shall continuously monitor the cumulative facility risks arising from all the implemented modifications. Further Article 41 of JV9 defines restrictions of the increases of risks due to modifications. In the case of the Krško nuclear power plant, the increase of risks may in no case exceed $1 \cdot 10^{-6}$ per year as regards the core damage frequency and $1 \cdot 10^{-7}$ per year as regards the large early release frequency.

The PSA model is updated regularly by the plant after each larger modification or at least once per fuel cycle. This is true also for mitigating severe accident measures in SAMG. After the accident management guidance is completed, it should be verified whether all important accident sequences are covered and whether risks are

reduced accordingly. The influence of modifications is evaluated using severe accident code. L2 PSA is performed also before the modifications are implemented. For example, the following proposed options regarding the containment integrity improvements were evaluated by L2 PSA:

1. Passive Autocatalytic Recombiners (PARs),

2. Containment Filtered Vent (FV) System,

3. Combined PARs and FV System.

The risk reduction assessment is based on the existing NEK PSA studies. It is done by evaluating the CET structure and identifying the relevant branches. The probabilities of the identified CET branches are then appropriately modified by employing the bounding assumptions regarding the risk from hydrogen burn (in the case of PARs) or the risk from containment overpressure (in the case of FV).

## 2.9.4 LINK WITH PERIODIC SAFETY REVIEW

The Slovenian National Ionizing Radiation Protection and Nuclear Safety Act, in addition to the routine reviews, require preparation and submittal of Periodic Safety Review (PSR) as prerequisite for obtaining of the operational license for the Krško NPP. PSR is performed each ten years. The purpose of the PSR is to assess the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects on the safety of the nuclear power plant. During PSR also the deterministic methods used for development and validation of emergency operating procedures and accident management program at the plant are reviewed.

## 2.9.5 LINK WITH LEGAL REQUIREMENTS

Ionising Radiation Protection and Nuclear Safety Act (from 2011):

- *"Article 112 (amending a licence): (4) A licence shall be changed ex officio:*

  *- when this is required for the protection of the environment or the life or health of the population for public benefit,"*

Regulation JV5: Rules on Radiation and Nuclear Safety Factors (from 2009):

- *"Article 62(4) Upon the plant life extension of the Krško nuclear power plant or extension of the service life of its SSCs, if approved, the facility operator shall undertake a study of the response of the nuclear power plant to severe accidents in accordance with Chapter 1.12 of Annex 1 and, based on the findings of this study, propose any appropriate measures and implement them as quickly as practicable.*

- *Article 62 (5) The Krško nuclear power plant shall comply with the requirements of subparagraphs 10 and 11 of Article 40 of these Rules at the latest within three years following the approval of the design life time extension of the nuclear power plant or its SSCs.*

- *Chapter 1.12 of Annex 1 – "1.12 Severe accidents": In addition to design-basis events referred to in the subparagraph 1 of Article 4 of these Rules, the response of the nuclear power plant to selected severe accidents shall be analysed to minimise harmful impacts of releases of radioactivity. A sequence of events shall be established to determine and implement reasonable preventive and mitigating measures. A combination of engineering judgement, deterministic and probabilistic methods may be employed, subject to realistic assumptions and reasonably adjusted acceptance criteria."*

The regulation JV5 from 2009 required from the plant to upgrade its systems, structures and components (SSCs) to enable coping with severe accidents after the plant lifetime was extended. After the Fukushima Dai-ichi accident the Slovenian Nuclear safety Administration (SNSA) ordered the plant to implement these measures in advance (based on Article 112 of Ionising Radiation Protection and Nuclear Safety Act).

## 2.10 BULGARIA

In Bulgaria, for the Kozloduy NPP, units 5 and 6, in order to ensure a high level of safety throughout the plant's operating lifetime, a systematic safety reassessment, termed Periodic Safety Review (PSR) is carried out at regular intervals to assess the cumulative effects of plant ageing and modifications, operating experience, technical developments and new regulatory aspects, as is defined at [20]. It is noted that the PSA is one of the main subjects of the PSR.

### 2.10.1 L2 PSA REGULATORY FRAMEWORK

In Bulgaria, the Nuclear Regulatory Agency (NRA) requests the development of a L2 PSA to support the periodic safety review of the Kozloduy NPP. The activities for development and use of PSA for the Kozloduy NPP, units 5 and 6 are planned and carried out in accordance with the requirements of the "Regulation on Ensuring the Safety of Nuclear Power Plants" [20].

### 2.10.2 LINK WITH LEGAL REQUIREMENT

The regulation [20] contains quantitative criteria for severe accidents - limits of a large release requiring immediate protective measures. Furthermore, these criteria are specified for new and for existing NPPs.

According to the regulation, Art.10:

"*(4) The frequency of a large radioactive release into the environment that requires undertaking of immediate protective measures for the population shall not exceed $1.10^{-6}$ events per NPP per year.*"

According to the "Transitional and final provision" from the current regulation regarding the existing NPPs:

"*3. the frequency of large radioactive release into the environment that require implementation of urgent protective measures for the public shall be lower than $10^{-5}$ events per NPP per year.*"

Furthermore, regarding to the probabilistic safety analysis, the regulations contains in the Art.21, p.6, the next requirements:

"*(1) Probabilistic safety analysis shall be carried out with the objective to:*

*4. assess the frequencies of severe core damage and large radioactive releases to the environment;*

*5. evaluate the frequencies and the consequences of the external events specific to the site;*

*6. identify SSCs that require design improvements or changes in operational procedures, leading to decrease of severe accident frequency or mitigation of their consequences;*

*7. assess the emergency operating instructions.*

*(4) Probabilistic safety analyses shall be used to support the deterministic assessments in the decision making for plant design and operation, for assessment of necessary changes of SSCs, operational limits and conditions, operating and emergency operating procedures and training programs of the operating personnel.*"

## 2.10.3 ROLE OF L2 PSA

The Section 7 "Emergency procedures and severe accidents management" from the Safety Guide "Use of PSA to Support the Safety Management of NPPs" [21], describes the PSA use in the development of the emergency procedures and the measures for severe accidents management, including in the assessment of their modification:

*"7.9. The effectiveness of the existing alternative or additional systems, equipment and measures should be assessed in the procedures for management of severe accidents with the help of PSA.*

*7.10. In the case of operator actions to manage accidents, PSA should clearly present the operator actions referring to specific emergency instructions and accident management procedures. To support such applications the method of analysis of human reliability used in PSA should be able to predict the impact of procedure changes.*

*7.11. The reflection of the operator actions in PSA level 1 supports the improvement of the procedures for accident management for these actions to prevent a severe damage of the core. PSA Level 2 with limited scope reviews the strategies for mitigation of the consequences from severe accidents.*

*7.12. The mitigation of the consequences from severe accidents should include identification and categorization of emergency sequences based on PSA, together with descriptions of the NPP behavior and weaknesses.*

*7.13. PSA supports the understanding of accident development, the identification of successful ways to manage and the strategies related to it, as well as the prioritization of safety characteristics to reduce risk."*

The integral demonstration of NPP behavior with the PSA methodology supports the reduction of potential negative effects from certain measures.

The current L2 PSA for the Kozloduy NPP, units 5 and 6 is based on the L1 PSA and represents the status of the units up to 2007 year concerning the systems and procedures included in L1 PSA, and status up to 2011 for the systems and procedures (e.g. SAMG) related to containment and severe accident aspects [41].

Presently, the L2 PSA study includes full power operation modes, low power and shutdown modes, and Spent Fuel Pool (SFP). The analyses are performed by using RiskSpectrum program code. The L2 PSA study is comprised of two parts: interface PSA and CET. Two types of models for CET have been developed: one for conditional probabilities calculations and another for the integrated model – a set of simplified CET's for each PDS group. The purpose of the first model is to be able to perform quick calculations and for sensitivity analyses as well. The simplified CET's are used for integral calculation of the model [41]. The supporting calculations are performed with MELCOR code. On the basis of PSA, a Risk Monitor was developed, which is used in everyday operation of Kozloduy NPP. Furthermore, other PSA applications are also developed or are under development - risk-informed testing, risk-informed maintenance, risk-informed Technical Specifications [30]. Also, based on the present PSA results, the effectiveness of the SAMGs are assessed qualitatively [41].

## *2.10.4 SAM OBJECTIVES TO BE REACHED*

The main objective of SAM strategies is to prevent and mitigate possible core damages as well as further radioactive releases beside the system of NPP physical barriers. The other main role of SAM is to return the critical safety functions in normal state as well as to ensure long term NPP safe state.

For the Kozloduy NPP, units 5 and 6, the purpose of the SAMG is to define beforehand specific actions to perform in order to [22] :

- avoid or limit radioactive releases;
- avoid or delay the possible loss of containment integrity in order to give more time to activate the Emergency Plan for public protection;
- bring the unit back to a controllable state. Keeping the corium under water inside the pressure vessel is a key objective in order to regain control of the situation.

Actions recommended in the SAMG aim at limiting the risk of radiologically significant radioactive releases in the short- and mid-term (a few hours to a few days).

According to the Updated National Action Plan of Bulgaria [31], the development and implementation of significant severe accident management measures at Kozloduy NPP was initiated before the Fukushima accident, in the frame of a Modernization Program of units 5 and 6 (in the period 2000-2007). The most substantial severe accident management provisions, implemented before Fukushima Dai-ichi accident, include installation of containment filtered venting system (scrubber type), installation of hydrogen recombiners, development of L2 PSA and SAMGs.

For the Kozloduy NPP, units 5 and 6, symptom-based emergency operating procedures (SB EOP) were developed on the bases of protection of fundamental safety functions with approach similar to the Westinghouse one, as step-by-step procedures with description of main and alternative operator actions in two-column format. SBEOP are controlled and maintained adequate to existing powers through strict on-site rules on verification and validation which envision multiple checks before these are introduced into operation. Analytic validation is conducted on the original technique provided by Department of Energy (DOE USA) [30].

Criteria for transfer from SB EOP to SAMG were defined.

SAMGs for units 5 and 6 were developed on the basis of system analysis of the processes and phenomena during severe accidents (PHARE Project BG 01.10.01, "Phenomena investigation and development of SAMG"). The purpose of system analysis is to define the basis for knowledge of processes and phenomena, occurring at progress of severe accidents, all the technical means (equipment and systems), which allow for reaching of the set purposes in the course of severe accident management (SAM). SAMGs were developed which follow the format of SB EOP. Strategies defined for severe accident management are the following [31]:

- Pressure reduction in the primary circuit;
- Pressure reduction in the secondary circuit;
- Water injection to the primary circuit;

- Water injection to the secondary circuit;
- Pressure reduction in the containment.

Following the stress tests, additional severe accident management actions have been implemented, e.g. the project for plugging of the ionization chamber channels in the walls reactor vessel cavity to prevent basement melt-through and early containment by-pass; the project for installation of wide-range temperature sensors to monitor the reactor vessel temperature and installation of additional recombiners. The scope of the EOPs has been extended to cover reactor shutdown states and spent fuel pool accidents. Also, an update of the L2 PSA was done taking into account L1 PSA results from 2010 [41]. The SAMGs have been developed to involve two sets of documents – SAMGs for Technical Support Team and SAMGs for MCR/ECR [31].

# 3 <u>IDENTIFICATION OF SAM STRATEGIES</u>

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. PSA can very well quantify the risk reduction due to such SAM, or help designing the SAM (e.g. positioning the PARs inside the containment, or define the design requirements for venting systems).

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 particular after the Fukushima 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 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:

- It has to be decided whether only safety grade equipment may be considered, or whether also operational equipment can 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 table below presents the main risk issues and objectives in case of severe accident phenomenon, and some corresponding SAM strategies able to avoid or to limit radiological releases. This table is just a set of examples and does 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.

*This table can be completed during the report review phase and additional examples are welcome.

| 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) |
| 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 |
| 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 or vessel uplift | RCS depressurization<br><br>Control in-vessel water injection<br><br>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)<br><br>Limit water in reactor cavity<br><br>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) |

| Risk/Objectives | SAM Strategies or design provisions |
|---|---|
| **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)<br><br>Optimize geometrical features: large area for corium spreading, large width of the basemat (core-catcher for new design, upgrade for existing NPPs)<br><br>Suppress containment bypass in the basemat (e.g. close pipes)<br><br>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 …<br><br>Monitor containment atmosphere conditions<br><br>Control containment spray system activation |
| Control the containment pressurization | CHRS, FCVS, …<br><br>Apply containment venting system<br><br>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<br><br>Isolate leakage<br><br>Use circuit with intermediate heat exchangers to avoid direct contamination of the environment<br><br>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 a SFP accidents | Strategy if the SFP is inside the reactor containment |
| | Strategy if the SFP is outside the reactor containment |

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

## 4.1 INTRODUCTION

This chapter does not repeat information on the basic phenomenology but concentrate on the impact of SAM strategies in the L2 PSA and on the potential impact of internal and external hazards on those strategies. However, human reliability modelling for event sequences emanating from external hazards is not yet state of the art. Only few comments in the following sections address this issue.

For each strategy (and its combination with other strategies), the following items are analyzed:

- types of support studies and analysis needed for L2 PSA to evaluate the impact of the strategy;
- specificities regarding the plant design and the different reactors states;
- way to implement the strategy into L2 PSA: human reliability, equipment accessibility, equipment and I&C availability, positive and negative impact of the strategy …;
- application for SAM strategies improvement.

## 4.2 EMERGENCY TEAMS (EMERGENCY TEAM ACTIVATION, SAMG ENTRY, ROOMS HABITABILITY, COMMUNICATION, INSTRUMENTATION)

### 4.2.1 EDF&IRSN, FRANCE

#### 4.2.1.1 Emergency team composition and activation

##### 4.2.1.1.1 Status

During the implementation of the emergency operating procedures (EOPs), when the kinetic of the accident allows it, the entrance into the SAMG is preceded by the establishment of the national crisis organization and the internal emergency plan.

The emergency provisions in France include special organizational arrangements and emergency plans, involving both the operators and the authorities:

- the internal emergency plan is initiated on criteria into the EOPs and defines the organizational provisions and the resources to be implemented on the site; its implementation induces the activation of the national crisis organization;
- the off-site emergency plan to protect the population in the short term of the event.

On the plant site, the crisis organization is composed of:

- the operating crew in control room[5];
- a local organization with several teams (for site and reactor management, communication, support to the crew in control room, local operation and radioprotection, local measures in environment, …).

Outside the plant, this organization is completed by:

- the National Emergency Response Team: EDF experts (with AREVA technical support) assisting in analysis and decision making;
- the Nuclear Rapid Action Force (FARN): after the Fukushima accident, EDF has strengthened the emergency response organization both with equipment and human resources. Integrated in the emergency response organization, the FARN's main aim is to be capable of responding in less than 12 hours to reinstate water, electricity and air supply at the nuclear power plant where the accident has occurred. It is fully operational in an autonomous manner within 24 hours;
- the French Nuclear Regulatory Authority (ASN) and IRSN technical support experts.

### 4.2.1.1.2 EDF L2 PSA modelling

The available team composition, depending on time, is taken into account while evaluating the probability of Human failure for required actions in the L2 PSA. In case of hazards, feasibility of human actions is checked and a penalization factor can be added if the hazard makes the human action more difficult.

### 4.2.1.1.3 IRSN L2 PSA modelling (900 and 1300 MWe PWRs)

The modeling L2 PSA hypotheses for availability of crisis teams are the following:

- local emergency teams: 2 hours are necessary for taking control of the crisis management procedures after activating the internal emergency plan;
- national crisis organization: 4 hours are necessary to be operational after activating the internal emergency plan;
- longer periods of time are considered if the internal emergency plan is not activated;
- at the moment, FARN organization is not yet taken in consideration in L2 PSA, as well as the impact of hazards on the crisis team availability.

For each PDS from L1 PSA interface, thermal-hydraulics representative sequences are used. According to the time when the crisis teams are operational, SAMG actions are considered at SAMG entry or later.

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[5] In France on the 900 MWe, 1300 MWe and N4 series, the operating team is composed of 2 operators, a shift supervisor, a safety engineer and several field operators.

## 4.2.1.2 SAMG entry

### 4.2.1.2.1 Status

EPR (PWR): Criteria for SAMG entry are Core temperature above 650°C or dose rate. In the shutdown states, if Core temperature is not available the only criterion is dose rate.

French Fleet (PWR): Criteria for SAMG entry are Core temperature above 1100°C or dose rate. The higher temperature level for Core temperature for the Fleet regarding EPR criterion is because EPR has been originally designed to manage core melt accidents, so efficiency of specific design components are taken into account for EPR (for example in-vessel water injection strategy regarding core catcher efficiency, cf. §4.3.1.1). In the shutdown states, if Core temperature is not available the only criterion is dose rate.

Following gamma dose rate criteria, taking into account time after shutdown, are used (both EPR and Fleet):

| T= time after shutdown | S = criterion (dose rate inside containment) |
|---|---|
| T < 1 hour | S = 500 Gray/h (5 $10^4$ Rad/h)[6] |
| 1 hour < T < 6 hours | S = 100 Gray/h |
| 6 hours < T < 5 days | S = 50 Gray/h |
| 5 days < T < 1 month | S = 10 Gray/h |
| T > 1 month | S = 5 Gray/h |

### 4.2.1.2.2 EDF L2 PSA modelling

The SAMG entry failure is taken into account in the Human failure evaluation for actions required in the L2 PSA, via the "diagnosis failure" parameter, depending on the accidental scenario (see §4.3.1.2).

### 4.2.1.2.3 IRSN L2 PSA modelling

Regarding availability of measures, four situations can be distinguished:

| | No SBO and/or instrumentation failure | SBO and/or instrumentation failure |
|---|---|---|
| RPV close | Core temperature and dose rate measures available in the control room | Core temperature available in the electrical building (local action) - dose rate measure unavailable |
| RPV open | Core temperature measure potentially unavailable - dose rate measure available in control room | Core temperature and dose rate measures unavailable – SAMG entry is done on request of crisis team |

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[6] The value of 500 Gray /h inside the containment before 1 hour is the limit between dose rate due to clad failure and dose rate due to core melt.

Success or failure for SAMG entry depends on the availability of measures and the kinetic of the thermal-hydraulic scenario, based on the timeframe between internal emergency plan activation and SAMG entry. For example, the table below gives the failure probabilities for RPV close (these values are obtained with the HORAAM model, cf. 4.3.1.3.2):

| Time between emergency plan activation and SAMG entry | No SBO | SBO |
|---|---|---|
| Fast kinetic (< 4h) | $10^{-2}$ | 1 |
| Medium kinetic (4h< <12h) | $10^{-3}$ | 1 |
| Slow kinetic (>12 h) | $10^{-4}$ | $10^{-1}$ |

Late SAMG entry is also considered in some cases (fast kinetic, non-activation of internal emergency plan).

All timeframes used for the HRA are obtained from ASTEC calculations which consider success or failure of SAMG actions. For all PDS (and associated accident scenario calculated with ASTEC), the time needed for activation of local and national emergency teams has been considered.

## 4.2.1.3 Room habitability

### 4.2.1.3.1 Status

Habitability of the control room can be challenged because of hazards or because of radiations.

In case of hazards leading to control room evacuation (for instance smoke from fire or unavailable operating system commands from the control room), a dedicated control room with spare commands is used. Regarding radioactivity, control room habitability has been checked for severe accident without loss of containment (even with Filtered Containment Venting).

### 4.2.1.3.2 EDF L2 PSA modelling

Room habitability is supposed to be ok in the L2 PSA for severe accident without loss of containment. For each local action required in the SAMG, radiological feasibility of this action has been checked (for example local action for opening the venting system described in § 4.8.1 for the French Fleet).

If the containment is lost, no credit for mitigation action is taken into account in the L2 PSA at all, so the rooms' habitability is not a concern.

### 4.2.1.3.3 IRSN L2 PSA modelling

Room habitability is not explicitly considered in IRSN L2 PSA except for the containment venting (higher failure rate if the site is already contaminated).

Some analyses have already been performed in this area, for example, the risk of contamination of the main control room after some containment penetrations leakages (contamination by ventilation).

### 4.2.1.4 Communications

**4.2.1.4.1 Status**

Communications between site and national crisis team are provided by 2 diversified wire systems. Additionally communication by satellite is available.

**4.2.1.4.2 EDF L2 PSA modelling**

A communication failure probability is supposed negligible regarding the other failure probabilities in the L2 PSA.

**4.2.1.4.3 IRSN L2 PSA modelling**

Availability of the communication means (in the control room and to crisis teams) is given for each PDS from L1 PSA interface, on the basis of data reliability and electric power availability. This influences the factor "Information and measurement means" of the HORAAM model (§ 4.3.1), and the HRA failure probability.

### 4.2.1.5 Instrumentation

**4.2.1.5.1 Status**

EPR (PWR): Core temperature and dose rate values are available to detect severe accident.
French Fleet (PWR): Core temperature and dose rate values are available to detect severe accident. Moreover, measure of the containment pressure allows detecting if the pressure reaches the threshold for containment venting (between 5 and 6 absolute bar). New sensors for hydrogen detection and vessel rupture are being installed.

**4.2.1.5.2 EDF L2 PSA modelling**

Instrumentation failure probability is supposed negligible regarding the global failure probability to enter the SAMG and is not modeled in the L2 PSA.

**4.2.1.5.3 IRSN L2 PSA modelling**

Additional (simple) modelling for instrumentation is being developed. Availability of core temperature/dose rate/containment pressure measures can be known for each PDS from L1 PSA interface (on the base of data reliability and electric power availability). This will influence factors "Information and measurement mean" and "Difficulty for the operator" of the HORAAM model (§ 4.3.1), and the HRA failure probability.

## 4.2.2 IEC, SPAIN (BWR)

### 4.2.2.1 Status

SAMG entry is required due to:

- Certain amount of hydrogen or radioactivity in drywell and containment (core damage).
- RPV Water level below minimum water level for steam cooling (2/3 TAF approx.).

Unlike PWR designs the core temperature is not a criterion of core damage control, there is no instrumentation for that and the coolability is conservative guaranteed with a control of a sufficient submergence of the core.

SAM Team consists of three roles:

- One shift manager (reserve) in charge of SAMG of RPV.
- One shift supervisor (reserve) in charge of SAMG of Containment and SFP.
- One coordinator, which is a person that is part of the Emergency Team with the role of coordinating both SAMG and communicate with the Emergency Manager Team.

Control Room Team communicates to the SAM Team when SAMG entry conditions are reached, meanwhile Control Room Team operate with EPG. When shift manager of SAM Team considers that they are prepared to take control of the situation they communicate this to the Control Room Team and start to operate with SAMG, maintaining the support of the Control Room Team. SAMG are used until the end of the emergency is declared. There is no need for re-operation with EPG because all safety functions of the Plant are also controlled from the SAMG.

SAM Team activation and arrival to the corresponding room is in a procedure and is trained in the emergency drill.

### 4.2.2.2 L2 PSA modelling

L2 PSA takes credit of the SAM Team decision on SAMG in a similar way on EPG, based on the similar structure between SAMG and EPG and the support of the Control Room Team into the SAM Team. There is no methodology to evaluate the impact due to differences between Control Room Team decision and SAM Team decision but it is recommended to include sensitivities about that. These sensitivity analyses permit us to identify which decisions are more critical and define to optimize a decision structure for that.

The functionality of Control Room has been evaluated with PSA for several events including flooding and fires. Other external hazards have been evaluated for Control Room with deterministic criteria.

Although there is not a seismic PSA developed, a previous evaluation has determined that the equipment and instrumentation in the Control Room is guaranteed in case of earthquake.

An evaluation of habitability of control room in a SBO with vessel failure and containment venting has demonstrated that it is not necessary to leave the Control Room if the emergency filtered system is activated before the venting.

SAM Team Room is considered in the same area of the Control Room so it has the same characteristics in terms of habitability.

Analyses have been done to evaluate the availability of instrumentation in severe accident conditions with the alternatives for the main functions.

After the Fukushima Dai-ichi accident, new capacities have been implemented into the Plant to permit an efficient management of the emergency when the Control Room is unavailable due to an external hazard (e.g. aircraft crash). An alternative emergency room has been built, sufficiently separated from the Control Room with enough means for survival during the emergency, including the severe accident phase, and also Extended Damage Mitigation Guidelines (EDMG) have been implemented covering all the areas involved during the emergency (management, technical, operational, radiological, medical, external support...). If a specific PSA for external hazards were developed these capacities would be considered into the analyses.

## *4.2.3 TRACTEBEL, BELGIUM*

### 4.2.3.1 <u>SAMG entry</u>

The criterion for SAMG entry is based on core exit thermocouples. The SAMG should be applied if the core exit temperature is higher than 650°C and actions to cool the core are not successful in the Emergency Operating Procedures. Practically, the transition to SAMG is requested from the following WOG Emergency Response Guidelines:

- FR-C.1 « Response to Inadequate Core Cooling » (when all recovery actions have failed),
- ECA-0.0 « Loss of all AC power »,
- FR-S.1 « Response to nuclear power generation/ATWS ».

An alternative entry criterion has been defined to consider the possible unavailability of the core exit thermocouples. This criterion is based on containment dose rate taking into account the time since shutdown.

In WOG SAMG, the Technical Support Centre (TSC) staff is responsible for the application of SAMG. In case TSC staff is not operational, the control room staff has one dedicated guideline to allow them managing the severe accident.

In L2 PSA, SAMG entry is quantified based on human reliability analysis (see dedicated chapter for Belgium).

### 4.2.3.2 <u>Support of L2 PSA studies for staff trainings</u>

The results of L2 PSA studies are used to support the staff trainings. The importance and positive impacts of the human actions are emphasised during trainings by comparing L2 PSA results with and without consideration of certain human actions. The human actions concerned are typically:

- SAMG entry;
- Containment isolation in case of failure of the isolation signal;
- Isolation of the affected steam generator(s) in case of SGTR;
- RCS injection and depressurisation;
- Containment sprays operation;
- Refilling of RWST for long term SAM.

## *4.2.4 SSTC, UKRAINE*

### 4.2.4.1 <u>Emergency team composition and activation</u>

### 4.2.4.1.1 Status

Onsite and offsite Utility accident response organization, responsibilities, communication procedures and interfaces are defined in specific Utility guide entitled "Provisions for severe accident management" and in NPPs Emergency Response Plans.

Prior to on-site crisis center activation, the actions prescribed by EOP and SAMGs are performed by shift personnel being directed by unit shift supervisor. Transition to SAMG procedures is directly defined in relevant EOP steps as well as in SAMG entry conditions (see details in ch.4.2.4.2). When transition conditions are reached or other

indications of expected accident progression to SA stage are available the local (on-site) crisis center team, plant senior management and emergency response brigades are notified. Notification time prescribed by the utility guide is 15 min for on-site crisis center and 30 min for emergency brigades.

Following activation of the on-site crisis center the overall responsibility for arranging the accident management, coordination of various tasks performed by teams involved, activation of emergency plan, requesting off-site support, etc., is vested upon the On-site Emergency Response Manager (ERM) who is advised by the on-site engineering support group (ESG). ESG includes experienced NPP staff with in-depth knowledge of plant systems and operation as well as of accidents management and consequences mitigation.

Responsibilities of the on-site ESG include:

- evaluation of current state of the plant and its systems/elements;
- selection of appropriate accident management strategies and actions;
- establishing and providing recommendations on accident management measures to be implemented to the ERM chief;
- instructing operators on actions to be performed based on the decisions of ERM chief.

Transferring of SAM functions from MCR operators to the on-site ESG is performed according to written procedure which is the part of SAMGs set.

In addition to on-site crisis center, the Utility emergency situations commission is activated with the main goals:

- to coordinate activities at the Utility level;
- establish communication and information exchange with Regulatory Authority, related ministries and other outside institutions and organizations;
- provide support (e.g., to mobilize emergency brigades, equipment or other resources needed to support affected NPP) as requested by the On-site Emergency Response Manager.

Other resources (e.g., local and regional civil protection authorities, departments of the State Emergency Services of Ukraine) can be involved depending on the nature and severity of the accident as defined in Emergency Response Plans.

### 4.2.4.1.2 L2 PSA modelling

Considering that L2 PSA for Ukrainian NPPs was actually performed before SAMGs are developed it accounts mainly the basic (the most important) actions (see ch. 2.7.2) and does not reflect all SAM details including those associated with emergency response organization. To account for the latest SAMGs revision the update of L2 PSA is to be performed during periodic safety re-assessment. It shall be noted that SAMGs are arranged so as all critical and time-sensitive actions are to be performed regardless of crisis center actuation. Therefore emergency response modeling is expected to affect L2 PSA results only after additional SA management measures other than those controlled from MCR are implemented at Ukrainian NPPs.

### 4.2.4.2 SAMG entry

### 4.2.4.2.1 Status

Though exact EOP to SAMG transition conditions and their justification are still being actively discussed between the industry and regulatory authority, preliminary conditions are specified in current EOPs and SAMGs to provide operator with direct guidance to be applied if the accident occurs.

Similar criteria are used for both VVER-1000 and VVER-440 reactors. For power operation case transition from EOPs to SAMGs is prescribed at core exit temperature exceeding 450 °C. As a backup criterion the time of SG feed water unavailability in the case of SBO is used. Criteria for SFP SAMG entry are SFP level decrease below 430 cm for VVER-1000 and 300 cm for VVER-440, high SFP decrease rate (loss of SFP coolant accident indication), and timing criteria (depending on SFP load) for the cases when level measurement instrumentation is not available (SBO case).

For unsealed reactor state an abnormal increase of radioactivity criterion is selected as an indication of SAMG entry necessity.

### 4.2.4.2.2 L2 PSA modelling

In L2 PSA, the SAM actions are modeled depending on severe accident scenario under evaluation. Human reliability analysis for specific L2 PSA questions is based on the estimates of:

- time needed for diagnostic and mechanistic actions;
- available time span for actions implementation.

Other factors are also considered (see, examples in ch.4.5.5.3.2).

Available time span is determined from the results of deterministic analyses and is estimated as a difference between the maximal action initiation time that allows to reach the actions objectives and time required to reach correspondent SAMG entry criteria (decision time to initiate actions is also considered).

### 4.2.4.3 Room habitability

### 4.2.4.3.1 Status

To ensure main and emergency control room (MCR, ECR) operability and habitability during SA the air conditioning systems were modified to withstand harsh conditions and seismic impacts, and for VVER-440/213 units (Rivne NPP Units 1,2) iodine filters were installed. In the case of SBO it is envisaged to provide power supply to the emergency lighting, communication equipment, MCR and ECR air conditioning and heating from mobile diesel-generators.

Considering the measures taken to ensure MCR and ECR operability and habitability it is expected that external hazards will have insignificant effect (if any) on human reliability of controlling FCVS operation.

### 4.2.4.3.2 L2 PSA modelling

In current L2 PSA for Ukrainian NPPs the room habitability is not explicitly addressed. This issue was indicated during the state review of PSA results as the one to be addressed in future L2 PSA and SAMG improvement activities.

## 4.2.4.4 <u>Communications</u>

Information exchange and coordination of various emergency response activities are performed using on-site announcement and notification system, phone system lines, paging, radio communication systems and other means. Notification of plant management, emergency response staff, on-site and Utility crisis center staff is performed by automated notification system.

Communication between engineering support group and MCR operators, reporting of the important parameters, plant and equipment state is currently performed using dedicated direct phone lines and announcement system. Accident and post-accident monitoring system (see details in ch.4.2.4.5) which is planned to be installed by the end of 2017 shall also provide data transfer to the on-site crisis center.

Testing and maintenance of existing communication systems is performed on the regular basis, therefore their failure is expected to be very low comparing to other factors affecting SAM.

## 4.2.4.5 <u>Instrumentation</u>

Abilities of the design instrumentation systems which are available to operator to monitor the plant state are limited mainly to normal operation, transients and design basis accidents conditions. To overcome this limitation correspondent measure on accident and post-accident monitoring system implementation is included in Complex (Integrated) Program of Ukrainian NPPs Safety Improvement with a scheduled completion in 2017. This measure involves upgrade of the existing instrumentation, installation of extended instrumentation for BDBA and SA and their integration into computerized monitoring system with all components qualified for correspondent accident and hazards conditions. The system is equipped with batteries allowing its autonomous operation during long-term station blackout for at least 8 hours. Further electrical supply of the system is provided by the same mobile diesel generator.

Extended instrumentation set for beyond design and severe accidents includes monitoring of the next main parameters:

- core exit temperature (extended range), cold and hot legs temperatures, RPV temperature, temperature inside containment and in SFP
- reactor and SG levels, SFP and containment sump level;
- primary circuit and containment pressure;
- hydrogen, oxygen and/or steam concentrations inside containment;
- dose rate monitoring inside containment.

Upgrade of design instrumentation systems was initiated and implemented partially at some of NPPs units. When installed the system will improve operator capabilities on monitoring the plant state and need to be addressed in updated EOPs and SAMGs, as well as accounted in L2 PSA.

## 4.2.5 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| Emergency team activation, rooms habitability, instrumentation, … | |
|---|---|
| Emergency Team | L2 PSA shall be able to identify scenarios: <br> • where the emergency teams can fail to manage the beginning of the severe accident due to context factors like time constraints, external hazards, extreme conditions …, <br> • 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. |

# 4.3 ANALYSIS OF HUMAN ACTIONS

## 4.3.1 EDF & IRSN, FRANCE

### 4.3.1.1 Status

Entry into SAMG represents a parting from EOPs in terms of strategy. The SAMG objective is no longer to prevent core meltdown, but rather to maintain the confinement of radioactivity. The EDF SAMG defines two types of actions:

- "Immediate actions": short term actions that can be launched immediately and that do not need National Emergency Response Team expertise. To implement these actions, operators need only the permission of the Local Management Command Centre, which is required to apply SAMG;
- "Delayed actions": long term actions that can be delayed and / or that need National Emergency Response Team expertise. Implementation of delayed actions may require a real-time analysis of their advantages and disadvantages, depending on the context.

### 4.3.1.2 EDF L2 PSA modelling

In L2 PSA, HRA type C is performed by application of two types of methods:

- for actions embedded in fault trees, a simplified and penalizing method is used;
- for actions embedded in event trees, a detailed and more realistic method is used.

For both methods, the following times and durations are defined:

- $t_C$ : time at which the operating organisation reaches a configuration that is compatible with execution of the operator action. In practice the maximum value among the following events is retained: occurrence of the initiating event (or other event posterior to it) requiring implementation of an action, appearance in the control room of the first information representative of the event, the arrival / organization of the crisis teams authorized to decide whether an action is to be implemented.

- $t_P$ : time of reaching the criterion or criteria on which the instruction of the action(s) are conditioned.

- $t_U$ : time after which execution of an action no longer meets its objective, which is aimed, directly or indirectly, at restoring or maintaining one of the 3 safety functions.

- $d_A$ : duration necessary to execute an action. It corresponds in practice to the sum of the durations necessary for execution of the action itself (i.e. not including execution of the diagnosis or drawing up a strategy or a prognosis) and travel prior to implementation of the action (case of actions carried out locally).

- $M_A$ : anticipation margin. $M_A = t_P - t_C$

- $M_L$ : starting margin. $M_L = t_U - t_P - d_A$

The chart below represents the chronology of a typical case:



*Time*
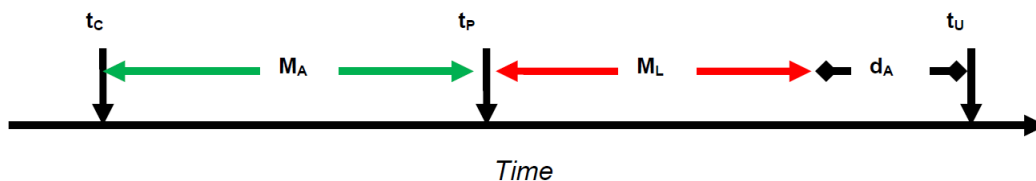
### 4.3.1.2.1 Actions embedded in fault trees

In order to obtain representative PSAs, recovery of a failed automatic signal or necessary reconfiguration of a system shall be taken into account (e.g. for L2 PSA: local re-closure of containment isolation valves, local re-closure of the equipment hatch, etc.).

For obvious reasons of technical feasibility, these types of HRA-C missions cannot be modeled as function events in the event trees, but shall instead be modeled within the system fault trees, closer to the modeled components. Consequently and given the generic nature of these specific HRA-C missions in the PSA, it is not possible to use a detailed HRA-C method. Therefore, they are modeled with a "basic event" whose probability of failure is obtained with a simplified and penalising method. This simplified method is mainly based on available time to carry out the action, using $M_A$ and $M_L$ margins defined above.

Dependencies with I&C (commands and information required for the action) can also be taken into account in a simplified way. On the other hand, dependencies with other HRA-C missions (credited in L1 or in L2 PSA) cannot be addressed simply. However, it shall be verified (with a coverage objective of 95%) for all assessed consequences that there are no minimal cutsets containing two (or more) HRA-C missions with (at least) one modeled via the

simplified method. If necessary, the consideration of a probability dependency for these missions shall be analysed.

### 4.3.1.2.2 Actions embedded in event trees

The detailed method used for actions embedded in the event trees as "function events" is based on a certain number of concepts stemming from the detailed HRA method MERMOS used for L1 PSA at EDF :

- Systematic approach to operation,
- Functional model of the operating organisation,
- Construction of failure scenarios dealing with following functions: Diagnosis, Prognosis, Strategy and Action.

However, it consists of the following notable differences:

- Simplification of the failure scenario construction sequences by reducing parameters to be estimated by the analyst,
- Use of a more general terminology compatible with extension of the operator organisation to the crisis teams and the various guides completing or replacing post-accidental operating procedures (SAMG in particular),
- The possibility of dealing dependencies with I&C failures.

## 4.3.1.3 IRSN L2 PSA modelling (900 and 1300 MWe PWRs)

HRA is performed by application of two models:

- for immediate actions, the PANAME[7] model is applied (this model has been initially developed to provide HRA data for L1 PSA models); modules of PANAME are adapted to the severe accident context; the main adaptation consists of introducing a worsening context factor, and a short time for the recovery of the operator's errors ; moreover, a distinction is made between actions for the reactor (mission time of 30 min) and actions for the containment (mission time of 1 h);
- for delayed group of actions, the IRSN has developed the HORAAM model (Human and Organizational Reliability Analysis in Accident Management, [23]) to provide HRA data for L2 PSA models. It is designed to model the failure of the emergency response while managing a core melt situation.

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[7] PANAME is a first generation HRA method. It is adapted from EDF HRA method "FH6".

### 4.3.1.3.1 Immediate actions modeling

The model is based on the time when the internal emergency plan is activated ($T_{IEP}$) and the time when the core exit temperature reaches the threshold for SAMG application ($T_{SAMG}$). As an example, the table below gives human error probabilities for immediate actions: for the reactor (e.g. RCS depressurization) and for the containment (e.g. containment isolation), without and with SBO.

| | | Human error probabilities for immediate actions | | | |
|---|---|---|---|---|---|
| | | Reactor | | Containment | |
| Kinetic of the scenario | Degree of involvement of the emergency response teams | Without SBO | SBO situations | Without SBO | SBO situations |
| $T_{SAMG}$ - $T_{IEP}$ < 2 h | Emergency organization not available | 1 | 1 | $4.9\ 10^{-2}$ | 1 |
| $T_{SAMG}$ - $T_{IEP}$ < 4 h | Only local emergency organization available | $1.9\ 10^{-2}$ | 0.27 | $2\ 10^{-2}$ | 1 |
| $T_{SAMG}$ - $T_{IEP}$ > 4 h | National emergency organization available | $7.2\ 10^{-3}$ | 0.11 | $7.2\ 10^{-3}$ | 1 |

**Dependencies of human errors between L1 and L2 PSA**

A lot of sequences of core melt are sequences where a human action has failed during EOPs application before SAMG's application. Therefore, actions which are both required by the EOPs and by the SAMG can't be considered independently.

L1 PSA human errors are taken into account in L2 PSA according to the following 3 steps:

- 1 - a decision tree developed by IRSN is used to determine the level of L1/L2 dependency, which is based on the number of human errors in EOP (transmitted by the PDS attributes) and the kinetic of the scenario: 4 levels of dependencies are obtained: low, moderate, high, complete;
- 2 - failure probabilities are calculated with the PANAME model without consideration of any error in the L1 PSA sequence;
- 3 - final failure probabilities are calculated with a "grid of dependency effect", based on the Swain-Guttman model.

The table below gives human error probabilities for immediate actions applied to the reactor, taking into account L1/L2 dependencies.

| | Level of L1/L2 dependency | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | Without | | Low | | Moderate | | High | | Complete | |
| | No SBO | SBO | No SBO | SBO | No SBO | SBO | No SBO | SBO | No SBO | SBO |
| Emergency organization not available | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 | 1 |
| Only local emergency organization available | $1.9 \cdot 10^{-2}$ | 0.27 | 0.1 | 0.29 | 0.19 | 0.36 | 0.53 | 0.63 | 1 | 1 |
| National emergency organization available | $7.2 \cdot 10^{-3}$ | 0.11 | 0.05 | 0.15 | 0.15 | 0.23 | 0.5 | 0.55 | 1 | 1 |

The table below gives human error probabilities for immediate actions applied to the containment, taking into account L1/L2 dependencies and without any SBO (in case of SBO situations, it is assumed a systematic failure).

| | Level of L1/L2 dependency | | | |
|---|---|---|---|---|
| | Without | Low | Moderate | High |
| Emergency organization not available | $4.9 \cdot 10^{-2}$ | 0.1 | 0.19 | 0.53 |
| Only local emergency organization available | $2 \cdot 10^{-2}$ | 0.1 | 0.19 | 0.53 |
| National emergency organization available | $7.2 \cdot 10^{-3}$ | 0.05 | 0.15 | 0.5 |

**4.3.1.3.2 Delayed actions modeling**

The "Human and Organizational Reliability Analysis in Accident Management" (HORAAM) model was developed to take into account human actions in L2 PSAs [23]. The model is based on the observation of the French crisis exercises. From this study, seven influence factors were selected. Experts were asked to discuss the relevance of these influence factors (table below) and to quantify their weight. Human error probabilities outcome from the decision tree varies from $10^{-4}$ to 1:

- a combination of several "unfavorable" modalities rapidly leads to a failure probability of 1;
- if all the influence factors take the modality "favorable", the failure probability is $10^{-4}$.

| Influence factors | Description |
|---|---|
| Degree of involvement of the crisis organization | Local crisis organization on the plant site or the whole national crisis organization. |
| Decision time | Time necessary to obtain, check and process information and make a decision about the required action. This influence factor has 3 modalities "short", "medium" or "long". |
| Information and measurement means | This influence factor refers to the quality, reliability and efficiency of all measurements and information available in the control room and means of transmitting them to crisis teams. This influence factor has 2 modalities "satisfactory" or "unsatisfactory". |
| Decision difficulty | Difficulty in taking the right decision. This influence factor has 3 modalities "easy", "medium" or "difficult". |
| Difficulty for the operator | Difficulty of the action independently of work conditions: quality of the procedures, experience and knowledge in the control room or in the plant. This influence factor has 2 modalities "easy" or "difficult". |
| Difficulty induced by environmental conditions | This influence factor describes on-site conditions in which the actions decided upon, have to be performed (radioactivity, temperature, smoke, gas, exiguity,..) This influence factor has 2 modalities "normal" or "difficult". |
| Scenario difficulty | Difficulty of the global context of the current accident scenario in which a decision must be made. This influence factor has 2 modalities "easy" or "difficult". |

#### 4.3.1.3.3 Hazards modeling

At the moment, human reliability modelling is only applied for internal events (L2 PSAs for external hazards are not yet been developed). Nevertheless, both PANAME model and HORAAM model are well suited to take into consideration the hazard context. For instance, the parameters "Difficulty for the operator", "Difficulty induced by environmental conditions" and "Scenario difficulty" of the HORAAM model can be adapted in case of external hazard.

## 4.3.2 GRS, GERMANY

Dynamic reliability analysis of operator performance

An essential element of a L2 PSA taking into account SAM is the analysis of operator performance. Plant operators, system components and the physical process are interacting parts of a complex system like a nuclear power plant that has to respond to specific conditions and events. The traditional methods of fault tree and event tree analysis are not capable of explicitly treating the time dependent interactions of the plant process variables (temperature, pressure etc.), system components and operator actions.

An approach to overcome these problems has been proposed in the framework of a dynamic reliability analysis [24]. A so-called "Crew Module" allows simulating human performance as a separate dynamic process, which evolves in parallel and interacts dynamically with the system process. The "Crew Module" can operate in combination with the probabilistic dynamics tool Monte Carlo Dynamic Event Tree (MCDET) and any deterministic dynamics code simulating the process and system dynamics. This combination allows considering mutual dependencies between system components, physical process quantities, human actions and any relevant random events influencing the man-machine-system.

In the "Crew Module" alternative operator strategies, communications between crew members, time durations of actions, recovery of failures, the influence of any performance shaping factors (i.e. stress, complexity of task etc.), errors of omission as well as errors of commission in principal can be modelled. Stochastic events regarding the performance of human actions can be taken into account. The "Crew Module" does not account for the mental process and the cognitive behavior of the operators.

Applying the "Crew Module" in the framework of a dynamic reliability analysis allows a more realistic modelling of dynamic operator plant interactions and can be important, for instance, in assessing the efficiency and relevance not only of emergency operating procedures (EOP) but also of maintenance, repair and any other human performances.

This approach could also be used for the verification and improvement of SAM strategies. So far, however, the application of the MCDET method has been restricted to specific aspects of a PSA (e.g. the emergency operating procedure "secondary side bleed and feed" which is required in a German pressurized water reactor (PWR) after the loss of steam generator feedwater supply) and requires substantial computational effort.

It can be concluded that combining a model of crew behavior with a model of the plant systems and physical processes is a very promising approach for validating SAM actions. These models can be implemented in a common probabilistic framework. With such a tool it is possible to realistically evaluate SAM actions. However, the approach requires a fast and stable running simulation tool and an efficient Monte Carlo framework. The latter seems to be available, see e.g. the GRS MCDET development, and also the simulation tools typically applied in L1 PSA seem to be suitable for such purpose. However, the presently existing integral accident analysis codes (e.g. MELCOR or ASTEC) applied in L2 PSA need much user support for complex and meaningful analyses. Therefore, applying a combination of crew models, event simulation tools and Monte Carlo techniques can be recommended for extended L1 PSA. Also specific important topics in L2 PSA should be addressed by such methods, e.g. the issue of transition from high pressure core melt scenarios to low pressure scenarios. However, for a complete L2 PSA, improvement is still needed for the accident simulation tools.

## 4.3.3 TRACTEBEL, BELGIUM

### 4.3.3.1 L2 PSA HRA methodology

In the framework of the update of L2 PSA for Belgian NPPs, the Belgian Safety Authorities requested the development of L2 PSA HRA methodology.

The HRA for L2 PSA aims to generate Human Error Probabilities (HEP) values that reflect the possibilities for human failure within the structure of decision-making and implementation in the SAMG. This is achieved by identifying and representing the key points of failure within the process of applying the SAMG and by appropriately selecting and representing SAMG actions within the L2 PSA model.

The L2 PSA HRA methodology is mainly based on the HRA methodology for the L1 PSA of the Belgian units. The Technique for Human Error Rate Prediction (THERP) methodology [34] and the Standardized Plant Analysis Risk – Human reliability analysis (SPAR-H) methodology [35] complete the set of references used. The L1 PSA methodology for HRA is applied as far as possible for consistency to the L2 PSA. The THERP methodology is used as

a basis for the determination of the different factors of Human Error Probability (HEP) in L2 PSA. SPAR-H methodology is used complementary to THERP methodology as additional information not present in THERP methodology is provided. The THERP and SPAR-H methodologies are American methodologies (NUREG/CR) and their use is consistent with the American approach selected globally for the Belgian L2 PSA.

The human errors that are considered in the L2 PSA are related to the human failure events that can occur during accident scenarios in which core damage has occurred. Each human failure event is a basic event that has to be quantified. The HRA methodology aims at establishing rules for the quantification of those human failure events. In the methodology, a human failure event is defined as an event for which failure results from the failure to perform a task (in which diagnosis and actions are included).

To help for the quantification and to take into account the specificities of SAMG, each task is decomposed in successive subtasks. The selected successive subtasks are the following ones:

1. Failure to check the parameters

2. Failure to enter the correct guideline

3. Incorrect assessment of availability of means

4. Omission to perform actions due to evaluation of negative impacts

5. Failure of decision for actions

6. Omission or commission error to transmit action to the control room

7. Omission or commission error in the execution of transmitted action

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

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

According to these explanations, the different subtasks can be classified as diagnosis or action subtasks: five subtasks are related with diagnosis and two with actions. It has to be noticed that for very early phase tasks, not all the selected subtasks are applicable.

**Classification of the different subtasks.**

| Subtask | Type of subtask |
|---|---|
| Failure to check the parameters | Diagnosis |
| Failure to enter the correct guideline | Diagnosis |
| Incorrect assessment of availability of means | Diagnosis |
| Omission to perform actions due to evaluation of negative impacts | Diagnosis |

| Failure of decision for actions | Diagnosis |
|---|---|
| Omission or commission error to transmit action to the control room | Action |
| Omission or commission error in the execution of transmitted action | Action |

For each subtask, the human error probability has to be assigned. The human error probability formula that has to be applied for each subtask is an adaptation of the formula that is used for the L1 PSA that takes into account the possibility to recover the human error.

The general formula that is used for HEP of a subtask in L2 PSA is:

$$P_{HEw/od} = P_B \times P_{NR1} \times P_{NR2} \times PSF$$

with  $P_{HEw/od}$: Human Error Probability without dependency

$P_B$: Base (nominal) probability

$P_{NR1}$: Non Recovery Probability by a member of the staff responsible for the subtask

$P_{NR2}$: Non Recovery Probability by the members of the checker staff for the subtask

PSF: Performance Shaping Factor

The base or nominal probability is the probability of the human error on a subtask without considering any possibility of recovery or the influence of Performance Shaping Factors (PSF).

Due to the specificities of the organization during a severe accident, the possibilities for recovery are specific to L2 PSA: recovery is possible by a member of the staff responsible for the subtask or by the members of another staff that can be considered as the "checker" of the subtask. According to the WOG SAMG usage, all the subtasks for SAMG are under the responsibility of the Technical Support Centre (TSC) staff, except the last subtask which consists in the execution of the action and is under the responsibility of the Control Room staff. For each subtask, the possibility for recovery by a checker staff is screened according to the organization and the SAMG. The possible checker staffs are the Control Room staff or the Emergency Operation Facility (EOF) staff (or the Technical Support Centre staff for the subtask concerning the execution of the action). The recovery possibilities for the different subtasks are shown next:

**Recovery possibilities for the different subtasks.**

| Subtask | Responsible staff | Checker staff |
|---|---|---|
| Failure to check the parameters | TSC | Control room |
| Failure to enter the correct guideline | TSC | - |
| Incorrect assessment of availability of means | TSC | Control room |
| Omission to perform actions due to evaluation of negative impacts | TSC | - |
| Failure of decision for actions | TSC | EOF |
| Omission or commission error to transmit action | TSC | - |

| to the control room | | |
|---|---|---|
| Omission or commission error in the execution of transmitted action | Control room | TSC |

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

Dependency according to SPAR-H methodology is also applied between the different tasks. According to SPAR-H, the following combination of factors contributes to error dependency:

- Same crew: relates to similar mindset, use of similar heuristics, tendencies to tunnel vision…;
- Same location: the control; display, or equipment must be the same or located within the same relatively restricted area, such as the same panel;
- Lack of additional cues: additional cues exist if there is a specific procedural callout or a different procedure is used, or additional alarm(s) or display(s) are present;
- Close succession of the next human error probability (from within seconds to a few minutes).

The ratings of the various combinations of the factors correspond to zero, low, moderate, high or complete dependency. For each rating, one formula defined in SPAR-H has to be applied. In case the dependency is complete, the human error probability equals to one.

The analyst has to identify the dependencies between the different human failure events in relation with the structure of the APET. Success of an intervening human failure event (HFE) breaks the dependency between a preceding HFE and a subsequent HFE.

### 4.3.3.2 Use of expert judgment

Due to the fact that the methodologies used as references (THERP and SPAR-H) are not developed specifically for L2 PSA HRA and its related specific severe accident management guidance and decision-making process, it is considered, according to [36], that the assignment of human error probabilities is not so obvious and particularly the assignment of the different levels for performance shaping factors. Consequently, the confidence level for the quantification requests expert judgment. It should be applied to quantify each human failure to enhance the confidence for human error probabilities.

Accordingly with NUREG-1150 approach [37], the Belgian L2 PSA takes into consideration the use of expert judgment for the basic event quantification. Expert judgment is only applied in case the assessment of the confidence level for the available sources of information, either literature information or plant-specific engineering calculations (or combination of both), indicates that the quantification cannot be supported by the available sources of information.

Expert judgment is: "Expression of opinion, based on knowledge and experience, that experts make in responding to technical problems. Specifically, the judgment represents the expert's state of knowledge at the time of response to the technical question" [37]. Practically, expert judgment includes the following steps [37]:

1. Selection of issues and participants
2. Elicitation training
3. Presentation of issues
4. Preparation of issue analyses by the participants
5. Discussion of issue analyses and elicitation of expert's judgments
6. Aggregation of expert's judgments

The full set of MELCOR 1.8.6 supporting calculations are used by the experts to help them assigning the probabilities. The base cases do not include any action. Several variants for the base cases take into account accident management actions (with different possibilities depending on system availabilities and on the decision process).

Concerning the documentation, a synthesis of the quantification is made for each basic event and can be found in the basic event quantification document. The final assigned probabilities are provided but also the assigned probabilities of each participant. The references used and the reasoning of each participant are briefly explained.

### 4.3.3.3 Application of the HRA methodology

The application of the HRA methodology for L2 PSA has provided human error probabilities ranging from $1x10^{-3}$ to $3x10^{-1}$ without dependency and from $3x10^{-2}$ to $6x10^{-1}$ with dependency. The two most reliable actions with dependency are the transitions of RCS injection and containment sprays operation into recirculation mode during the late phase. The two less reliable actions taking into account dependency are the establishment of the existing connection between the spray lines and the safety injection lines during the very early phase (which is considered as a highly complex action and not usually trained) and the use of alternative means for RCS injection during the early phase.

A specific sensitivity analysis focusing on the impact of the human actions on the severe accident outputs of interest has been performed.

This analysis is divided into four parts:

- The global impact analysis focuses on the influence of performing all the human actions belonging to the SAMG. The analysis shows the positive impact of SAMG actions on the end-state of the containment and on the releases (early and late).

- The phase impact analysis looks at the influence of performing the set of human actions belonging to a specific phase. The analysis shows the positive impact on the end-state of the containment and on the releases (early and late) of performing human actions during the very early and early phases. The actions performed during the early phase have a greater impact on the containment end-states and late releases than the actions performed during the very early phase.

- The function impact analysis focuses on the influence of performing the set of human actions belonging to a specific function. The function impact is looked at for the functions which can be dealt with through

different human actions: containment heat removal, RCS injection and RCS depressurization. The analysis shows the positive impact on the end-state of the containment and on the releases (early and late) of performing human actions belonging to the functions described above. The functions RCS injection and RCS depressurization have a greater impact on the containment end-states and late releases than the containment heat removal function.

- The <u>independent impact analysis</u> detailed the sensitivity coefficient of each human action; it allows ranking the different human action according to their influence on the output variability. The analysis shows the importance of the transition to the SAMG and of the injection actions. The positive impact on the early releases of isolating the steam generator in case of a tube rupture has been emphasized. The positive impact on the early and late releases of isolating the containment has been demonstrated. Other human actions having a main positive influence on the late releases have been identified through this process: the addition of NaOH to the water of the containment sumps for pH control or the interconnection between the safety injection lines and the spray lines.

Practically, SAMG improvements are planned focusing on the SG isolation action and on the action describing the possible connection between the safety injection lines and the spray lines.

Moreover, during SAMG training program, extra attention will be given on the following issues:

- Bypasses of the containment are sequences inducing early releases to the environment.
- The impact of the containment isolation action is positive regarding fission product releases.
- The transition to the SAMG is important for an efficient management of the severe accident.
- Among the set of early actions, the affected SG isolation action in case of SGTR is of main importance.
- The RCS injection and depressurization actions are important for prevention of vessel failure.
- The containment heat removal actions (mainly the use of the sprays and fan coolers) have a positive impact on the reduction of the late releases.

## 4.3.4 NUBIKI, HUNGARY

In the framework of the SAM verification using L2 PSA, the reliability of human actions was determined. During this work, the conditions of the human interventions, the available time for decision, for intervention, the quality of the guides and the environmental circumstances were taken into account. A practical example is given below.

Electric energy is necessary to open valves for reactor cavity flooding which is a nodal question in the containment event tree. In case of station blackout including loss of AC power from the grid and from the emergency diesel generators only the additional severe accident diesel generator can supply energy. The probability of electric energy support by the severe accident diesel generator depends on the answers on the following when going through the whole accident management process:

- Who is supposed to make a decision to use the severe accident diesel? Is that person available? (An external event can block access routes and communications by the responsible person(s))?
- Who will perform the transport of the severe accident diesel? Where will be those persons before the initiating event?

- Where is the diesel stored? An earthquake or other external event can cause damage to the building of the mobile diesel thus causing diesel failure or blockage to transportation?

- How many and what kind of transport vehicles are available? Is it possible to start up and operate that vehicle(s) (environmental conditions)? If the dedicated vehicle fails, can replacement or repair be realistically credited? If yes, how long time would it take?

- Can there be blockages on the transport route that would prevent the transport? (The transport is supposed to start at the beginning of the event; therefore radiation is not a problem. However, the consequences of an external event (structural damage, debris, fallen trees, etc. can impact on the transport area).

Overall, the above questions address the contextual conditions and performance influences that are seen important to assess the likelihood of successfully connecting the severe accident diesel generator to ensure power supply to essential AM equipment. What is finally important is whether the diesel is connected in time or not. The time available for the decision and interaction was determined by accident analysis. Then the question was whether that time window would be sufficient for intervention or not.

Expert judgment was applied to answer this question and the underlying sub-questions discussed above.

The answers were provided by experts of the Paks NPP making use of the experience gained during severe accident management exercises (drills).

Human intervention is needed to open the valves of water drainage from the sump into the reactor cavity. This intervention is included in the SAMG, so its success is dependent on the use and quality features of the SAMG, as well as on some other performance conditions. While SAMG quality impacts mostly on the decision phase, other conditions may influence the decision as well as the execution phases of the intervention. For example, the consequences of a strong seismic motion or time pressure can impact on the likelihood of the intervention in both phases. The human error probability was determined on the basis of the method worked out for human reliability analysis in the L1 PSA. The key element of the approach is a decision tree that yield human error probabilities through evaluating the effects of performance conditions found important in the human reliability analysis. The following performance conditions (factors) were taken into account in the decision tree:

- the available and necessary time for decision and interaction,

- the environmental circumstances (effects of the initial event) of the intervention,

- types of equipment to be operated and method of operation,

- task complexity (difficulties of the emergency situation and the assessed level of cognitive challenges),

- presumable stress level as a manifestation of stressors,

- availability and quality of signals that can be used for diagnosis,

- actuators used to carry out the task,

- in case of out of control room interventions, the environmental conditions (e.g. dose rate ) associated with the intervention, as well as conditions for local action and means of communication with control room/technical support center staff,

- types and level of training for the expected task (including local interventions in particular), e.g. basic and training, refresher training classroom and/or simulator training, etc.,

- information on any operating experience for the intervention,

- quality of the guide, clarity, wording, decision support information.

## 4.3.5 SSTC, UKRAINE

### 4.3.5.1 Status

As it was already mentioned in chapter 2.7.2 the development of SAMGs for Ukrainian NPPs was based on the results of evaluation conducted in the framework of L2 PSA. More detailed analysis was further performed directly in the framework of SAMGs development and analytical justification activities. Therefore currently some differences in operator actions accounted in L2 PSA and those prescribed by SAMGs exist.

### 4.3.5.2 L2 PSA modelling

The main actions currently accounted in L2 PSA are RCS depressurization using PRZ PORVs or emergency gas evacuation system lines, establishing injection to the reactor coolant system. If RCS depressurization is performed successfully, such severe accident phenomena as direct containment heating and high pressure melt ejection are excluded from L2 PSA event trees.

Operator actions related to containment sump refilling in order to enable long-term melt cooling at the ex-vessel phase are also modeled in L2 PSA. But still L2 PSA does not contain other actions prescribed by SAMGs or associated with planned NPP modernizations (e.g., containment venting). To account for the latest SAMGs revisions the update of L2 PSA is to be performed during periodic safety re-assessment.

## 4.3.6 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| Human actions | |
|---|---|
| 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.<br><br>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:<br><br>• the dependencies between human errors before and after severe accident entry,<br><br>• the impact of context factor on human errors. This can be developed for the extended PSA approach (internal and external hazards …). |
| 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). |

# 4.4 FEEDING STEAM GENERATORS WITH WATER

The main purposes of filling the SGs are 1) protection of the SG tubes from creep rupture and 2) ensuring heat removal from primary circuit. Feeding SG may also 3) reduce releases of radionuclides into environment through damaged SG.

Tubes of steam generators represent not only boundary of primary circuit, but also boundary of containment, which means that steam generator tube rupture (SGTR) may lead to direct release of fission product into environment. This is why maintaining the SG barrier intact is one of the most important SAMG strategies. This strategy is initiated if low level in a SG is identified – SG tubes are uncovered or are about to uncovered.

Feeding SGs is one of generic Westinghouse SAM strategies and for some NPPs (PWR, e.g. Krsko or Temelin), it has the highest priority (SAG-1). The reason for this is that creep rupture may occur quite shortly after entrance into SAMG, so the water level in the SG should be assessed as soon as possible.

Beside the above mentioned positive effects of filling the SG, there may occur also some negative aspects caused by injection of cold water into hot dry SG or by depressurization of SG (feeding SGs from low pressure sources):

- thermal shock may cause a loss of integrity of SG body (not tubes),
- steam generated in SG may transfer fission products through open safety valve into the atmosphere,
- creep rupture of SG tubes in case of high temperature of SG tubes and significant pressure difference between primary and secondary circuit.

Control room operators in cooperation with technical support centre have to consider all positive and negative effects (before starting feeding SGs) and choose the most appropriate source of water.

If feeding of SG is not available, operators are supposed to return into Diagnostic Flowchart (DFC) or original guide and try to mitigate other threats caused by the severe accident.

Note: Isolation of broken SG is usually part of strategies aimed on limitation of radioactive release.

## 4.4.1 AREVA, GERMANY

### 4.4.1.1 Status

There are many reasons for filling the SG with water. Some of the main reasons are:

- Even in case of core heatup, secondary side feed and bleed may be successful as long as the SG tubes are not yet blocked with hydrogen.
- Feeding of a broken steam generator in case of SGTR will provide a filtering effect on the fission product release due to the pool scrubbing effect.
- Feeding of a non-broken SG is also important in high pressure scenarios, as it removes the potential for induced SGTR.

For all these reasons, German SAMG recommend feeding dry SG when possible. However, when it becomes clear that SG feed is not available for some SG, they shall be isolated, to avoid a direct bypass in case of induced SGTR.

All these reasons are relevant for L2 PSA and should be modeled if relevant guidance is available for the analyzed plant.

### 4.4.1.2 L2 PSA modelling

In the Event Tree model, two alternative function events are modeled. In the case of no SGTR, there is a risk of induced SGTR if one SGTR is not fed, therefore the feeding of all SG shall be queried.

In case of SGTR, the feeding of the broken SG or, if not possible, the isolation of the broken SG shall be modeled. Note however that L2 PSA isolation may not be exactly the same as L1 PSA isolation.

For both cases, the necessary manual actions should be modeled consistent with the actions which have already failed in L1 PSA to get into the severe accident, if any. In German procedures, the corresponding manual actions are described in EOP and are recommended with high priority in the SAM guidelines (with reference to the relevant EOP sections).

Due to the large relevance on the source term, especially for SGTR as initiating event, this action is explicitly modeled in the Level 2 PSA. Concerning human reliability, as in most cases one or more human actions have failed in the Level 1 PSA, the human action is not modeled explicitly, but instead it is considered as a subsequent action with moderate dependency to the Level 1 PSA actions (in particular primary side bleed and feed).

## 4.4.2 JSI, SLOVENIA

For Slovenian reactor, PWR type, injecting into the SGs is severe accident strategy number one. The purposes of injecting into the SGs are to protect the SG tubes from creep rupture, to scrub fission products that enter the SGs via tube leakage and to provide a heat sink for the RCS [33]. Besides positive there may be also negative impact by accelerating the release of fission products to the atmosphere. There are fix and mobile pumps available to feed SGs. Some of the pumps may be prevented from injecting into the SGs due to high SG pressure. If this is the case, then SGs have to be depressurized by SG power operated relief valves (PORVs) or using steam dump system to allow low pressure feed injection. SG PORVs can be operated also by local manual action.

If there is no electric power, decay heat removal can be achieved with auxiliary feedwater turbine driven pump (AFW TDP) and steam relief into the atmosphere through SG PORVs. For the first 4 hours (or more), there is compressed nitrogen in bottles to operate valves (SG PORVs and control valves for AFW TDP). During that period alternative source of compressed air can be established or it can be manually controlled the speed of AFW TDP and manually released the steam from SGs to control the decay heat removal. SG PORVs can be also opened locally using compressed air from portable diesel compressor and local pressure regulators or manually. As an alternative to the SG PORVs the main steam safety valves can be used for depressurization of SGs. Even if AFW TDP is not operable, plant has available portable firewater pumps onsite, with variety of injection flowpaths to SGs and variety of water sources. Fire truck can also be used.

The modelling of SG injection is not explicit in L2 PSA but implicit. Namely, SGs are modeled in containment event tree (CET). Various failure modes are then taken into account (SG tube rupture, overfilling of SG, isolation failure).

## *4.4.3 TRACTEBEL, BELGIUM*

### 4.4.3.1 Status

In WOG SAMG, the injection into the SG aimed at protecting SG from tube rupture, scrubbing FP that enter SG via tube leakage and providing a heat sink for the RCS.

### 4.4.3.2 L2 PSA modelling

In the APET, the injection into SG, both in the Very Early phase (thus before the SAMG entry, see VEAM_SGDEP in Figure 1) and in the Early phase (see EAM_SGDEP in Figure 2), is introduced taking into account the availability of systems (#SG for status of the SGs, and #PPORV for status of the PPORVs), the initiating-event (#INIT) and the human reliability analysis related to human actions as VEAM/EAM_SGDEP (i.e. Very Early / Early Accident Management action of SG DEPressurisation with injection) and VEAM/EAM_PPORV(i.e. Very Early / Early Accident Management action of PPORV opening).

The injection into the SG has an impact on the evaluation of RCS pressure during the Early phase (PRCSREP in Figure 1) and before vessel failure (EPRCS in Figure 2).

The possibility to have a RCS repressurisation (EREP in Figure 1) due to the loss of secondary heat sink due to hydrogen accumulation in the primary side of SG U-tubes is also considered.

| #INIT | #SG | VEAM_SGDEP | #PPORV | VEAM_PPORV | #PRCS | EREP | PRCSREP | # |
|---|---|---|---|---|---|---|---|---|
| | | | | | HIGH | NO | HIGH | **3a.01** |
| ATWS | | | | | MEDIUM | NO | MEDIUM | **3a.02** |
| | | | | | LOW | NO | LOW | **3a.03** |
| | | | OPPORV | | | NO | LOW | **3a.04** |
| | | | | | HIGH | NO | HIGH | **3a.05** |
| | | | FPPORV | | MEDIUM | NO | MEDIUM | **3a.06** |
| category 1 | | | | | LOW | NO | LOW | **3a.07** |
| | | | | YES | | NO | LOW | **3a.08** |
| | | | APPORV | | HIGH | NO | HIGH | **3a.09** |
| | | | | NO | MEDIUM | NO | MEDIUM | **3a.10** |
| | | | | | LOW | NO | LOW | **3a.11** |

**Figure 1: Part of the subtree CP-S03a - In-vessel RCS pressure evolution (repressurisation in early phase)**

| #INIT | EREP | #SG | #PPORV | PRCSREP | EAC | VEAM_SGDEP | EAM_SGDEP | VEAM_PPORV | EAM_PPORV | SGTR_I | HLSL | ESORV | EPRCS | # |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LLOCA | | | | | | | | | | NONE | NO | NO | LOW | 3b.001 |
| MLOCA | | | | | | | | | | NONE | NO | NO | LOW | 3b.002 |
| | | | OPPORV | | | | | | | NONE | NO | NO | LOW | 3b.003 |
| | | | | | | | | | | YES | NO | | LOW | 3b.004 |
| | | | | HIGH | | | | | NONE | | | YES | MEDIUM | 3b.005 |
| | | | | | | | | | | | NO | | LOW | 3b.006 |
| | | | | | | | | | | | | NO | HIGH | 3b.007 |
| | | | FPPORV | | | | | | | YES | NO | | LOW | 3b.008 |
| | | | | MEDIUM | | | | | NONE | | | YES | MEDIUM | 3b.009 |
| | | | | | | | | | | | NO | | LOW | 3b.010 |
| | | | | | | | | | | | | NO | MEDIUM | 3b.011 |
| | | | | LOW | | | | | NONE | NO | NO | | LOW | 3b.012 |
| | YES | | | | | | | YES | NONE | NO | NO | | MEDIUM | 3b.013 |
| | | | | | | | | | | | | | LOW | 3b.014 |
| | | | | HIGH | | | | YES | NONE | NO | NO | | MEDIUM | 3b.015 |
| | | | | | | | | | | | | | LOW | 3b.016 |
| | | | | | | NO | | | | YES | NO | | LOW | 3b.017 |
| | | | | | | | | NO | NONE | | | YES | MEDIUM | 3b.018 |
| | | | | | | | | | | | NO | | LOW | 3b.019 |
| | | | | | | | | | | | | NO | HIGH | 3b.020 |

**Figure 2: Part of the subtree CP-S03b - In-vessel RCS pressure evolution (depressurisation in early phase)**

In case of SGTR, the injection into an affected SG (EAM_FWASG in Figure 3, a similar action in the Late phase exists also) and the isolation of an affected steam generator (EAM_SG_ISO in Figure 3, a similar action in the Late phase exists also) has an impact on source term evaluation.

As shown in Figure 3, during the early phase and the vessel failure phase, FP in the gaseous phase inside the SG can: go back into the RCS (E/VF__SG_RCS_R), be released to the environment atmosphere (E/VF__SG_ATM_R), be released to the containment (E/VF__SG_CT_R), or be retained in the SG due to scrubbing (E/VF__SG_RET). If it is possible to isolate the affected SG and the isolation is successful (i.e. EAM_SG_ISO being YES), it is assumed that the FP in the gaseous phase will be retrieved back from the SG to the RCS. In case it is not possible to isolate the affected SG or in case the isolation has failed (i.e. EAM_SG_ISO being not considered or NO), the possibility of injecting into the affected SG will be evaluated. If it is possible to inject water into the affected SG and the injection is successful (i.e. EAM_FWASG being YES), scrubbing of FP inside the SG is expected, thus a better FP retention.

| #INIT | #SG | EAM_SG _ISO | E/VF_SG _ISO | EAC | EAM _FWASG | E/VF _FWASG | E/VF_SG _RCS_R | E/VF_SG _ATM_R | E/VF_SG _CT_R | E/VF_SG _RET | # |
|---|---|---|---|---|---|---|---|---|---|---|---|
| CAT1 | SG | YES | YES | | | NO | 100 | 0 | 0 | 0 | fp.10a.01 |
| | | | | | YES | YES | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.02 |
| | | NO | NO | | NO | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.03 |
| | FSG | YES | YES | | | NO | 100 | 0 | 0 | 0 | fp.10a.04 |
| | | NO | NO | | | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.05 |
| | NAC | YES | YES | | | NO | 100 | 0 | 0 | 0 | fp.10a.06 |
| | | | | YES | YES | YES | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.07 |
| | | NO | NO | | NO | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.08 |
| | | | | NO | | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.09 |
| CAT2 | SG | | NO | | YES | YES | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.10 |
| | | | | | NO | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.11 |
| | FSG | | NO | | | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.12 |
| | NAC | | NO | YES | YES | YES | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.13 |
| | | | | | NO | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.14 |
| | | | | NO | | NO | 0 | §E/VF_SG_ATM_R | 0 | §E/VF_SG_RET | fp.10a.15 |
| CAT3 | SG | | NO | | YES | YES | 0 | 0 | §E/VF_SG_CT_R | §E/VF_SG_RET | fp.10a.16 |
| | | | | | NO | NO | 0 | 0 | §E/VF_SG_CT_R | §E/VF_SG_RET | fp.10a.17 |
| | FSG | | NO | | | NO | 0 | 0 | §E/VF_SG_CT_R | §E/VF_SG_RET | fp.10a.18 |
| | NAC | | NO | YES | YES | YES | 0 | 0 | §E/VF_SG_CT_R | §E/VF_SG_RET | fp.10a.19 |
| | | | | | NO | NO | 0 | 0 | §E/VF_SG_CT_R | §E/VF_SG_RET | fp.10a.20 |
| | | | | NO | | NO | 0 | 0 | §E/VF_SG_CT_R | §E/VF_SG_RET | fp.10a.21 |

**Figure 3: Subtree FP-S10a – Distribution of FP released from SG during early phase and at vessel failure**

Note that the modeling of the SAM related to feeding steam generator with water in the Belgian L2 PSA has been given in this paragraph as an example. The modeling of other SAM actions is similar to this, and will not be detailed further in the document for confidentiality reasons.

## 4.4.4 SSTC, UKRAINE

### 4.4.4.1 Status

SG feeding strategy effectiveness was evaluated under SAMGs analytical justification activities. The results of analyses confirmed the initial assumption that implementation of SG feeding strategy at late core damage stage has very little effect (if any) to the melting progression. Nevertheless the strategy allows achieving several goals, namely:

- reduce the potential of SG tubes creep rupture caused by high temperatures at the primary side;
- providing a filtering effect on the fission product release in the case of SG tube integrity loss.

Considering the goals listed, SG feeding strategy is currently incorporated to SAMGs as the 3rd priority strategy (after RCS depressurization and RCS water injection strategies).

### 4.4.4.2 L2 PSA modelling

Current L2 PSA does not specifically addresses the operator actions on implementing the strategy. However the effect of SG secondary side water availability is implicitly accounted by eliminating SG tubes creep rupture failures

in SA sequences which are not associated with substantial SG depletion or/and which progress at low primary pressure.

## 4.4.5 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| 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. |

# 4.5 CORIUM COOLING / WATER INJECTION STRATEGY (IN-VESSEL COOLING, EXTERNAL FLOODING OF RPV, CORIUM STABILIZATION IN THE CONTAINMENT …)

The water injection strategy during the core degradation aims at cooling the corium. A key issue of L2 PSA is to determine the success probability of the measure, depending mainly on the status of the core when injection begins and on the injection rate. But the measure should not be considered in L2 PSA only for the positive impact (cooling the corium in order to prevent the vessel rupture, stabilizing the molten corium in containment, wash-out the containment atmosphere…). All potentially negative impacts should also be considered and modeled, e.g.: hydrogen production kinetics increase, containment atmosphere de-inerting, RCS pressure increase, containment pressure increase, ex-vessel steam explosion risk, increase of contaminated liquid quantity, possible corrosion due to using mineralized water, criticality issues when injecting water on degraded core geometry.

The availability of water systems inside the containment should also be taken into account in their capability to mitigate the source term. In that strategy, Fukushima has highlighted the possibility of managing the accident with portable equipment. Although such equipment is provided to avoid core damage, it could also be used in severe accidents. The effectiveness of such equipment in mitigating the source term has to take into account that all water injected into the containment has to be confined. This chapter addresses this issue.

## 4.5.1 EDF&IRSN, FRANCE

### 4.5.1.1 In-Vessel water injection strategy

#### 4.5.1.1.1 Status

EPR (PWR): Currently in-vessel water injection is not allowed as soon as core melt has begun (on SAMG entry criteria, see §4.2.1.2.1). This is because the core catcher system would lose efficiency if the vessel happened to fail while water injection is active.

French Fleet (PWR): The means of in-vessel injections considered are safety injection pumps (high or medium pressure, low pressure), accumulator tanks and a list of additional fix or mobile pumps (borated water only).

EDF SAMGs define some rules for in-vessel water injection:

- for EPR, water injection is prohibited during in-vessel accident progression;
- for the French Fleet, water injection is allowed with conditions:
  - the flowrate must be sufficient to avoid fast hydrogen production at the beginning of the core degradation (no restrictive conditions, for hydrogen control, regarding in-vessel water injection for 900 PWRs - recent modification done by EDF);
  - the primary pressure must be controlled to avoid high pressure vessel failure.

### 4.5.1.1.2 EDF L2 PSA modelling

EPR (PWR): In the L2 PSA, only detrimental effect for unexpected water injection (human failure + availability of water injection) is able to be modeled. This probability is assumed to be very low and is currently not modeled.

French Fleet (PWR): If safety injection becomes available due to the accidental scenario itself (for example scenario with initial high pressure sequences where a depressurization occurs in due time rendering the low pressure systems functional), in vessel corium retention and severe accident phenomena are calculated with MAAP. Additionally, human actions (e.g. repair of components or use of any other available system to inject water) may be modeled in the L2 PSA if they lead to significant detrimental effect on hydrogen risk and DCH, but currently no credit for in vessel corium retention is made for these human actions.

### 4.5.1.1.3 IRSN L2 PSA modelling (900 and 1300 MWe PWRs)

The L2 PSA modelling of water injection effects during in-vessel accident progression takes into account:

- the corium cooling and stabilization;
- the hydrogen production and its combustion;
- the vessel over-pressurization and possibility of vessel circumferential failure, DCH, … ;
- the containment over-pressurization.

This has been done on the base of ASTEC calculations and with specific modelling which complements the results obtained with ASTEC for each PDS. For instance: concerning the risk for the containment by hydrogen combustion, a dynamic PSA application ([40]) was developed; this methodology can provide an estimation of the containment failure probability vs. the water injection and the spray system activation time (these probabilities were then introduced in the APET).

Uncertainties have been taken into consideration as far as possible (the quality of the modelling in the IRSN L2 PSA suffers from the status of knowledge and modelling of corium reflooding).

Comment: Significant modifications of water management strategies in case of severe accident are being analyzed in France. Information obtained from L2 PSA will be useful.

## 4.5.1.2 External flooding of RPV

### 4.5.1.2.1 Status

For 900, 1300 and 1450 MWe PWRs, the vessel cavity can be flooded in case of spray system activation. Nevertheless, no voluntary cavity flooding and external cooling of RPV is considered for 900, 1300 and 1450 MWe PWRs.

For EPR, the vessel cavity remains dry and external flooding of RPV is not part of the severe accident strategies.

### 4.5.1.2.2 IRSN L2 PSA modelling (900 and 1300 MWe PWRs)

The analysis of the vessel behavior takes into account some heat exchanges with the cavity. But these heat exchanges are limited by the vessel insulator.

In-vessel retention is credited in the L2 PSA APET in case of in-vessel water injection (situations with a low primary circuit pressure, a low residual power and a sufficient in-vessel water injection).

Ex-vessel steam explosion has been modelled in L2 PSA for situations with a flooded cavity.

## 4.5.1.3 Ex-vessel water injection strategy

### 4.5.1.3.1 Status

EPR (PWR): Passive core melt cooling by IRWST (In containment Refueling Water System Tank) is achieved once the corium spread into the core catcher. The long term severe accident management is part of the design.

For the French Fleet (PWR): There are not yet strategies for the long term accident management until plant stabilization but EDF is designing NPP modifications for ex-vessel corium stabilization. The principles of these modifications are now being discussed in France and are the following:

- increase the area available for corium spreading;
- take advantage of the mechanisms observed in corium concrete experiments (melt ejection, water ingression, …) if it is proven that they help cooling the corium by water;
- optimize corium submersion by water with new provisions allowing water arrival in the reactor cavity zone after corium spreading.

The L2 PSA results will be used in complement with deterministic analysis to check that, for Gen II NPPs (after upgrade), the conditional probability of basemat penetration is very low for all scenarios of core damage accident.

The existing L2 PSAs do not yet consider these future NPPs upgrades.

### 4.5.1.3.2 EDF L2 PSA modelling

EPR (PWR): A probability of failure of the core catcher system is taken into account in the L2 PSA, leading to basemat failure.

French Fleet (PWR): If the spray system is available, a conditional probability (depending on the accident scenario) of core melt stabilization in the vessel pit is taken into account. If not, basemat failure is supposed certain.

### 4.5.1.3.3 IRSN L2 PSA modelling (900 and 1300 MWe PWR)

The IRSN L2 PSAs include detailed analysis of the corium concrete interaction with and without late flooding. All issues are considered: corium cooling, gas production, containment pressurization, impact of late spray system activation.

Even with limitations on SAMG for long term accident management, the IRSN L2 PSAs consider for some scenario that there is no basemat penetration by the corium 15 days after the initiating event. This result was obtained without using the latest knowledge on corium concrete interaction and effects of water (see OECD MCCI SOAR report 2016, to be published soon). This was one reason to recommend (in 2009), for the French Gen II PWRs, implementing solutions for the ex-vessel corium stabilization based on existing knowledge on corium concrete interaction. As explained above, this is today one objective associated to the French Gen II LTO program.

## 4.5.2 GRS, GERMANY

### 4.5.2.1 In-Vessel water injection strategy

If water is or becomes available for in-vessel injection in any phase of the accident, it is considered for German NPPs that it should be used for this purpose without reservations. Concerns about potential drawbacks are addressed as follows:

- It is possible that there is a certain increase of the hydrogen production rate due to flooding a molten core. However, these concerns are derived from small scale experiments with few fuel pins and may not be transferred directly to real core conditions with a variety of fuel pin conditions. In fact, deterministic analyses showed only little (if any) additional hydrogen generation. Since all German NPPs are equipped with passive autocatalytic recombiners, the hydrogen will mostly be recombined.

- It is possible that flooding is too late or not sufficient in order to cool the core, and the injected water may not have the desired effect. However, since in German reactors there is no alternative use for the water (there is no containment spray), the water is not wasted.

- In BWRs, there are some RPV bottom injection possibilities, e.g. at the control rods. The flow rate of these injections can be managed to a certain extent. If maximum flow rate is available, core melt may be avoided, even if all other injection fails.

As a rough estimate for the efficiency of injection, reference [8] can be applied. It is based on an evaluation of many experiments. For application in PSA, the German PSA guidelines [7] suggest a correlation between the probability for successful injection and the core status when injection begins.

As a typical result for a modern German PWR, about 30% of all core damage sequences end up in an in-vessel core retention (TMI-type scenario) [9]. The majority of such cases are initial high pressure sequences where a depressurization occurs in due time rendering the low pressure systems functional.

What is not considered in present German PSAs are human actions (e.g. repair of components) which would lead to injection into a previously damaged core, except those that are enabled by already available low pressure systems after a late depressurization of the primary system. Since the time window between begin of core melt and the point of "no-return" (even with water injection) for the core degradation is in the order of 1 h or even less, this omission may not be very significant.

### 4.5.2.2 External flooding of the RPV

In German PWRs, no strategy exists for ex-vessel cooling. In BWRs, the containment can be flooded to a certain extent. However, due to the specific and complicated design at the BWR RPV bottom, successful ex-vessel cooling does not seem to be possible without significant refitting measures.

## 4.5.3 IEC, SPAIN (BWR)

In SAMG water injection to RPV is required including external sources and portable equipment. Currently, only external sources are considered in the L2 PSA.

Containment flooding is required when RPV injection is not possible or is not enough and also in case of LOCA or when RPV has failed.

Analyses have been done to prioritize these strategies from the perspective of source term releases, using the MAAP5 code to determine the source term released to the environment and the RASCAL4.3 code to determine the off-site radiological consequences.

In SAMG, with the containment flooding the RPV could also be flooded in case of LOCA or in case of failure and could prevent MCCI. Containment flooding in L2 PSA is used to reduce MCCI risk and therefore late containment failure.

Vessel corium retention is considered in SAMG and also in L2 PSA, based in reference documents and supported by specific analyses with severe accident codes. These analyses support vessel corium retention during the first phase of corium relocation, but for the L2 PSA, only 50 % of the probability is considered successful, except when a local action is required, where the vessel corium retention is considered failed (no credit that the action could be done in a sufficient short period of time to prevent the vessel failure).

The external cooling of RPV is not considered in SAMG or in L2 PSA by design limitation.

Risks of water injection considered in L2 PSA are hydrogen combustions and DCH.

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.

Water injection modelling in L2 PSA could be optimized for shutdown modes because the cooling probability of the debris inside the vessel and outside the vessel is higher than for full power, the time availability for the actions is probably higher gand systems with lower flow rates could be used. In any case, additional detailed analyses are needed to support specific criteria for shutdown L2 PSA. SAMG for shutdown modes are currently being developed.

## *4.5.4 INRNE, BULGARIA*

### 4.5.4.1 In-Vessel water injection strategy

#### 4.5.4.1.1 Status

For the Kozloduy NPP, units 5 and 6, in-vessel water injection strategy consists of the two main parts:

- primary circuit depressurization;
- primary circuit water injection.

The primary circuit water injection [22] is the most important and effective strategy in an attempt to stop SA progress. The successful application of the strategy at an early (before significant core damages) phase of accident progress guarantees suppression of hydrogen generation, prevention of a reactor vessel failure and elimination of the effects of the melt and concrete reaction. Submerging the melt in the reactor vessel is the first and the most important step on the way to reach a stable controlled state of the unit fuel as a whole.

As a mean to supply coolant to the primary circuit, available trains of the safety systems (SS) can be used – Emergency core cooling system - high pressure (ECCS-HP), low pressure (ECCS-LP), make-up and blowdown system trains (system TK). If during in-vessel phase of severe accident the operators manage to deliver sufficient coolant to the core, there is a chance to cool down fragments and melting product from the reactor core and to localize this in the reactor vessel. These actions are covered by the SAMG [30].

The aim of primary circuit depressurization is to prevent emergency scenarios that occur under conditions of primary circuit high pressure:

- early reactor vessel failure;
- direct containment heating;
- steam generator (SG) tube creep rupture.

The application of this strategy would reduce the negative effects of submerging the melted fuel that has the potential to result in a drastic reactor pressure increase. Reactor depressurization enables the use of low head pumps for submerging the reactor fuel. The strategy will be applied through the following systems:

- pressuriser safety valves (Pz SV);
- pilot operated relief valve (PORV);
- the primary circuit gas removal system.

#### 4.5.4.1.2 L2 PSA modelling

The L2 PSA modeling takes into account the capability of classified equipment for severe accident management as well as phenomena studied in SAM such as hydrogen burning (for in vessel and ex vessel phases), high temperature creep rupture, HPME and DCH, in-vessel steam explosion, and also availabilities of the systems and their effect on the in-vessel accident progression (LPIS, HPIS, Spray System, make-up system) [41], [42]). Human actions are also considered.

### 4.5.4.2 In-Vessel melt retention by external cooling of reactor vessel

The strategy initially was assumed only for VVER-440 as successful and now is under developing for VVER-1000. The VVER-440 was shut down and they are in the process of decommissioning. So, it is in interest only VVER-1000.

The possibility for external- reactor vessel cooling was analyzed. It was defined that in some options of severe accident progress damage of the vessel cannot be avoided [30], [31]. It is assumed now that it was due to lack of information. It is considered further investigation of this strategy as promising. The recent study shows that it could be successful. Because that an EU project is under conduction to avoid uncertainty in performing this strategy.

### 4.5.4.3 Ex-vessel management of corium in cavity

The strategy for management of cavity integrity requested some additional measures to be done. It is already plugged surrounding tubes, which are used for neutron controlling. They have been considered as first challenges during ex-vessel management, when melt will bypass containment in first 30 minutes. The next challenge is axial ablation. To avoid melt through concrete basement it is assumed to increase basement area by opening the door to increase heat exchange area of corium. In this way will be increase twice the area. It is also assumed additional options for further area increasing. The next measure that is studied is covering the concrete by ceramics tile which should resist to temperature above 3000 °C. It is also assumed water injection in cavity to cool down melt, which request additional study. These measures are under development.

## 4.5.5 SSTC, UKRAINE

### 4.5.5.1 In-vessel water injection strategy

#### 4.5.5.1.1 Status

This strategy refers to a set of actions aiming in termination of the core degradation and preserving the reactor vessel integrity. Successful implementation of the strategy allows localizing the corium inside the reactor vessel and prevent occurrence (or mitigate the development) of the following phenomena which can result in a loss of containment integrity:

- reduce the hydrogen generation in case of level restoration in the core,
- terminate reactor core degradation,
- prevent molten corium discharge into the containment compartments,
- prevent detonation conditions occurrence in the containment,
- contribute to fission products precipitation in the primary circuit.

Importance of this strategy for reduction of radioactive releases frequency was indicated in L2 PSA and further confirmed in SAMGs analytical justification (AJ).

Analyses of in-vessel injection strategy included scenarios with early and late restoration of RCS water supply to identify available timeframes and injection flow rates needed. Since overall mass of hydrogen generated in the case of water injection (with a rate that is sufficient to terminate core degradation and to restore water level in

the reactor core) is lower than in the case without water injection, it is concluded that in-vessel injection does not increase threat of containment failure caused by global hydrogen burn. However, increase of steam generation rate need to be taken into account since it affects containment pressure.

The primary means of in-vessel injection foreseen by the original VVER design are high pressure safety injection (SI) and make-up pump. Decrease of RCS pressure (which is the one of high priority actions) is needed to allow for high pressure SI and can be performed with pressurizer PORV or emergency gas removal system (the later requires power supply from house loads busbars or other connection to be established). Deep RCS depressurization that can be reached via PRZ PORV allows to establish water supply from low pressure SI. Necessity to provide additional RCS water injection means for BDBA/SA conditions is being evaluated under CSIP.

Timing of SI restoration is quite important for reaching the goals of this strategy. Thus for VVER-1000 the analyses demonstrate that SA progression is terminated if RCS water injection is started before core melt damages the core lower support plate. Otherwise RPV failure could not be avoided regardless of available SI means used to provide water supply.

After RPV failure injection to RCS is used to establish water supply to the cavity in order to provide ex-vessel corium cooling.

### 4.5.5.1.2 L2 PSA modelling

In-vessel water injection in current L2 PSA for VVER-1000 is accounted in top questions of containment event tree (or decomposing event tree) and correspondent FT. The questions considered include availability of emergency core cooling systems at the onset of core damage and potential failures of these systems before RPV failure.

According to deterministic analyses results the success criteria for prevention of RPV failure is injection to the primary circuit by three HPIS trains or by one LPIS train. Also for high pressure PDS the operator actions on RCS depressurization via PRZ PORVs or emergency gas evacuation system are considered. More details on RCS depressurization modeling are provided in ch.4.6.5.

### 4.5.5.2 External flooding of RPV

External RPV flooding is aimed in corium retention inside the reactor vessel by preventing RPV melt-through thus precluding SA progression to the ex-vessel phase. Currently the strategy is not implemented in SAMGs and is not considered in L2 PSA. Modernizations related to this strategy were successfully implemented for several European units with VVER-440 reactors (e.g., at Paks NPP, Hungary). At the same time feasibility of the strategy for VVER-1000 reactors is still under evaluation. Considering existing European experience for VVER-440 reactors this measure is included to CSIP for Rivne NPP units 1 and 2.

It shall be noted that external RPV flooding is the only means to prevent VVER-440 containment failure if in-vessel water injection was not implemented successfully. This is caused by the fact that the reactor cavity door is a part of containment boundary that makes it impossible to arrange ex-vessel melt spreading and cooling without loss of containment integrity.

### 4.5.5.3 Ex-vessel water injection

### 4.5.5.3.1 Status

Ex-vessel water injection strategy is applicable to the late phase of SA. In the process of SA progression the strategy could be applied after reactor vessel failure and the subsequent recovery of the same systems which are used for in-vessel water injection strategy. During the operation of RCS injection systems the coolant flows through melted-through opening of the reactor vessel and enters the reactor cavity. As a result, the corium which to some extent has already been spread over the reactor cavity and (depending on the cavity door state) adjacent compartment can be flooded.

The success of the strategy depends on the thickness of corium layer (in the case of debris has spread over the large area) or on the debris porosity if compact debris bed has formed. The analyses demonstrate that debris flooding is successful if the debris layer thickness does not exceed 10-15 cm. The later depends on the melt temperature, melt composition and melt ejection (pour) rate. All these factors have significant uncertainties and require further in-depth studies.

For VVER-1000 if spreading area is bounded by the reactor cavity compartment, two concurring processes are expected. The first one is associated with the radial ablation and containment failure through ionizing chambers channels, and the second one is the cavity door melt-through. Since L2 PSA estimates and more recent analyses indicate relatively high door melting time (approx. 4-5 hours), L2 PSA containment event trees consider the cases with cavity door failure caused by high pressure in the reactor cavity or door closed case for RPV failure at low pressure.

There is a potential to increase probability of the strategy success by allowing for debris to spread over the large area. Therefore correspondent measures are being evaluated in the framework of CSIP and In-depth Evaluation of SA Phenomena Program.

### 4.5.5.3.2 L2 PSA modelling

Ex-vessel water injection in current L2 PSA for VVER-1000 is accounted in top questions of containment event tree (or decomposing event tree) and correspondent FT. The questions considered include availability of emergency core cooling systems (HPIS/LPIS) at the RPV failure time, restoration of these systems at the ex-vessel SA phase, and their potential failures of these systems. For ECCS restoration questions the operator actions on recovery of 2/3 LPIS trains or 1/3 LPIS plus 3/3 HPIS trains are considered.

Restoration of ECCS depends on the failure complexity and timing of failed component repairing. The analysis results demonstrate that the majority of ECCS components can be restored during the accident and restoration time varies from few minutes to several hours. For human reliability analysis the maximal HPIS and LPIS components restoration times were selected. Available time was determined based on thermal-hydraulic calculations in support of L2 PSA. In the result, the following performance shaping factors (PSF) were chosen:

- "Available time" PSF was selected as "large";
- "Emergency situation's effect" PSF was taken as "heavy";

- "Decision making" PSF is "extremely heavy";
- "Man-machine interface" was selected as "adequate" since an action is performed at the location of failed component with subsequent testing locally and from MCR;
- "Instructions quality" is "weak" since during L2 PSA development SAMGs were not completed, and other plant instructions that describe these actions for SA are not available.

## 4.5.6 JSI, SLOVENIA

### 4.5.6.1 In-Vessel water injection strategy

#### 4.5.6.1.1 Status

If establishing of a secondary heat sink was not accomplished, operators would establish feed and bleed flow to the reactor core. To get to severe accident conditions, the abilities of high head safety injection, low head safety injection and charging pumps was either lost or severely degraded. Without water present in the core, the core will continue to heat up and will begin to relocate due to melting. The only way to prevent core from relocating to RPV lower head is to restore injection into RCS. There are fixed and alternate means for RCS injection. Alternate means include also portable equipment.

In addition to benefits there are drawbacks with injecting water into RCS, which have potential to negatively impact the accident progression by accelerating the fission products release. The negative impacts for all means of RCS injection may be creep rupture of SG tubes, containment pressure increase, containment flooding and containment overpressure severe challenge. In such a case limitation on RCS injection is made. For example, in case of containment overpressure severe challenge the RCS injection is terminated.

#### 4.5.6.1.2 L2 PSA modelling

For L2 PSA analysis both positive and negative impacts are accounted for in supporting deterministic analyses (using MAAP computer code). PSA models consider the success criteria and failure probabilities as recognized in supporting analyses.

Regarding optimizing the water management the results of PSA are used for determining the most probable sequences and these sequences are modelled in detail with deterministic analyses to provide sufficient information about sufficient water inventory requirements. HRA is part of L2 PSA mostly as support to systems (setting recirculation, containment isolation) while for recovery actions is not used. The reliability of systems is modelled. The existing mobile equipment is not modelled in L2 PSA. Finally, the safety upgrade program is ongoing and those systems are not yet part of L2 PSA.

### 4.5.6.2 Ex-vessel water injection strategy

#### 4.5.6.2.1 Status

The containment flooding to establish cooling of the core material is used as long term strategy for Slovenian PWR when other strategies have been ineffective. Namely, a controlled stable plant state may include flooding the

containment to submerge the core material remaining in the reactor vessel. The preferred mean is containment spray. Other means are refuelling water storage tank gravity drain, portable severe accident management equipment (SAME) pumps and fire trucks. There are several benefits [33]. First, water in the containment sump can be used for emergency core cooling systems (ECCS) injection or containment spray if it subsequently becomes available. Second, water on the containment floor can quench the core debris following vessel failure and prevent molten core concrete interaction and basement melt-through. Third, fission products released from core debris on the containment floor would be scrubbed. Injecting into containment without establishing long-term heat removal would not prevent containment failure, but according to the analysis would significantly delay containment failure for more than a day. Negative impacts for all means of flooding are loss of equipment and instrumentation and containment overpressure severe challenge. The loss of instrumentation and equipment can be drastic side effect for flooding the containment; therefore it is considered as a last resort.

### 4.5.6.2.2 L2 PSA modelling

For L2 PSA analysis same approach is used as for in-vessel water injection strategy (see section 4.5.6.1).

## 4.5.7 TRACTEBEL, BELGIUM

In WOG SAMG, the injection into the RCS aimed at protecting SG from tube rupture, scrubbing FP that enter SG via tube leakage and providing a heat sink for the RCS.

### 4.5.7.1 In-vessel water injection strategies

In SAMG, one should always inject water into RCS if equipment and water resources are available (those two aspects being verified in the evaluation of the APET).

The injection into RCS depends on RCS pressure (for the availability of equipment) and has an impact on RCS pressure (before vessel failure). It has also an impact on core damage extent, hydrogen production, risk of in-vessel steam explosion and risk of vessel failure.

After vessel failure, injection into RCS has an impact on corium cooling in cavity (via failed vessel) and on hydrogen production.

In recirculation mode, the possibility to have containment sumps plugging impeding recirculation is quantified.

In-vessel water injection has also an impact on FP releases into containment (scrubbing of FP).

### 4.5.7.2 External flooding of RPV

External flooding of RPV is considered in generic WOG SAMG. Currently, it is not possible to have an efficient external cooling of RPV for Belgian NPPs. It is however introduced and assessed in L2 PSA. The injection into containment and cavity (via an existing connexion) has an impact on the assessment of ex-vessel cooling in early phase.

### 4.5.7.3 Ex-vessel water injection strategies

The injection into containment and cavity (via an existing connexion) has an impact on corium cooling in cavity in late phase (depending on the presence of water before or after vessel failure). The risk of ex-vessel steam explosion has to been quantified in case water is present in cavity before vessel failure.

In the L2 PSA of one certain unit, in which the ex-vessel core debris cooling strategy consists of injecting water before vessel failure to promote core debris quenching, it is revealed by the L2 PSA results that the ex-vessel steam explosion risk due to the presence of water is non-negligible. Note that the quantification of the ex-vessel steam explosion risk is mainly based on MC3D[8] calculations and structural analyses. Nevertheless, by further result analysis, it is demonstrated that the global containment failure probability with the presence of water before vessel failure remains significantly lower than the one without, confirming the benefits of the ex-vessel core debris cooling strategy in this unit.

Finally, ex-vessel water injection has also an impact on FP releases from corium into containment atmosphere (scrubbing of FP).

## 4.5.8 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| Corium cooling / water injection strategy | |
|---|---|
| In-vessel water injection | A precise link must be done between state of the unit/ knowledge and L2 PSA assumptions; a dialog between L2 PSA teams and researchers is needed, in order to know how L2 PSA sequences can be modelled with physical codes (modelling of corium reflooding ...). |
| | 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 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. 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. |

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[8] MC3D: A multidimensional Eulerian code which is developed by IRSN and which is mainly devoted to Fuel Coolant Interactions (FCI).

| 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 bypass. 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 occurrence probability of energetic phenomena (hydrogen and carbon monoxide explosion, steam explosion, HPME and DCH) able to threat the confinement of radioactivity, <br> • Reduce as far as possible the risks of containment bypass, <br> • 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 conservativisms 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. |

# 4.6 RCS DEPRESSURIZATION

In all L2 PSA, high pressure core melt sequences are considered a high risk because high pressure core melt ejection from a failed RPV bottom will create very severe containment loads. But in PWRs it is also possible that the hot gases generated by the molten core that flow through the steam generator tubes can cause thermal loads that endanger tube integrity. This may cause a potential leakage path from the primary to secondary circuit and a containment bypass with very high radionuclide releases, and it is an additional reason for RCS depressurization.

It is usually foreseen by SAMG to depressurize the primary circuit when core melt is imminent. NPPs are equipped with different kinds of safety and relief valves, which may require AC power, DC power or compressed air to be opened. It should be ensured that it is possible to open the required number of valves even if normal power supplies are lost. Furthermore, safety valves should be qualified for severe accident conditions (e.g. pressure, temperature, radiation) and seismic hazards. Human actions are generally needed for RCS depressurization. A L2 PSA should take into account all equipment and human NPP features.

## 4.6.1 EDF&IRSN, FRANCE

### 4.6.1.1 Status

The RCS safety valves have a key role in case of severe accident to limit the in-vessel pressure and avoid DCH or induced steam generator tube rupture. Opening the pressurizer safety valves is one of the first immediate actions required by the French SAMG (if the valves are not already opened manually by applying EOPs). Conservatively, no credit is given to the possibly stuck valves in the open position.

For the 900, 1300 and 1450 MWe PWRs, to avoid any unwanted closure of these valves (due for example to electrical cables failure after irradiation) during the in-vessel progression of accident, EDF is modifying the electrical command of the valves. Importance of this upgrade was enhanced after the Fukushima Dai-ichi accident. For SBO situations, local actions allow valves opening.

For EPR, dedicated valves have been designed for severe accident.

### 4.6.1.2 EDF L2 PSA modelling

For all reactors, RCS depressurization is required as soon as core melt has begun. Then L2 PSA takes into account only detrimental effects if relief valves remain closed, for reactor states where RCS is pressurized. Failure on opening relief valves is due to:

- Human failure (failure when locally opening relief valves in degraded conditions in case of electric losses);
- System failure: probability of stuck relief valve unable to open or relief valve closure if the open position is not guaranteed (in case of electrical losses or if not qualified for severe accident irradiation rate).

Considering RCS depressurization failure, induced SGTR, DCH and lift up of RPV are taken into account regarding detrimental effects.

### 4.6.1.3 IRSN L2 PSA modelling (for 900 and 1300 MWe PWRs)

The L2 PSA considers both system and human failure for the safety valves opening during in-vessel accident progression.

Dependencies (human and material) with L1 PSA are kept with the PDS attributes. For situations where the primary circuit pressure is controlled by the safety valves (very small LOCA or no LOCA), detailed analyses were also performed to take into account the fact that the containment pressure increases the necessary force for opening the safety valves.

All consequences in case of a depressurization failure are considered (DCH, I-SGTR, vessel displacement …).

For L1 PSA situations where the safety valves are stuck in the open position, the corresponding PDSs are modelled with the ASTEC code as LOCA scenarios in vapor phase. For these situations, there is no risk of high in-vessel pressure.

## 4.6.2 GRS, GERMANY

For all reactors, RCS depressurization is required as a preventive measure in order to prevent core melt. For BWRs, this is an automatic action, for PWRs it is manual. Since it occurs before core melt, it is a L1 PSA issue. Only if depressurization does not succeed before core damage, it becomes a L2 PSA issue.

The time window between begin of core melt and RPV bottom failure in high pressure cases is in the order of 1 h. Therefore, the success probability for depressurization (which failed in this case before core melt per definition) can be assumed to be limited for most cases. However, for fast scenarios the limited diagnosis time to perform primary depressurization may have a significant contribution to the failure probability of the manual action. If the L1 PSA goal to avoid core damage is relaxed in L2 PSA, and instead the goal "avoid RPV failure" or "avoid high pressure RPV failure" is used, there may be a significant improvement in the success probability of the manual action.

More important than depressurization by SAM seem to be accident related phenomena which lead to depressurization: failure of a hot leg / surge line, or failure of a safety valve in stuck open position. Therefore, present analysis and research efforts at GRS are directed at such issues.

## 4.6.3 IEC, SPAIN (BWR)

### 4.6.3.1 Status

RPV depressurization is required by RPV water level control for injection with low pressure systems in EPG and from the beginning in case of SAMG entry.

### 4.6.3.2 L2 PSA modelling

L2 PSA considers failure of depressurization considering the human failure and the system failure. Safety and Relief Valves (SRV) could also be opened with portable equipment with its corresponding procedure as included in SAMG, but is not yet taken into account in L2 PSA.

L2 PSA takes into account negative impact of depressurization failure, as hydrogen combustions and HPME and DCH; and positive effects as well, as lower probability of fuel coolant interaction inside and outside RPV and higher coolability of the dispersed debris.

Also, L2 PSA can be used to analyze the late SRV failure (closed) in a SBO situation by depletion of batteries. The RPV re-pressurization causes the loss of the core coolability supported with low pressure injection systems.

## 4.6.4 INRNE, BULGARIA

### 4.6.4.1 Status

For the Kozloduy NPP, units 5 and 6, for the reactor coolant system depressurization, multiple options are available to ensure that high pressure core melt scenarios are prevented.

The main technical provisions for primary circuit depressurization and preventing the evolution of severe accident at high pressure are the pressure relief valves of the pressurizer and the primary circuit gas mixture emergency

removal system. An additional option for primary depressurization is the drain valves of the RCP sealing water system.

In order to ensure a practical possibility for using the primary circuit gas mixture emergency removal system under conditions of severe accident evolution, a modification of the system valves electrical power supply was performed, ensuring redundancy of electrical power supply of the respective valves from batteries [31].

The Kozloduy NPP VVER 1000 reactors current design has have technical provisions for reactor coolant system depressurization to avoid high pressure melt ejection, which are available in SBO conditions. The required operator's actions are described in the emergency instructions.

### 4.6.4.2 L2 PSA modelling

L2 PSA analysis [41] takes into account detrimental effects such as high temperature creep rupture, HPME and DCH, hydrogen generation. The L2 PSA considers both equipment and human failure during in-vessel and ex vessel accident progression.

## 4.6.5 SSTC, UKRAINE

### 4.6.5.1 Status

RCS depressurization strategy is of a highest priority in Ukrainian NPP SAMGs and is required to achieve two main goals, namely to allow for implementation of RCS injection strategy and to preclude RPV failure at high RCS pressure. Two primary means for implementing the strategy are foreseen:

- pressurizer pilot operating relief valves (PRZ PORVs), and
- emergency gas removal system (EGR).

Both means require operator actions for RCS depressurization and are accounted in L2 PSA considering potential hardware failure and operators reliability. Usage of PRZ PORVs is more preferable since it allows to decrease pressure down to LPIS shut-off head thus providing more alternatives for implementing the water injection strategy. Unlike EGR valves all three PRZ PORVs are powered from essential power supply busbars.

Since none of the valves are qualified for SA conditions their functioning and operability needs further confirmation. SA equipment qualification measure under CSIP is intended to evaluate this issue for the main SA equipment and supporting systems/elements needed.

### 4.6.5.2 L2 PSA modelling

In modelling of RCS depressurization strategy in L2 PSA it was considered that it could:

- prevent RPV failure by implementation of RCS injection;
- eliminate the threat associated with direct containment heating and high pressure melt ejection phenomena;
- increase zirconium oxidation if water injection rate is not sufficient to terminate core degradation.

Modelling of RCS depressurization actions is performed by incorporation to the fault trees of several basic events representing successful opening of PRZ PORV and of EGR valves. HEP is estimated using decision tree developed for Level 1 PSA purposes. The following performance shaping factors (PSF) were accounted:

- "Available time" PSF was selected based on deterministic analyses results;

- "Emergency situation's effect" PSF has been taken as "heavy";

- "Decision making" PSF is "extremely heavy";

- "Man-machine interface" was selected as "good" since an action is performed at MCR and operator is able to monitor the response without any delay;

- "Instructions quality" is "weak" since during L2 PSA development SAMGs were not completed, and other plant instructions that describe these actions for SA are not available.

FT containing Basic Events which model Human Errors at RCS depressurization are connected with headers of decomposing ET (auxiliary ET for to simplify a containment ET logic). Sample fault tree is shown below.
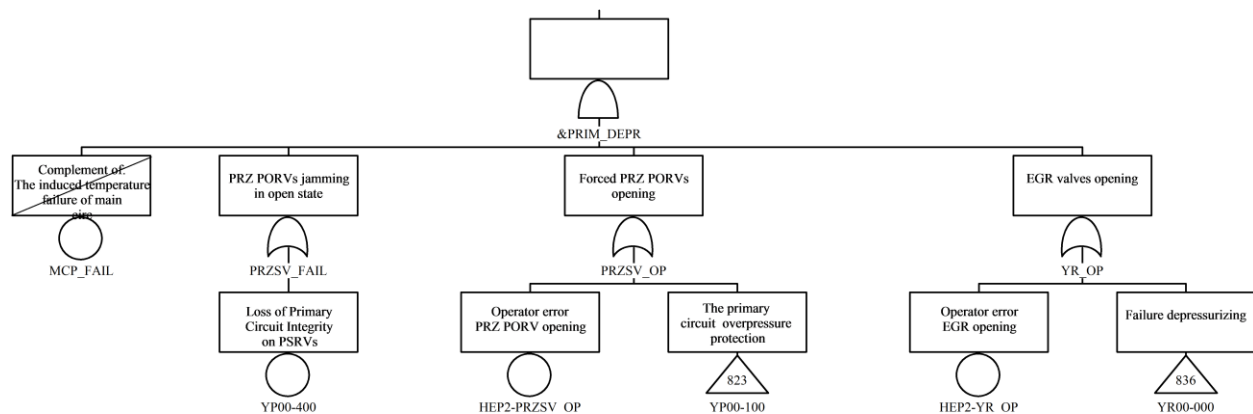


**Figure 4: Modelling of RCS depressurization**

## 4.6.6 JSI, SLOVENIA

### 4.6.6.1 Status

For Slovenian reactor, PWR type, RCS depressurization is prevention strategy, performed before core melt and instructed by emergency operating procedures (EOPs). However, lowering the RCS pressure is also one of the top priorities of SAM. Since operators are instructed to depressurize in EOP as response to inadequate core cooling, either an operator error or equipment failure must have occurred if the RCS is still pressurized, or there is loss of all AC power. The main positive impacts of depressurization during SA are decreased potential for high pressure melt ejection (as consequence of RPV failure due to high RCS pressure), decreased potential for creep rupture of steam generator tubes and lower pressure will allow more injection sources to inject into RCS. The preferred mean is dumping steam from the steam generators, followed by pressurizer power operated relief valves (PRZ PORVs) and other RCS depressurization paths. The Slovenian plant is provided with two onsite portable air compressors which could restore instrument air to PORVs, and portable generators for providing necessary power to motor operated valves (opening letdown path or reactor vessel head valves for depressurization).

Finally, additional pressurizer PORVs is planned to be installed, which will be qualified for design extension conditions (DEC) events [32].

### 4.6.6.2 L2 PSA modelling

Regarding the valves opening and valves failures all equipment addressed in the deterministic analyses and needed to mitigate the accident is modelled in L2 PSA. For example: all containment isolation valves, injection and recirculation lines...

Regarding optimizing the RCS depressurization function it should be noted that RCS depressurization is mostly modelled in L1 PSA. For sequences, which model core damage at high RCS pressure, L2 RCS depressurization is modelled. The results of RCS depressurization optimization showed that additional PRZ PORVs are needed and will be installed within plant safety upgrade program as stated above.

Open position of valves is guaranteed in case of electrical losses, severe accident conditions and external hazards. Namely, all valves important to safety are designed with fail safe function. Additionally Slovenian PWR uses alternative power/air supplies for valves manipulation.

## 4.6.7 TRACTEBEL, BELGIUM

RCS depressurisation is considered in the evaluation of RCS pressure before vessel failure which has an impact on vessel failure mode and related events (such as High Pressure Melt Ejection, Direct Containment Heating, Rocket mode failure).

It has to be added that in addition to the action to depressurise RCS (by PPORVs or SG), other events are considered to evaluate RCS pressure before vessel failure: induced SGTR, hot leg or surge line failure and the possibility to have a stuck-open PPORV as a result of cycling.

RCS depressurisation with PPORVs, hot leg or surge line failure and a stuck-open PPORV as a result of cycling have an impact on containment conditions (pressure, activity).

## 4.6.8 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| 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. |
| 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 7during 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). |
| 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.<br><br>L2 PSA can be used to analyze the possibility of late SRV closure, for example :<br><br>• closed in a SBO situation by depletion of batteries,<br>• closed manually by the operators (error or simply because the situation seems to improve – e.g. after RCS flooding)<br><br>Several issues are of interest for L2 PSA :<br><br>• the RPV re-pressurization can causes the loss of the core coolability supported with low pressure injection systems,<br>• 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,<br>• 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,<br>• the coupling between containment heat removal, RCS pressure and water injection possibility can be of crucial importance :<br>    o for example, this has conducted to the loss of steam driven water injection pumps for the Fukushima unit 2 and 3,<br>    o the containment pressure increase has a direct impact on the RCS pressure and may make some low head pumps unavailable.<br><br>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. |

# 4.7 CONTROL OF FLAMMABLE GASES

A core melt accident may lead to an intense production of non-condensable and flammable gases (hydrogen, carbon monoxide) in the containment during in-vessel and ex-vessel accident progression. A major problem is the existence of fuel cladding made of Zr, which is being considered as part of the first layer of defence-in-depth and as part of "conservative design". However, under SA conditions it becomes one of the highest risk sources. It is worth mentioning that some plant features which may be beneficial in normal operation and design basis accidents can turn out to be negative under SA conditions. Several SAM strategies have been proposed and implemented in order to cope with that challenge. Within L2 PSA the risk assessment emanating from combustible gases is a well-established topic. Within extended PSA considerations in ASAMPSA_E focus will be on assessing countermeasures and SAM against this challenge.

## 4.7.1 GRS, GERMANY

Passive autocatalytic recombiners (PARs) are installed in all German NPPs. In deterministic accident analysis (mostly done with MELCOR code), PARs and their action are modelled as realistic as possible. It turns out that under specific circumstances (e.g. low steam content, little atmospheric convection) the PARs are not able to absolutely prevent hydrogen combustions. However, for PWR, the residual hydrogen combustions are by far not

threatening the containment. PARs do not only recombine Hydrogen, but also Carbon Monoxide, which is generated by core concrete interactions.

From a probabilistic point of view, the following considerations are due:

- PARs are purely passive systems, therefore no action whatsoever is required to start them, and there are no related issues of reliability.

- PARs are designed for severe accident conditions (steam, temperature, radiation …). However, they would probably be damaged if they are directly hit by water impact from a large loss of coolant. But this would affect only very few of all PARs and not significantly influence the overall performance. Consequently, in a L2 PSA the PARs are assumed to be acting as designed.

- PARs get hot when operating. They emit hot gas and occasionally also sparks. This could be seen as an ignition source. Therefore, if an ignitable atmosphere exists, the PARs will not only be recombining, but they also act as igniters. This is, in principle, a safety enhancing feature, like igniters. However, since PARs have no design requirement to be igniters, there is an uncertainty about this property, which should be taken into account in L2 PSA.

- Depending on the accident progression (in particular on the large hydrogen generation by core-concrete interaction), PARs can use up all available oxygen inside the containment, and residual hydrogen accumulates inside the containment. This is not a threat for the containment (no combustion is possible without oxygen), but any leak or release (venting) of the containment would contain hydrogen. In a L2 PSA performed by GRS [9], this has led to a significant probability for hydrogen burns in the venting system and associated damage to the venting filter.

- The issue of hydrogen containing leaks from the containment is even more significant for BWRs, see the Fukushima experience.

## 4.7.2 EDF&IRSN, FRANCE

### 4.7.2.1 Status

SAM equipment and strategies for hydrogen control are the following:

- PARs;
- instrumentation for detection of hydrogen in the containment (thermocouples in PARs);
- restrictive conditions (for hydrogen control) for in-vessel water injection to avoid hydrogen production in containment with high kinetics for 1300 and 1450 PWRs (no water injection with small flowrate at the beginning of core degradation), and for EPR (no water injection during in-vessel core degradation);
- no restrictive conditions (for hydrogen control) regarding in-vessel water injection for 900 PWRs (recent modification done by EDF);
- restrictive conditions for spray system activation (if the spray system is not in operation after core melt, operators must avoid its activation during 6 hours to avoid containment atmosphere de-inerting). For reactor shutdown states with open primary circuit, no restrictions are considered in SAMG.

Note: Other restrictions for in-vessel water injection exist to avoid vessel pressurization (if break size is below 7.62 cm (3 inch)).

### 4.7.2.2 EDF L2 PSA modelling

For all reactors control of flammable gases is achieved by passive autocatalytic Hydrogen and carbon monoxide recombination. L2 PSA studies take into account the flammable gases concentration versus time thanks to recombination laws in MAAP code. For the French Fleet (PWR) containment spray is not currently allowed during 6 hours after core melt if spray was not active before core melt, because of hydrogen volume fraction increase by condensation of vapor (de-inerting). Then an inappropriate human action is modeled into the L2 PSA. This could be updated if the benefit of pressure reduction by spray is proved to be sufficient to avoid containment failure by gas combustion (ongoing studies).

### 4.7.2.3 IRSN L2 PSA modelling (for 900 and 1300 MWe PWRs)

A large number of accident scenarios are calculated with ASTEC, allowing hydrogen concentration calculation in the reactor containment or adjacent buildings.

All information obtained by these ASTEC calculations have been used to implement simplified physical modelling in the L2 PSA APETs for:

- o    the containment atmosphere composition evolution (air, H2, steam, …),
- o    the impact of in-vessel and ex-vessel water injection on containment atmosphere composition,
- o    the impact of spray system activation on containment atmosphere composition during in-vessel and ex-vessel phases.

The APET includes also

- a dynamic modelling of human actions which allows taking into account:
  - o     the coupling between the human actions and physical/material status of the reactor,
  - o    the correct and not correct applications of SAMG and their impact on the reactor.
- the equipment failure modelling after hydrogen combustion (arbitrary values for the failure rate – this will be improved later),
- a modelling of the H2 transfer in the annulus reactor building (for 1300 PWRs with a double containment),
- flammability and AICC[9] pressure calculations during in-vessel and ex-vessel phases and comparison of pressure peak with containment fragility curves.

## 4.7.3 NUBIKI, HUNGARY

It is important to control the concentration of flammable gases in the containment because the burn or detonation of the gas mixture can cause the failure of the containment and leads to an early large radioactive material release into the environment from the VVER-440/213 containment. This containment is a special one with small volume, relatively low failure pressure and with an airlock where the blowdown pushes the air and with bubble

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[9] Adiabatic isochoric complete combustion

condenser trays. Due to the special design, the steam can inert the containment for certain time. The containment atmosphere will be flammable after condensation of the steam. The flammability depends on the flammable gases, the oxygen and inert gases concentrations. The flammable gases can be hydrogen ($H_2$) and carbon monoxide (CO). Inert gases can be steam ($H_2O$) or carbon dioxide ($CO_2$). Severe accident calculations were made and it was found that the hydrogen is a real challenge for the VVER-440/213 containment and hydrogen management is necessary.

The possible main actions to control the flammable gases in the containment atmosphere are:

- to prevent or to decrease the flammable gas source from metal-steam reaction or from molten core-concrete interaction,
- to avoid that the gas mixture in the containment atmosphere will be flammable,
- to decrease the flammable gas concentration to avoid containment overpressurization due to burn or detonation.

The prevention or termination of core melt influences the first strategy. The assessment of this strategy needs the determination of probability of the successful termination of core melt. In case of hydrogen production, the calculation of hydrogen concentration is necessary in a deterministic way. If the hydrogen concentration is higher than 4%, it means there will be flammable gas concentration in the containment. The minimum approach is the determination of AICC (Adiabatic Isotropic Complete Combustion) pressure versus time and to compare it with the containment fragility curve. In this conservative way, the probability of the containment failure was calculated. If the conservative method gave too high frequency of early containment failure, a more accurate method is used: calculation of the probability of hydrogen ignition, and from this and from the calculated best estimate hydrogen burn pressure, the probability of pressure load was calculated. The convolution integral of the probability of containment load and containment fragility was prepared. It supplies the probability of the containment failure and from it, the frequency of the containment failure due to flammable gas burning can be calculated.

If the hydrogen concentration is higher than 11 vol% and the gas mixture is flammable, the Deflagration to Detonation Transition shall be taken into account. This modifies the load curve. In case the hydrogen concentration was higher than 15% we assumed detonation which causes containment damage.

The second strategy needs to inject inert gases into the containment, so that the containment atmosphere does not contain much oxygen fractions. The feasibility of the strategy depends on the type of the containment. Our small containment can be filled with nitrogen or CO2 at the end of the maintenance. If there is no oxygen or the oxygen concentration is less than 5 vol%, the gases in the containment will not be flammable. Due to the containment leakage rate this was unrealistic. The other possibility is to fill the containment with steam at the beginning of the accident, before the flammable gas production starts. The steam can be produced by the evaporation of the water in the core or from the steam generator, but it can be condensed. A suitable steam concentration could not be ensured during the accident, as has been checked by deterministic calculations.

The third strategy is to decrease the flammable gas concentration. The hydrogen management can be performed by:

- different type of hydrogen igniter (spark, catalytic) and/or
- hydrogen recombiners.

For these devices, the hydrogen ignition and burn cannot be avoided. The probability of containment failure due to induced hydrogen burn was calculated with similar methods as described previously. The ignition probability is determined in a different way. The intentional ignition probability (in case of hydrogen igniter) depends on the local gas compositions at the place of igniters, the igniter type, the failure probability of igniter's power supply and the human behavior (SAMG, measuring system reliability and human error). The ignition due to recombiners depends on the recombiner type, the time and rate of recombination and the gas composition at the place of recombiner. We examined the ignition concept, the recombiners concept and the combined method.

According to calculations and engineering judgment, the recombiners concept was selected. The determination of the number of recombiners was based on deterministic and probabilistic calculations. The uncertainty calculations showed that the capacity of 30 large severe accident hydrogen recombiners is sufficient to save the containment about 95% probability at 90 % confidence level.

Traditional PSA tools are suitable to examine the effectiveness of a standalone hydrogen management system.

However the strategies are interrelated, the first strategy is when the core/fuel is cooled in a successful manner. The delayed or not sufficient water injection into the overheated core can increase the hydrogen production. Similar connection exists with the pressure decrease by spray system and the recombiners effectiveness. The spray system condenses the steam in the containment atmosphere, therefore increases the flammable gas concentrations. The effect of spray system and water injection on the overheated fuel depends on the timing of action and the flammable gas control strategies. The SAMG handle this, because it describes when the spray system can be used. To avoid the increased hydrogen production, the SAMG determines the optimization of water injection into primary system.

The best way of handling these complex and time dependent processes may be the dynamic PSA. However the dynamic L2 PSA is too complex and complicated for this purpose. Therefore we partly solved this problem by a traditional event tree method. The water injection into the primary system was asked two times. First question asks if the water injection occurred before a large part of the core heats up and partly melted, the second one asks when the core melts and the reflooding causes increased hydrogen production. The time for the questions was calculated for each branch of each plant damage state. It means that the question in the containment event tree is time dependent. Similar method is used for the spray system. This method is conservative, but until now we could not use a better one.

Different severe accident management strategies can be examined as separate systems including the assessing the adequacy of the hardware, information, guidelines and human probability. Finally the severe accident strategies should be checked as a whole SAM system.

Depending on the nuclear power plant, the hydrogen burning and the effect of pressure and temperature load in the reactor building may also be necessary to be examined during the open containment shutdown state. The hydrogen source from the spent fuel pool and burn in the reactor hall or containment should also be examined taking into account SAM.

## 4.7.4 IEC, SPAIN (BWR)

### 4.7.4.1 Status

Hydrogen risks are currently managed with igniters (active system), but passive autocatalytic recombiners (PAR) are going to be installed shortly. Both systems have been designed (number, type and location) to manage the hydrogen release by limiting severe accident sequences. Also, carbon monoxide can be removed with these systems.

SAMG also consider containment spraying and containment venting as a support for reducing containment pressure to prevent reaching the Hydrogen Deflagration Overpressure Limit.

### 4.7.4.2 L2 PSA modelling

L2 PSA takes into account probability of igniters' availability considering instrumentation failure, human failure and system failure. One of the two divisions of igniters can be supplied by DC power. Portable equipment with its procedure can be used for igniters supply but it is still not considered in L2 PSA.

When the ignitors are available, the combustion risk on the containment integrity is considered negligible. Only, for scenarios with this system failed, the combustion risk is analyzed. The combustion risk is analyzed with severe accident codes on limiting cases (nor auto-ignition nor preliminary burns) and using the adiabatic isochoric complete combustion (AICC) to compare with containment failure pressure. The analysis is limited to the detonation-deflagration transition limit (12%). L2 PSA model gives credit to preliminary burns by electrical equipment located into the containment for non-SBO sequences and a containment failure probability value by an effective combustion is assigned. With hydrogen concentration value higher than 12% a direct containment failure probability is assigned. Uncertainty analyses are implemented on these values to fix the importance of the phenomenon.

L2 PSA could support in long term SBO sequences the optimization of management of power supply for active systems.

L2 PSA could support the hydrogen risk regarding hydrogen generation analyzing the water injection in different accident phases.

L2 PSA could also support the hydrogen risk management improving the modelling of containment spraying to take into account the benefit of reducing containment pressure to prevent reaching the Hydrogen Deflagration Overpressure Limit.

L2 PSA could be used to determine the impact of using passive or active systems to manage the hydrogen risk.

## 4.7.5 INRNE, BULGARIA

### 4.7.5.1 Status

For the Kozloduy NPP, within the Modernization Program, 8 passive autocatalytic recombiners (PARs) for each containment of units 5 and 6 have been installed, for hydrogen risk management in case of Beyond Design Bases Accidents. An additional analysis was made, which shows that their capacity is sufficient also for controlling the hydrogen from the in-vessel phase of a severe accident [31].

In order to cover the whole severe accident evolution, including an ex-vessel phase, additional 15 PARs for each containment of the units 5 and 6 of Kozloduy NPP have been installed.

The SAMG is covering the conditions when the containment environment is reaching inert conditions. It is possible in this situation, that the concentration of hydrogen becomes more than 10%, which will create a risk of flammability (or detonation) in case of establishing a leak to the environment. For long term management of hydrogen behavior, further investigation is considered.

### 4.7.5.2 L2 PSA modelling

For the Kozloduy NPP, according to the L2 PSA study [41], the hydrogen burning is included explicitly in the CET. The impact of this phenomenon is mainly based on the MELCOR analyses and contemporary understanding of phenomena behavior.

## 4.7.6 SSTC, UKRAINE

### 4.7.6.1 Status

The main hydrogen mitigation strategy chosen for Ukrainian NPPs envisages application of passive autocatalytic recombiners (PARs) eliminating the necessity of support systems or operator actions. To this moment all units are equipped with PARs to recombine hydrogen generated during design basis accidents. Installation of PARs to control hydrogen concentration during severe accidents is performed under CSIP. The set of recombiners for SA is already installed at South-Ukraine NPP (SUNPP) units 1 and 2.

Productivity and location of PARs is selected so as to prevent global hydrogen deflagration and flame acceleration conditions either at in-vessel or ex-vessel phase. Correspondent justification is provided (to be provided) as a part of PARs installation documentation. Analyses demonstrate that PARs intensify convection inside containment thus improving hydrogen mixing and homogenizing steam-gas mixture inside containment. Relatively high hydrogen concentrations exceeding the deflagration limit can be reached in some of the compartments (e.g., PRZ relief tank compartment). Nevertheless, since local deflagration does not represent a threat to the overall containment integrity it is not planned to install igniters in addition to PARs.

SFPs located inside containment can contribute significantly to the overall hydrogen generation, which is driven by larger Zr inventory comparing to the one in the reactor core. However hydrogen generation in SFP starts significantly later than in the reactor core and at that time oxygen is already consumed partially by recombination of hydrogen released from the reactor. For example, even in the case of emergency unloading (i.e., all fuel assemblies are located in SFP) severe fuel damage occurs not earlier than 6-10 hours after termination of SFP cooling.

Carbon monoxide contribution to the flammable gases concentration is considered of low significance for Ukrainian NPPs since the concrete used for the containments are of low carbonate content.

Even though overall PARs productivity is estimated taking into account a potential mixture de-inertization caused by intentional or spurious containment spray actuation, SAMGs provide the necessary cautions and restrictions for operators on initiation of containment spray.

### 4.7.6.2 L2 PSA modelling

Since PARs were not installed at the moment of L2 PSA development, their operation is not considered in current version of L2 PSA. The risks associated with hydrogen burn and detonation is modeled explicitly in containment event trees taking into account specific conditions expected in the containment compartments.

## 4.7.7 TRACTEBEL, BELGIUM

Thanks to the implementation of the PARs in all Belgian NPPs, there is no specific action for hydrogen risk management in L2 PSA.

Nevertheless, a tool has been developed by Tractebel in order to assess, with a best-estimate approach, the risk of containment failure due to hydrogen before and after vessel failure.

Several steps are needed for the hydrogen risk assessment with this tool. Firstly, an expert judgment is performed to determine generic parameters related to ignition, propagation, flammability and combustion modes. Various severe accident scenarios calculated with the MELCOR code version 1.8.6 are then used to determine the atmosphere composition and thermal-hydraulics behavior of the containment. The containment structural integrity is described by probabilistic fragility curves. Finally, the developed tool allows combining the generic parameters with severe accident calculation results in order to obtain the global risk for the containment. It is also able to propagate uncertainties on the input data with random sampling.

By the application of this tool, it has been observed in the L2 PSA studies of certain units that, the activation of the containment sprays system after vessel failure may lead to non-negligible hydrogen risks due to the rapid steam condensation in the containment. Consequently, the necessity for the evaluation of the hydrogen impact when depressurizing the containment has been emphasized in the accident management.

## 4.7.8 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| 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, inerters, …) 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. |
| 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: <br>- simple approach may be practical for NPPs with significant safety margin against flammable gas combustion effects, <br>- more complex or precise approach is needed if the safety margins are low (typically for some Gen II reactors) (see below). |
| Modelling the time | For the risk quantification by L2 PSA, if the safety margins against effect of hydrogen |

| dependent phenomena in the event trees | combustion are limited, it may be needed to model some coupled phenomena with dynamics modeling :<br><br>• kinetics of hydrogen or carbon monoxide release in the containment,<br>• time of spray system activation and kinetics of steam condensation,<br>• time of in or ex-vessel water injection and effect on flammable gas release in the containment.<br><br>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 …).<br><br>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 :<br><br>• on hydrogen production during core degradation with or without reflooding,<br>• on PARs efficiency,<br>• on PARs ignition effect,<br>• on time of combustion (stochastics phenomena except if a controlled igniter is used). |
| 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:<br><br>• load (pressure and temperature peak) on the containment walls and impact on their integrity,<br>• local load (pressure and temperature peak) on some key equipment for the SAM,<br>• effect of hydrogen leakage into auxiliary buildings or secondary containment,<br>• effect of hydrogen for the containment venting system.<br><br>Comment : recombiners or partial combustion reduce the amplitude of the pressure peaks in the containment ; nevertheless, approximatively 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. |
| 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. |

# 4.8 CONTAINMENT FUNCTION (ISOLATION, VENTILATION/FILTRATION OF AUXILIARY BUILDINGS, MANAGEMENT OF LIQUID RELEASE)

The containment function can be compromised by accident related loads, by loss or failure of the containment isolation, or by containment bypass. Containment by-pass is a special type of accident scenario, in which a path to the environment is created while the containment structure is still intact.

## 4.8.1 EDF&IRSN, FRANCE

### 4.8.1.1 EDF L2 PSA modelling

All the containment penetrations that can be opened in the different reactor states are analyzed in the L2 PSA. In case of command failure or station blackout a human factor for local manual closure of penetrations is taken into account (if this action is properly required in the procedures). Considering the delay for equipment hatch closure, no credit is currently taken into account for manual closure in the L2 PSA, excepted for EPR which is fitted with a rapid closing device.

### 4.8.1.2 IRSN L2 PSA modelling (for 900 and 1300 MWe PWRs)

The dependencies involved in containment isolation are taken into account, e.g. power supply to motor operated valves, DC power and possible battery back-ups to actuators, automatic isolation signals, etc. The manual actions are also identified and quantified with HRA methods.

Specificity of shutdown reactor states is that the hatches and wide penetrations can be opened when an accident occurs. Reactor states with « open containment » are of crucial importance due to the potential consequences of an accident. For the IRSN L2 PSA, main hatches and wide penetrations taken into account in the model are the following:

- equipment access hatch;
- penetrations of containment sweeping ventilation system;
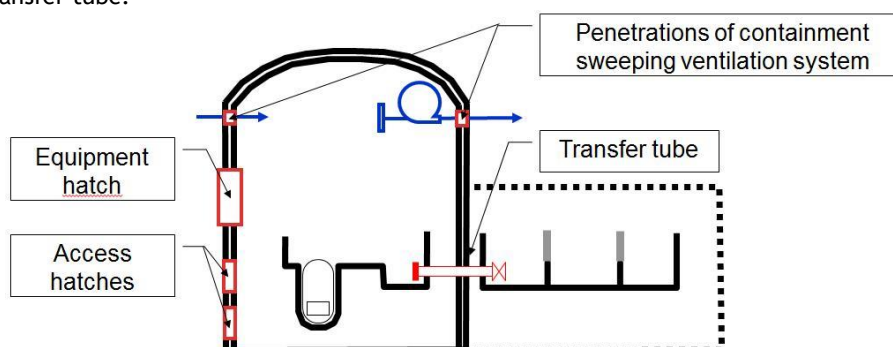- personnel access hatches;
- fuel transfer tube.



**Figure 5: Hatches and wide penetrations – French PWRs**

As an example, equipment hatch closure modelling is described hereafter.

The handling of the equipment hatch is not easy:

- the weight is about 30 tones;
- special hoisting winch or polar crane operations are needed;
- bolting operations require to work inside the containment;
- the crisis organization is needed to order the equipment hatch closure.

In addition, external electric sources are required for handling: closure is not possible if external electric sources are lost.

A global timeframe of 11 h is considered to close the equipment hatch (to include availability of crisis teams and operators on call, availability of polar crane, manual opening/closure time).

This action is not required by EOPs or SAMG but it could be requested by crisis teams: the HORAAM model (§4.3.1.3.2) is then applied to assess the failure probability (in case of SBO situations or time of SAMG entry less than 11 h, it is assumed a systematic failure).

Depending of the scenario, failure probabilities obtained with HORAAM model are the following:

| Kinetic | Failure Probability |
|---------|---------------------|
| Time of SAMG entry < 11 h | 1 |
| 11 h < Time of SAMG entry < 17 h | 1 if information and measurement means are "unsatisfactory" (SBO situations)<br><br>$10^{-1}$ if information and measurement means are "satisfactory" and the scenario "difficult"<br><br>$10^{-2}$ if information and measurement means are "satisfactory" and the scenario "easy" |
| Time of SAMG entry > 17 h | 1 if information and measurement means are "unsatisfactory" (SBO situations)<br><br>$10^{-2}$ if information and measurement means are "satisfactory" and the scenario "difficult"<br><br>$10^{-3}$ if information and measurement means are "satisfactory" and the scenario "easy" |

## 4.8.2 IEC, SPAIN (BWR)

The containment isolation is analyzed for all penetrations which failure may cause a non-negligible source term release. A study of penetrations behavior in severe accident conditions has been done and the results will be introduced or should feedback into PSA as well as other possible containment failures. Isolation is guaranteed in severe accident conditions.

L2 PSA considers the containment isolation failure only for SBO conservatively and therefore all the SBO sequences with late external injection systems availability are not considered because of the difficulty of the local actions.

The isolation failure area does not allow a late containment quasi-static pressurization but other dynamic processes are possible. The mass, energy and source term release though the area may cause a late failure of the systems located into the annexes buildings. Additionally, local actions could be limited especially at the severe accident phase. All these conditions are taking into account in the L2 PSA analyses.

As SAMG for shutdown states are still being developed, for the scenarios with RPV closed the current SAMG are acceptable to be used.

In L2 Shutdown PSA we have considered the containment and drywell opening and also the impact when the RPV is also opened.

L2 Shutdown PSA has been used for optimization of management of equipment hatch and personnel door (drywell or containment) considering the human reliability based on time availability and the complexity of the action and taken into account the environmental conditions. This analysis could support the development of the SAMG for shutdown states.

## 4.8.3 SSTC, UKRAINE

In PSA models the failure of containment isolation and containment failure during SA progression are considered separately. The containment isolation function is accounted for in interface model between L1 and L2 PSA. During SA progression the containment failure due to fast or slow pressurization is modeled in L2 PSA containment event trees. Sequences with containment bypass caused by primary to secondary breaks are explicitly modeled in L2 PSA as separate containment event trees and consider cases without operator actions (early phase ETs) and with RCS depressurization to minimize leakage rate (late phase ETs).

Since none of the containment isolation valves are qualified for SA conditions their functioning and operability needs further confirmation. SA equipment qualification measure under CSIP is intended to evaluate this issue for the main SA equipment and supporting systems/elements needed. Influence on containment penetrations is specifically requested to be addressed based on the results of state review of the Utility SA qualification plan.

At the station blackout (SBO) conditions the compressed air reservoirs which are part of compressed air system allow operation of the isolation valves. Later on the mobile diesel generators (MDGs) are used to provide power supply to air compressors (already available at SUNPP Units 1, 2. Other units will be supplied with MDGs as scheduled in CSIP and indicated in National Action Plan which is originated from post-Fukushima stress-test results. Recent activities on SAMGs development for shutdown states revealed vulnerability of containment isolation to SBO since the main containment transport gate is energized from house-loads power supply busbars. At some units there is a possibility to close the gate manually, however available timeframe is limited by radioactive releases and at the moment is deemed to be insufficient to consider correspondent actions as credible. Considering an importance of containment isolation function for shut-down states the detailed evaluation of this issue is started by the Utility to obtain more precise estimates of available timeframe and to propose necessary measures.

It shall be emphasized that even if the contribution of this vulnerability to L2 PSA release frequency estimates could be low (because of low initial plant damage state probability) its significance shall not be diminished due to potential consequences for the site radiological situation that could affect other units.

The following factors characterize the plant damage states which are important for a further severe accident progression point of view and are generally taken into account during development of containment event trees:

- electrical power supply availability at the moment of core damage onset;
- containment state;
- RCS pressure.

For example, two main containment states at the time of core damage which result in uncontrolled radioactivity release are:

- containment bypass (e.g., via ECCS heat exchanger leakages, instrumentation pipelines);
- failure of containment isolation.

These containment states are further considered in plant damage states grouping in the framework of Level 1 and Level 2 interface development, resulting in PDS groups with containment bypass and PDS with containment isolation failure. Due to impact of containment state on radioactivity release to the environment for these PDS separate containment event tree "Containment bypass" may be introduced. To distinguish PDS with successful containment isolation from the ones with non-isolated containment the correspondent top event is included into

this CET. Failure of containment isolation function is modeled in the fault tree that accounts reliability of containment isolation system components as well as human errors.

## 4.8.4 AREVA, GERMANY

In the L2 PSA, all containment penetrations which are potentially open are systematically identified. However, in SAMG, independent of this analysis, it is identified whether the containment is damaged or not isolated. In such a case, the annulus ventilation system is optimized in such a way to minimize fission product release into the environment. This strategy is geared towards directing the release of fission products through the available filters, even though these are not designed for a severe accident, rather than allowing a direct release of fission products. The actions include optimization of the annulus ventilation system as pressure control in the RB annulus and the auxiliary building, to avoid opening of a direct air path, and maximizing the heat removal from the containment, to reduce the flow of fission products out of the containment.

Despite the potential impact on the plant risk, as the filter efficiency cannot be proven for this use, and because it cannot be shown that in this case a large release can be avoided, this action is currently not modeled in L2 PSA.

The more likely case, however, is the use of the annulus ventilation system in case of containment leakage. Here, it has been shown in the German SAMG that the use of the annulus ventilation system in such a way that underpressure is maintained while at the same time minimizing the flow through the filters into the environment, leads to a significant lower source term. However, as the minimization of the source term for the intact containment case is not risk-relevant, also this action is currently not modeled in L2 PSA.

## 4.8.5  TRACTEBEL, BELGIUM

The failure of containment isolation has a direct impact on the containment status. In case of size equivalent to rupture, it will imply the end of the APET evaluation for containment performance. In case of size equivalent to leak, it can be isolated if possible and the releases during the failure timeframe are considered in source term evaluation.

Furthermore, it has been revealed by the L2 PSA studies of certain units that, the unavailability of the internal and extraction ventilations of the annular space[10] and of the extraction ventilation of the auxiliary building in some shutdown plant operating states has a significant negative impact on fission product retention. Consequently, it has been recommended to improve the guidelines to ensure the availability of the ventilation/filtration systems both in the annular space and in the auxiliary building in shutdown states.

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[10] Annular space: corresponds to a secondary containment in Belgium units. In the annular space, the extraction ventilation system assures the under-pressure condition inside, while the internal ventilation system filters fission products inside the annular space.

## 4.8.6 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| Containment function (isolation, ventilation/filtration of auxiliary buildings …) | |
|---|---|
| L2 PSA applications | L2 PSA can be used: <br><br> • 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), <br><br> • 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), <br><br> • 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, <br><br> • to examine consequences of any leakage (transfer of contaminated gas or liquid in auxiliary buildings, transfer of combustible gas …). |

# 4.9 STRATEGIES FOR CONTAINMENT PRESSURE CONTROL (CONTAINMENT VENTING, HEAT EXCHANGERS, CHRS …)

If actions aimed at cooling of the fuel debris, containment heat removal and control of combustible gases are not available or have not been successful, the containment pressure can increase to a level that threatens the integrity of the containment.

Containment venting or specific heat exchangers allow lowering the containment pressure, preventing damage to the containment structures and help controlling the leakage of radioactive products. A filtered venting system consists of vent pipes from the containment atmosphere (in BWRs possibly both from the drywell and/or from the wetwell gas space), filter unit(s), necessary valves and the piping from the filter(s) to the ventilation stack (or another exhaust location).

Heat exchangers allow containment steam pressure decrease and possibly avoid the need for containment venting and thus are also suitable to limit the radioactive release. Different technologies can be used (passive, active, air or water cooling …) but their robustness in severe accident conditions and to external hazards is a major issue. Some examples can be mentioned here :

- active recirculation circuits from the reactor containment sump (many Gen 2 PWRs, EPR, …) with heat exchangers (and possibly intermediate circuit before the heat sink to limit its contamination);
- passive circuit using water tank (for example HPR1000 [53]);
- containment steel liner spray from outside (AP1000 design, Loviisa reactor containment).

## 4.9.1 GRS, GERMANY

German PWRs have containment venting for containment pressure control which is equipped with filters for mitigating the releases to the environment. There are no other features which are implemented specifically in

order to control containment pressure, e.g. there is no containment spray. BWRs, of course, are equipped with a condensation pool which largely controls the BWR containment pressure.

The reliability of the venting system is assessed in L2 PSA, taking into account system failures (e.g. blocked valves), or human error when operating the system. As a consequence, there is a probability of a few percent that venting is not activated when required [9]. If venting fails, it is assumed that finally the containment will fail due to overpressure.

Several years after installing the venting system, additional deterministic analyses are being performed in order to check that the venting system has sufficient capacity to depressurize the containment in different accident conditions. This issue has been brought up when considering that in a station black out not only the core, but also the spent fuel pool (which is inside the containment in German PWRs) contributes to pressurization. Recent results indicate that the venting system will probably not always be capable of controlling the pressure to the desired extent. As a second consequence, the filters may experience beyond design loads. This is a preliminary and purely deterministic result – no L2 PSA approach has been made yet to determine the influence on the containment failure probability.

## 4.9.2 EDF&IRSN, FRANCE

### 4.9.2.1 Status

EPR (PWR): no containment venting has been designed as the containment structure has been highly improved; additionally heat removal can still be operated in case of station blackout by ultimate diesel power supply.

French Fleet (PWR): containment venting has been added to the original Westinghouse design to avoid any containment failure due to slow over-pressurization. A metallic filter in the containment can retain a large quantity of aerosols and a sand filter, outside the containment should retain the remaining aerosols. Nevertheless, improvement of filtration efficiency (for iodine and noble gases) is under discussion in France. The reinforcement of the venting system to seismic hazard is on-going (post-Fukushima decision).

SAMG requires the heating of the venting line to avoid the steam condensation and to limit the risk of hydrogen combustion within the venting line. Opening of the containment venting is decided by the crisis team, in a predefined range of containment pressure (typically around 5 absolute bar).

In case of shutdown states (SG not available) and loss of ultimate heat sink, the system can be opened by applying EOPs. For these situations, SAMG requires to close the system at the beginning of the severe accident.

### 4.9.2.2 EDF L2 PSA modelling

French Fleet (PWR): The L2 PSA takes into account possible human failure as the opening of the containment venting is manually operated. Considering venting failure, containment is supposed to be lost in this case in the L2 PSA.

### 4.9.2.3 IRSN L2 PSA modelling (for 900 and 1300 MWe PWRs)

The following issues are considered:

- heating the venting line (immediate action to do at the SAMG entry): this local action is modelled with the HRA PANAME model (§4.3.1.3.1) ; for SBO situations, the heating is not considered;
- opening the FCVS: the HRA HORAAM model considers that it is a difficult action (requiring the wearing specific equipment, mask…).

The conditional failure probabilities are between 0,001 and 1, depending on context factors.

In case of failure of the FCVS opening, the L2 PSA modelling considers non-filtered radioactive releases.

## 4.9.3 IEC, SPAIN (BWR)

Containment vent is a seismic hardened pipe with two isolation valves manually actuated from Control Room and locally in case of SBO using the nitrogen bottles or compressed air with DC power or portable diesel generator.

During the severe accident phase, the access for a containment vent local action is not possible. Additionally, local actions on a few valves are needed during a SBO sequence to cover the containment isolation function.

An evaluation of habitability of control room in a SBO with vessel failure and containment venting has demonstrated that it is not necessary to leave Control Room if the emergency filtered system is activated before the venting.

Containment venting is prepared for higher flow rates that those from severe accident phase, but the radiological instrumentation need some improvement to cover the severe accident phase.

External hazards are not currently analyzed in human reliability for L2 PSA.

Containment venting is also considered before severe accident in EPG to reduce pressure to prevent, among other issues, a venting in an early phase of severe accident.

In SAMG there is not a specific value to close the containment vent and so it is not modelled in L2 PSA, remaining open. Thus the source term release in a severe accident sequence with containment vent is the same than with containment failure, although the final structural state of the containment is very different.

Filtered containment venting is planned to be installed for the next years. This system will be designed to fully cover the severe accident conditions (hydrogen burns risk and radiological shield) and the open and close actions will be modelled into the L2 PSA. In these cases, a different treatment will be used for a control room actuation instead of a local actuation.

PSA could be used to know the impact of the early venting before the severe accident phase or the different open/close management during severe accident phase.

The plant has heat exchangers for suppression pool cooling and for containment spraying, used for the containment pressure control before containment venting and considered operable during severe accident.

In L2 PSA this equipment is used to credit internal water injection sources and to determine if the containment is pressurized or not.

## 4.9.4 INRNE, BULGARIA

### 4.9.4.1 Status

For the Kozloduy NPP, units 5 and 6, the different approaches are foreseen for the containment protection in accordance with the different threats to its integrity.

The following effects that would endanger the containment integrity are[22]:

- pressure increase caused by coolant leakage accidents;
- pressure increase caused by a process of direct heating of the containment atmosphere by the melt when it leaves the reactor;
- effects of burning and detonation of hydrogen produced by a steam-zirconium reaction and molten corium concrete interaction;
- steam explosion caused by the melt-water interaction outside the reactor vessel;
- loss of strength of the containment building structure caused by prolonged heating. In that case, a containment failure is caused by a combined impact of two factors: temperature and pressure.

For each of the listed effects relevant to SA progression, technical means are foreseen for removal of the risks to the containment integrity and to achieve the least possible release of radioactive products. Thus, the strategy ''containment conditions management'' is a combination of separate strategies (each related to a specific threat to the containment). Each of the threats, as well as their simultaneous impact, requires a complex approach in the management of the containment conditions, which is why they are combined in one strategy. It should be noted that as a result of the unit modernization programs, the following SA management systems that are directly related to containment protection have been installed:

- containment overpressure protection system through medium discharge;
- containment filtering and venting system;
- passive autocatalytic hydrogen recombiners;
- medium filtration.

The VVER-1000 has a containment designed for a maximum pressure of 4.9 bar (absolute). The structure and the properties of the design are the same as those of PWR-type containment. To provide protection against a slow pressure increase there is a spray system. A complementary filtering and venting system of the scrubber type has been installed. The operation of that system guarantees preservation of containment integrity and purification of the medium that is released outside.

If the Spray system is not running after first couple of hours it is forbidden to be used for containment depressurization. The operator should switch on manually scrubber if it is not activated after reaching its set point for supporting containment pressure.

### 4.9.4.2 L2 PSA modelling

For the Kozloduy NPP, according to the recent L2 PSA study [41], the availability of the system and their effect on the accident progression included in the CET model are Spray System and Passive Filter Ventilation system. Modeling approach of the systems has two aspects: system unavailability or conditional failure probability is done in a same approach as in L1 PSA and interface analysis and qualitative analysis for equipment located in containment is performed based on MELCOR results. The qualitative analysis determines whether system components in the containment status failed or not (due to beyond basis environment conditions) and system unavailability analysis analyses the stochastic nature of system failure possibility.

## 4.9.5 SSTC, UKRAINE

### 4.9.5.1 Status

Filtered containment venting system will be installed at all Ukrainian NPPs as SAM measure to prevent containment overpressure failure in the case of design containment spray system failure or inefficiency. Currently the measure is not implemented in a full scope at any of the units. However for SUNPP Unit 1 the first stage of this SAM measure allowing to perform unfiltered venting is completed and technical requirements to filtered equipment are prepared based on the results of analytical justifications. Proposed system layout partially utilizes existing venting system inside the containment (correspondent venting ducts are replaced with steel pipelines) while outside the containment new lines will be installed. It is assumed that the system operation will be controlled by operator.

Filtration effectiveness is selected so as to eliminate the necessity of evacuation outside the plant site (sheltering is allowed). However, specific design decisions on filtration method (wet or dry filtration) are not fixed yet. The necessity to install aerosol filter inside the containment to separate high diameter particles is recognized by the Utility and regulatory authority. In order to decrease on-site radiological consequences of FCVS operation the filtered mixture will be dumped through the ventilation stack of approx. 100 m height.

To ensure main and emergency control room (MCR, ECR) operability and habitability during SA the air conditioning systems were modified to withstand harsh conditions and seismic impacts, and for VVER-440/213 units (Rivne NPP Units 1,2) iodine filters were installed. In the case of SBO it is envisaged to provide power supply to the emergency lighting, communication equipment, MCR and ECR air conditioning and heating from mobile diesel-generators.

Considering the measures taken to ensure MCR and ECR operability and habitability it is expected that external hazards will have insignificant effect (if any) on human reliability of controlling FCVS operation.

While it is recognized that later FCVS actuation results in lower radioactive releases through the system there is no sufficient confidence the containment penetrations/seals are able to withstand high pressure, temperatures and radiation impact exceeding the ones considered in the design. Therefore the proposed strategy of FCVS usage is to activate system when the containment pressure reaches the maximal allowed design value (4.9 bar) for VVER-1000 units).

To ensure the hydrogen safety the system is inerted with nitrogen prior to its actuation. Existing analyses suggest that for successful strategy implementation only one cycle of FCVS opening/closure is sufficient. Nevertheless the system design shall provide sufficient nitrogen supply to allow for multiple FCVS usage cycles.

It is assumed that containment venting will be required in SA progression scenarios with slow containment pressurization. Fast containment pressurizations scenarios associated with potential hydrogen detonation are excluded via proper selection of PARs productivity and location. Available calculation results with combined PARs and venting system operation support this assumption. Thus existing L2 PSA fault trees for containment slow pressurization sequences need to be supplemented with human actions and correspondent system success criteria.

For VVER-440 units (Rivne NPP units 1,2) the containment failure caused by overpressurzation is excluded due to high containment leakage rate (approx.15% per day). The analyses results demonstrate that maximal containment design pressure (2.45 bar) is not exceeded even at the ex-vessel SA phase. Nevertheless, taking into account continuous effort of the Utility to decrease containment leakage rate at VVER-440 units and in order to minimize

uncontrolled radioactive releases it is decided to install venting system at these units with the main objective to provide filtered removal of gas mixture during SA at containment pressures below the maximal design value.

One of the ways to decrease containment pressure is to start spray train. SAMGs have special criteria on hydrogen concentrations (and oxygen) which allow spray start. PARs system is designed to withstand even inadvertent spray start in worst time (both in-vessel and ex-vessel phase). In that case the combustion is possible but with low resulting pressure and without flame acceleration.

### 4.9.5.2 L2 PSA modelling

Prevention of containment overpressure failure is modeled in current L2 PSA as CET top event. The conditional probability of containment failure is determined based on the analysis of the stability/strength of the containment structure. The containment failure mechanisms were subdivided as follows:

- failure caused by hydrogen combustion;
- failure due to static pressure increase which is not associated with hydrogen combustion.

To evaluate potential of containment overpressure during SA correspondent MELCOR analyses were performed.

## 4.9.6 JSI, SLOVENIA

### 4.9.6.1 Status

Slovenian reactor, PWR type, installed a fully passive containment filtered venting system (PCFVS). A filtered vent system effectively eliminates the 'slow pressurization' containment challenge mechanism. It does this by providing a means to vent the containment free volume via a high efficiency filter to the environment via a stack.

This system was required by the Slovenian regulator following the March 2011 Fukushima Dai-ichi nuclear power station accident. The PCFVS mainly consists of five aerosol filters inside containment, and an iodine filter inside the auxiliary building and various auxiliary components (such as valves and rupture disks) to ensure its fully passive operation during more than 72 hours. It is designed for severe accidents.

A compact and modular dry metal fibre filter to capture the aerosols instead of using a large water tank that other vent designs utilize was the first-of-a-kind design. This approach allows for significant flexibility on where the filter can be installed, and at Slovenian PWR, part of the filter was installed in the containment building. New plant stack was anchored on the reactor building. The dry filter vent system is maintenance-free system that does not require any auxiliary systems for chemistry control, heating, draining, and the like. The system is fully passive and does not require any external electric or other power sources during standby or in operational mode.

Generally, vented gas will be steam inerted (even if high in hydrogen) provided containment pressure is above 2-3 bar absolute. Dry filtered vent system is not expected to be vulnerable to hydrogen, since there is no mechanism for large scale condensation of steam within the system. The flammable mixtures are only likely to occur at the outlet to the environment – usually at the stack exit.

The SAMG has already been adapted to consider filtered vent. If the set point for critical containment pressure is not reached, containment heat sink depressurization sources like containment spray or portable severe accident management equipment pumps are preferred means. The preferred means for venting is PCFVS. If it cannot be

actuated, the unfiltered vent systems are to be used. In such a case the personnel in the vicinity of the vent path should be evacuated.

### 4.9.6.2 L2 PSA modelling

Regarding the impacts on human reliability in case of external hazards, the PSA analyses show that average human impact on accident mitigation (internal and external hazards) is around 34 %. In case of containment venting there is threat only to plant personnel, since the control room is closed in case of accidents and has a closed cycle venting. In case of accident all plant personnel is evacuated or is called in emergency centers. The filter clogging is not an issue as it is solved with appropriate design of the filters and therefore it is not taken into account this possibility. External difficult conditions (loss of electrical power supply and lighting, high radiation, etc.) and supply availability has no impact on venting performance as the filtered vent system is passive system. To optimize the containment venting, L2 PSA was used to provide the most probable core sequences that lead to plant damage states (PDS), where containment venting can be used for prevention of large releases. Because PCFVS is passive, the containment venting it is considered to be 100% successful.

Concerning heat exchangers the Slovenian PWR is such, that this is not applicable. In general, L2 PSA provides input to SAM. All initiators (internal and external) and their consequences are considered in SAM.

## 4.9.7 NUBIKI, HUNGARY

The assessment of the PSA provides the SAMG developer with key plant-specific information regarding the type and relative importance of the different modes of failure and challenges to fission product boundaries.

Appropriate accident management strategy may influence the results of the PSA analyses and mitigates the consequences of the accident. On the other hand, some strategies, beneficial for a given challenge, may also have negative impacts as far as another challenge is concerned. The risk of containment long term over-pressurization failure mode is decreased by:

- water injection into containment;
- containment venting;
- containment heat removal strategies.

### 4.9.7.1 Status

Until now, slightly modified systems are used for the containment pressure control during severe accident. The SAMG describes the use of these systems, which are the followings:

- « Filtered vent » of the containment through the sucking ventilation system (the aerosol filters were changed);
- Containment penetration cooling system;
- Spray system.

A new system under realization is the long term containment cooling system (severe accident spray system). The water source of this system comes from the containment sump (water from bubbler trays) and the new dedicated severe accident diesel will provide the energy supply for the water pumps. The air cooled heat exchangers will be located on the top of the localization shaft (containment). The pumped water flows through the heat exchanger. It

is cooled down and the cold water enters into the containment at the top of the localisation shaft. The cold water pipe inside the containment has hundreds of small holes to distribute the water in the localisation shaft at the level about 40 m, it works as a spray system.

The use of the "filtered vent" is determined in the severe challenge guideline, in a special part of SAMG. If the containment pressure has reached the point where containment failure is possible, this guideline will direct the Technical Support Center to depressurize the containment via venting. In spite of the obvious short-term negative impact: the release of nuclides, it prevents a later and larger release.

The effect of the use of the spray system is more complex. The SAMG describes the potential benefits of operating the spray system and it also identifies and evaluates any short and long term negative impacts. The guideline also lists the spray system limitations (under what conditions it can be used). Last but not least it clearly defines the actions necessary to put the spray system into operation.

### 4.9.7.2 L2 PSA modelling

The process taken in consideration for restarting of the spray system on the basis of SAMG is described below.

First the representative sequence was calculated by the MAAP code. Sensitivity calculations were performed for the effect of starting the spray system in a certain time window. The results included oxygen, hydrogen and steam concentration, hydrogen burn load as a function of time and the start-up time of the spray system.

The circumstances of the intervention were described in the next step including definition of the initiating event, availability of the spray system and other failures associated with the accident sequence in question. Generally, during a severe accident, the circumstances are complex (multiple failures lead to a severe accident). Moreover, if the initiating event is an area event (e.g. fire or strong earthquake), then the circumstances are very complicated. The procedures used before entering the SAMG also influence the probability of starting the spray system according to the SAMG.

The staff should first decide to examine and determine the availability of mitigation systems (spray, fan coolers). In the next step a decision has to be made whether the spray system should be started up or not, based on measured data (simulated by MAAP calculation in an exercise) on containment pressure and in 8 measurement points in the containment: temperature, hydrogen and oxygen concentration. Supporting pre-drawn diagrams are available that are used to determine the concentration of hydrogen in the containment and assess the potential for hydrogen burn on the basis of a three-colour (green, yellow, red) scheme. If the region coloured green/red applies, then it is a clear indication of the need to use the spray system. If the colour is yellow, then the situation is unclear. It is a limiting factor that the diagram is valid only for the atmosphere of saturated steam. If the calculations witness a situation where the steam is not saturated, then there are uncertainties as to the appropriate response to the accident.

There are also set points for spray operation: minimum pressure to avoid containment failure due to negative pressure, minimum containment water level and minimum water level in the ECCS tank to avoid pump failure.

Teamwork within the technical support center (TSC) as well as communication and co-operation between the TSC and the main control room crew are needed for the spray activation. The required level of co-operation was assessed medium on a three-point, behaviorally anchored rating scale.

The knowledge and training level of the staff is very much sequence dependent and it was considered medium in this case. There had been lectures and classroom training sessions, but not covering all types of accident sequences. The time available for preventing large fission product release by spray injection is also sequence dependent: it varies between 10 minutes (very short time) to 1 hour (long time). For the pressure reduction the available time is generally long or very long (several hours).

The probability of starting the spray system was determined by incorporating all the above information into the decision tree.

## 4.9.8 TRACTEBEL, BELGIUM

Currently, the containment pressure is mainly controlled by the containment sprays system in Belgian units. The installation of the FCVS system is still ongoing.

L2 PSA studies have allowed identifying two points of attention related to the operation of the containment sprays system:

- Firstly, in Belgian units it is possible to perform the "safety injection system to the containment sprays system connection" by human action, in order to backup the containment spray system pumps by the safety injection system pumps, in case of their unavailability. In this way, it is possible to control the containment pressure with the safety injection pumps.

  L2 PSA results have revealed a significant positive impact of a successful implementation of this "safety injection system to containment spray system connection". However, it has been found through the L2 PSA studies that the necessary manipulations of valves to establish the line-up for this connection to back-up the containment spray system by the safety injection system are not always clearly indicated in the guidelines of certain units.

  Consequently, it was recommended by the L2 PSA studies to complete the guidelines, in order to reduce the human error probability to perform this action.

- Secondly, it has been observed in the L2 PSA studies of certain units that, the activation of the containment spray system after vessel failure may lead to non-negligible hydrogen risks due to the rapid steam condensation in the containment. Consequently, the necessity for the evaluation of the hydrogen impact when depressurizing the containment has been emphasized in the accident management.

Finally concerning the containment venting, currently it is possible to perform a non-filtered containment venting in one of the Belgian units. This human action is foreseen in the guidelines of the unit. The L2 PSA study of this unit has revealed the following two points of attention regarding the containment venting action:

- Although foreseen in the guidelines, it has been shown by the L2 PSA study that the probability of considering the containment venting action is very low. This is due to the fact that the valves for the non-filtered containment venting have to be manipulated locally and they are located in a non-shielded area, leading to a very low accessibility to these valves.

Thus the necessity to have an appropriate location from which the FCVS can be manipulated has been emphasized.

- Moreover, although performing a non-filtered containment venting allows controlling the risk of potential containment failure due to long term pressurisation, it also leads to significant FP releases in the long term. This has clearly supported the interest for a FCVS.

## 4.9.9 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| Containment pressure control | |
|---|---|
| FCVS – Design | The L2 PSA provides information on:<br><br>- scenario leading to containment venting,<br>- safety improvement due to filtered containment venting,<br>- causes of FCVS failure,<br>- need, feasibility and inconvenience of multiple FCVS open/close cycles during severe accident phase.<br><br>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.<br><br>At design phase, L2 PSA can be used to support decision in FCVS (or alternative solution) construction and define functional requirements. |
| 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.<br><br>Considerable uncertainty is related to ignition probabilities. As shown by the Fukushima examples, ignition seems to be an almost random event.<br><br>In a L2 PSA performed for a PWR many years before Fukushima [9], 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.<br><br>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. |
| Others 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 :<br><br>- the system availability for accidents coming from L1 PSA,<br>- robustness for the defense-in-depth concept (are such systems still available if the situation has already conducted to fuel melt),<br>- risks associated to flammable gas : how is the containment atmosphere flammability controlled (typically effect of the steam condensation by spray, by external wall spray …) ?<br>- risk associated to the leakage on the circuits (typically the risk of sump contaminated water release).<br><br>After having explored all situations, L2 PSA shall help obtaining guidance for a safe |

| operation of these SAM. |

# 4.10 RADIOACTIVE RELEASE ISSUES (E.G. PH CONTROL IN THE CONTAINMENT, SOURCE TERM ASSESSMENT)

The limitation of radioactive release into the environment is one of the main SAM strategies. The most efficient way to limit these releases is to maintain the containment integrity and tightness from the beginning of the accident to the NPP stabilization (at atmospheric level). Any uncontrolled failure or bypass of the containment during a fuel melt accident can induce catastrophic offsite consequences, in particular during the early phase of the accident. Maintaining the containment integrity is the first priority of SAM strategies.

As objective during severe accident on the first place is to minimise radioactive releases into the environment from the very high source term available inside the containment. Some of the most important challenges for limitation of radioactive releases into the environment are:

  i.   Failures of containment isolation valves to close properly upon request. Respectively measures for assurance of the containment isolation valves closure are needed in SAMG. These have to be applied also to the personnel hatches and hatches for observation and maintenance.

  ii.  Leakages on a protection systems during sump recirculation mode in equipment located outside containment. Respectively measures for prophylaxis and maintenance of equipment, control of seals and prevention of their violation, etc. are of high importance.

  iii. Failures of filtered venting system (FVS). Respectively measures for prophylaxis and maintenance of the FVS and assurance of thermal conditioning and other preparatory operation of FVS system are needed.

Iodine is a significant contributor to the doses caused by a severe reactor accident. This is due to the fact that iodine can exist in a highly volatile form that cannot be easily removed from the containment atmosphere.

pH in the sump is one important parameter in determining the formation of volatile iodine. The lower the pH of the sump water, the higher the fraction of volatile iodine. The sump water pH can be controlled by adding suitable chemicals to the sump. The equipment for feeding the chemical to the containment, as well as the storage tanks, should be protected against the hazards that may lead to, or result from, a severe accident.

## 4.10.1 EDF&IRSN, FRANCE

### 4.10.1.1 Status

EPR (PWR): pH control of the sump is achieved by Sodium Hydroxide injection (either safety or spray injection).

French Fleet (PWR): pH control of the sump is achieved by Sodium Hydroxide injection by spray system. On 1300 MWe and N4 reactors a pH control of the sump is also achieved by passive dissolution of sodium tetra borate. For 900 MWe, Silver concentration inside the control rod will fix iodine into the sump once melted, so no additional passive dissolution of sodium tetra borate is necessary.

### 4.10.1.2 EDF L2 PSA modelling

EPR (PWR): Sodium Hydroxide injection (either safety or spray injection) are modeled in the L2 PSA. Failure of these systems leads to specific release category.

French Fleet (PWR): For 1300 MWe and N4 reactors the Sodium Hydroxide injection system used to be modeled in the L2 PSA, leading to specific release category, but it is not relevant any more considering the passive dissolution of sodium tetra borate.

### 4.10.1.3 IRSN L2 PSA modelling (for 900 and 1300 MWe PWRs)

The APET distinguishes cases with and without spray system (for sodium hydroxide injection). Source term calculations take into account the sump pH control, the impact of silver in sump, and main chemical reactions associated to iodine.

Radioactive releases are calculated for all type of accidents. A very fast running code (MER) is used for that purpose. That gives a possibility to present the risks in function of frequencies and consequences of each accident. Efficiency of each measure that helps reducing the release can be assessed (aerosol deposit by spray, pH control, containment tightness control …).

For example, the pH control can:

- delay the gaseous halogens release (I2) from containment sumps;
- reduce significantly the molecular and organic iodine releases.

## 4.10.2 JSI, SLOVENIA

### 4.10.2.1 Status

For Slovenian reactor, PWR type, the pH control is obtained by trisodium phosphate (TSP) in crystalline form. The TSP is stored in baskets, placed on the containment building floor and distributed along the wall to ensure a uniform distribution and mixing in the post-LOCA recirculation water. The long-term pH is raised by dissolution of solid TSP. Adjusting the containment sump pH greater than 7.0 will result in retention of iodine in the containment sump water and may mitigate iodine releases. In the SAM the containment sump pH is of long term concern. Namely, reduced pH levels might also degrade the long-term retentive capacity of fission products of the accumulated water. In a case of low pH the SAM recovery action is injection of buffer solution into containment.

### 4.10.2.2 L2 PSA modelling

The pH control is passively achieved therefore these equipment is protected against hazards. There is no system used for pH control, therefore no PSA modeling is provided.

## 4.10.3 TRACTEBEL, BELGIUM

The pH control of water in the containment sumps is done by the addition of NaOH.

The control of pH has an impact on the source term evaluation, as it allows a better retention of FPs in the liquid phase in the containment. Indeed, the sump water pH control along with the containment sprays are the two

essential means to retain more FPs inside the containment and reduce the FP present in the containment atmosphere.

According to SAMG, the actions to perform the pH control by adding NaOH in both the Early and the Late phases are implemented in the L2 PSA model. The potential reduction of the FP inventory in the gas phase inside the containment depends strongly on the success of the SAM actions of pH control and of containment spray.

## 4.10.4 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT

| 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. <br><br> For example: <br><br> • reliability of the sodium hydroxide injection system, <br><br> • pH control with passive system, <br><br> • pH control during the latest phase of the accident. <br><br> 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 : <br><br> • reactor containment leakage rate shall be as low as possible, <br> • 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, <br> • high pH reduces gaseous iodine, <br> • 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)), <br> • containment heat removal with no containment venting is the best solution to limit accident consequences, <br> • containment venting system shall include filtration device or limit cliff edge effects. <br><br> 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. |

# 4.11 SAM STRATEGIES FOR SPENT FUEL POOLS (SFPS)

General SFP issues are covered in deliverable D40.6 of the ASAMPSA_E project. The present section deals with particularities of SAM related to SFPs.

SFPs require water to cover the fuel not only to prevent fuel damage, but also to provide shielding against gamma and neutron radiations emitted by the spent fuel.

Depending on the shielding provided by the fuel-building structure, the loss of the shielding effect of the water can severely restrict the movement and operation of staff in the vicinity. The focus of mitigation measures is, therefore, generally on recovery and maintenance of water levels and temperature in the SFP.

The requirement for cooling of spent fuel diminishes as the fission products in the fuel decay. The time after fuel discharge is therefore a key factor. Gas-reactor fuel can generally survive in air after some days of cooling, but light-water reactor fuel requires longer times before it can be stored in air without some degree of fuel damage. The EPRI pilot application [44] states that based on MAAP results, the minimum time to successful air cooling in the SFP for PWR fuel assemblies is approximately 230 days following a 1/3 core offload. By contrast, BWR fuel assemblies which are smaller and have more relative surface area may be air coolable in a shorter amount of time (e.g., ~100 days following a 1/3 core offload [45]). These criteria are highly dependent upon the arrangement of the spent fuel in the SFP. However, when the fuel is freshly discharged from the reactor, without water covering the fuel and providing cooling, the zirconium cladding can reach ignition temperatures and the fire can spread to older fuel. Analysis suggests that distributing freshly-discharged fuel throughout the SFP can significantly improve the cooling compared to concentrating the freshly-discharged fuel and hence increase the grace time before ignition. Segregation of the SFP into regions can also potentially affect the amount of fuel at risk.

## 4.11.1 GRS, GERMANY

Until now, there was no L2 PSA for spent fuel pool accidents in Germany. However, GRS has performed some research on accident in spent fuel pools which allow drawing some conclusions which may be of interest for an extended L2 PSA. The following considerations apply to PWR and BWR spent fuel pools alike. Specifics for these two reactor types follow below.

- Depending on the operation mode, spent fuel pool and RPV are connected or not by the fuel transfer bay. Therefore, different scenarios can develop. In principle, the connection increases the possibilities for mutual cooling of RPV and spent fuel pool, but on the other hand the accident is aggravated if both fuel repositories are affected.

- The spent fuel pool has, depending on the operation mode, very different inventories and decay heat levels. During normal reactor operation, when the pool is not fully loaded and contains a low decay heat, it may be virtually impossible to arrive at melting fuel without the unrealistic assumption of a large and uncontrolled coolant leak. However, when the pool is fully loaded in shutdown mode, melting can certainly be reached.

- After boiling has started in the spent fuel pool, the adjacent atmosphere may no longer allow access to the pool. After the pool level has dropped significantly (but still covering the fuel), the radiation from the fuel additionally will render access almost impossible. One will also have to take into account that the fuel transfer machine may be stopped with an elevated fuel element. Therefore, the simple idea of approaching a pool and adding water by a makeshift device is not really feasible in many cases.

It is necessary to distinguish between German PWRs, having the spent fuel pool inside the containment, and German BWRs which have the pool outside the containment and inside the reactor building.

For PWRs, all SAM which helps maintain the containment function (PARs, filtered containment venting) are applicable to spent fuel pool accidents as well. However, the following issues need particular consideration:

- Previous severe accident analysis and L2 PSA considered only reactor core as source of loadings. The conditions inside the containment (e.g. convection, steam content, pressure) will probably be different when considering a fuel pool accident. This requires an additional set of accident analysis.
- The fuel pool is open to the containment dome, it is located at a higher position inside the containment than the RPV and it is larger. Therefore, the heat load from the melting fuel pool to the containment dome may be significant. The consequences for the containment loading and failure probabilities should be addressed. Potential SAM (not yet implemented) to address spent fuel pool accidents could be heat removal from the outside of the containment steel shell, e.g. by ventilation or water spray, or early venting of the containment atmosphere.

For BWRs, the only barrier between the spent fuel pool and the environment is the reactor building. This building is well protected against external impact, but not against internal loads from a melting fuel pool. The following issues need particular consideration:

- In principle, there is not much mitigation potential against a very severe release to the environment.
- Access to the pool in order to perform SAM may be restricted by radiation or steam.
- Hydrogen generation from zirconium oxidation and from core-concrete interaction may be significant. Combustion of the hydrogen can exceed the building load capacity (see the Fukushima experience). Potential SAM might be to install PARs, or increase ventilation rates. The latter procedure, however, will also tend to increase releases to the environment.

## 4.11.2 UJV, CZECH REP

### 4.11.2.1 Status for VVER Reactors

Basically, there are two different types of VVER reactors spread in European countries: VVER-1000 and VVER-440. Considering severe accident management for SFP, there is a fundamental difference between these reactors – while SFP of VVER-1000 is located inside the containment, SFP of VVER-440 is located outside the containment in the reactor hall (neither sprays nor passive autocatalytic recombiners are available).

**VVER-440**

Generally, the main strategy is the same as for the reactor: providing water to cover the fuel assemblies. More specifically the main actions of SAM are:

1. evacuation of employees from the reactor hall and isolation of the reactor hall,

2. proper settings of ventilation systems,

3. find and settle possible water sources and paths into SFP.

PSA was used for identification of the most probable scenarios:

- Heavy load drops,
- SFP leakage,
- Loss of SFP cooling system.

Deterministic analyses, calculated for these scenarios, provided interesting outcomes related to timing of the accidents (typically very long time windows given by low residual heat) and decontamination factor of reactor hall

for the fission products released from SFP. Release of fission products from SFP to reactor hall may be limited if 1) ventilation flow above the SFP is turned on and 2) a cover of the SFP is on place.

Typical consequences of external events may be damaged structures of buildings (esp. reactor hall above the SFP) and loss of electric power. The main possible recoveries identified and confirmed by L2 PSA are 1) installation of alternative path for filling water into SFP operated by fire brigade from outside the reactor building and 2) installation of mobile diesel generators.

There is one more specific issue related to VVER-440. In Regime 7, when all fuel is removed from reactor, there are two layers of fuel assemblies in the SFP. Considering severe accident management for SFP, this regime is the most dangerous because of the faster progression of severe accident caused by higher residual heat in SFP.

**VVER-1000.**

Considering severe accident management in SFP, VVER-1000 reactors are typical PWR reactors. The SFP is located in the containment (in one layer for all reactor states), which beside others enables to use spray system or passive autocatalytic recombiners in case of severe accident (in opposite to VVER-440).

### 4.11.2.2 L2 PSA modelling

Currently, no L2 PSA is developed for SFP. UJV plan to perform such study for Temelin SFP (VVER-1000) in 2017.

## 4.11.3 EDF&IRSN, FRANCE

### 4.11.3.1 Status

For all French reactors, SFPs are located outside the containment, in the fuel building. No SAM strategies have been developed for SFP in France. Such accident must be "practically eliminated" and efforts are done on prevention of any accident.

### 4.11.3.2 EDF L2 PSA modelling

Currently no detailed L2 PSA is achieved for Spent Fuel Pool, as it is considered that any fuel melt sequence from L1 PSA would lead to large releases.

### 4.11.3.3 IRSN L2 PSA modelling

SFP accidents identified by L1 PSA are associated to accident with large radioactive releases. There is no quantification in IRSN L2 PSA event trees.

Nevertheless, IRSN is performing deterministic analysis to describe the degradation process of spent fuel pool assemblies in case of loss of cooling or coolant situations, for different reactor states (during refueling, or during normal operation). From the results of these analyses, IRSN will conclude on the interest or feasibility of any SAM strategies for severe accident in SFP.

Introduction in L2 PSA will be considered later.

## 4.11.4 IEC, SPAIN (BWR)

SFP has been included recently in SAMG with the temperature and level control and the main systems to inject to SFP including portable equipment, as consequence of the improvement requirements post Fukushima accident.

First PSA for the SPF has been developed recently, but only covering the Level 1 phase.

## 4.11.5 SSTC, UKRAINE

### 4.11.5.1 Status

For VVER-1000 units with SFPs located inside containment, the SA progression in reactor and SFP are influencing each other if the accident affects both the reactor core and SFP (as in SBO case). However, SA in SFP is characterized by slower progression rate. Thus, if the core is loaded the reactor SA is started much earlier (3-6h after initiating event occurrence) than SFP (1-2 days). If all the core is unloaded to SFP then severe fuel damage is expected 6-10 h after initiating event occurrence. Simultaneous SA in reactor and SFP is quite improbable since it requires the reactor core to be partially unloaded which could be expected only for very short period of time during refueling outage.

To cope with SA the water supply to SFP shall be established from the containment spray system either via dedicated pipelines or by sprinkling borated water through the spray nozzles located at the containment dome. Other means of water supply (e.g., SFP feed system) can also be used if house loads power supply is available. Otherwise correspondent pumps will be energized from mobile diesel-generators which are already installed at SUNPP units 1, 2 and scheduled for other units in CSIP. As a part of post-Fukushima measures the alternative water supply to SFP is foreseen by mobile diesel-driven pumps that can be connected to various existing water sources. Usage of all currently installed means to establish SFP water supply is prescribed in current EOPs and SAMGs.

However these means are confirmed to be effective only if structural SFP integrity is preserved. If SA is caused by extensive SFP leakage the only means to provide fuel cooling is water sprinkling from the containment spray nozzles. But effectiveness of this strategy is hard to justify analytically since complicated phenomena that could not be simulated with lumped parameter codes are involved.

Currently L2 PSA covers all states and types of accidents affecting SFP using simplified analytical assumptions considering low contribution to the overall radioactive release frequency. The latter is caused by slow accident progression that provides sufficient timeframe for operator intervention. The SFP contribution will be even lower after all related CSIP measures are implemented.

If SFP accident progresses to the MCCI phase, further SA dynamics depends completely on the total thermal power of fuel assemblies stored in SFP. Higher thermal power leads to higher ablation of concrete, hydrogen generation and containment pressurization rates, which is important for estimating the overall PARs and containment venting productivity. These factors and correspondent operator actions need to be accounted in updated L2 PSA models following implementation of correspondent measures at NPPs.

The loss of containment integrity due to base plate melting if MCCI progresses is deemed to be quite improbable considering that SFP pool is located at a higher elevation than the containment base. Following SFP floor melt-through, the melt-concrete mixture falls to the compartment beneath and is expected to spread out. Because of

increased concrete fraction in the melt the solidus temperature and volumetric heat load are quite low that improves melt spreading and further stabilization.

For VVER-440 reactors SFPs are located outside of the hermetic compartments. This fact helps to involve preventive measures but also increases the radiological consequences of SA progression. The main strategy is to establish water supply to SFP from available sources. As for VVER-1000 additional means for SFP water injection are to be implemented under CSIP.

### 4.11.5.2 L2 PSA modelling

Generally the following main attributes are considered in SFP PDS grouping logical diagram:

- whether the sequence is associated with containment bypass;
- whether the containment isolation is performed/maintained before SFP fuel damage occurs;
- whether the SFP integrity is preserved prior to fuel damage;
- whether the electrical supply is available;
- is the containment spray system long-term operability ensured;
- can emergency core cooling and containment spray systems be used to provide SFP feed and cooling.

For containment event trees the development of two factors is taken into account, namely, containment state and availability of electrical power supply at the onset of fuel damage. One of SFP L2 PSA feature is dependency between function of melt cooling and decreasing of inside containment pressure because of both this functions can be performed by spray system. Another important feature is that a number of fuel assemblies in SFP may vary depending on the unit's operation mode (power operation or refueling). Consequently, systems' success criteria may vary as well. The rest of methodology is similar to reactor facility L2 PSA and no any additional CET top events.

## 4.11.6 JSI, SLOVENIA

### 4.11.6.1 Status

In Slovenian reactor, PWR type, the SFP is located outside reactor, in fuel handling building. In 2014 the SFP has been included into SAM. The strategy used is to refill the spent fuel pool. The main objectives of this strategy are to prevent melting of fuel and to mitigate the fission product releases from the fuel handling building. First ventilation of fuel handling building is directed by normal ventilation, and if it is not available, it has to be attempted by opening doors and other openings of fuel handling building. For refilling there are standard means (use of permanent pumps), fire protection system and portable equipment (pumps, fire trucks). The water can be injected and/or sprayed. In progress is installation of fixed spray system around the SFP with provisions for quick connection from different sources of water. The installation of fixed spray system and mobile heat exchanger with provisions to quick connect to SFP, containment sump or reactor coolant system are part of the safety upgrade program requested by national regulator after Fukushima accident. Currently there is no PSA for SFP. However, in progress is regulator consideration of requiring a PSA for SFP.

### 4.11.6.2 L2 PSA modelling

As mentioned above the Slovenian PWR implemented SAM strategies for all applicable events for SFP. Also all reactor states are considered in SAM. Plant is already in phase of implementation of SPF L2 PSA. When available, optimization of SAM will be performed.

## *4.11.7 LESSONS FOR L2 PSA AND SAM STRATEGIES IMPROVEMENT*

| SAM strategies for SFP | |
|---|---|
| Risk assessment | It seems that "real" SAMs for mitigating SFP accidents are being developed just recently. However, even if no SAM strategies exist, L2 PSA can be used to discuss the risk contribution of SFP and justify or not further considerations. |
| 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. |

# 4.12 LINKS WITH EXTERNAL HAZARD

There is a need to consider feasibility of command and control of the NPP systems during external events. For example, the recent paper [46] states that in the frame of current SAMG, there is no analysis of the practical aspects of recovery actions in the worst case scenario. It was assumed that even during the severe accident progression, there would be something available for a successful recovery action, the feasibility of which was not seriously investigated. Nevertheless, 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 …." [47].

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).

NEI 06-12 [48] stated the following when describing the purpose of EDMGs:

"The term, "extensive damage," is used to connote the potential for spatial impacts that are quite broad. Such damage may not only affect equipment, but may affect the ability of plant operators to monitor plant conditions

and gain access to equipment in portions of the plant. In addition, due to the nature of some beyond design basis threats, it is possible to envision combinations of failures which might be considered of negligible probability in traditional severe accident analysis. Thus, the boundary conditions applied for EDMGs are substantially different from those used in defining plant operating procedures and even severe accident management guidelines (SAMGs). EDMGs are not a replacement for normal emergency operating procedures (EOPs) or SAMGs. Rather, EDMGs are developed on a plant-specific basis to allow the site to define the kinds of responses that may be appropriate in the event such conditions occurred." Two types of EDMGs were considered [48]: Initial Response EDMGs and Technical Support Center (TSC) Response EDMGS. The scope of these Initial Response EDMGs would include [48]:

- 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.

The purpose of the initial response EDMGs [48] is: "to define the actions to be taken in the event normal procedures and/or command and control structures are not available. The entry conditions for this EDMG might include loss of plant control and monitoring capability due to a large explosive or fire. This could take the form of damage to the control room and alternate shutdown capabilities, or loss of all AC and DC power, or all of these. An example of such a condition might involve a large fire or explosion that affected the main control room, control room personnel, and alternate shutdown capability. In such a condition, it is possible that remote instrumentation may not be available and the availability of main control room personnel may be in question. In such a condition, a number of immediate actions could be required, without the benefit of normal command and control functions."

For example, to locally start TD AFW pump (turbine driven pumps of SG feedwater system) access to the building should be confirmed (radiation levels and temperatures permit access). From the special equipment portable lighting may be required. When plant damage of key structures is assessed (containment, control building, auxiliary building, turbine building, intake structure), visible damage, accessibility and equipment status/system integrity is considered. Establishing EDMGs for initial site operational response would allow utilities to "pre-think" their strategy if normal command and control is disrupted.

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 requirements (10 CFR 50.54(hh)(2) and subject to NRC inspection. Following Fukushima Dai-ichi accident, the recommendation on strengthening and integration of Emergency Operating Procedures, Severe Accidents Management Guidelines, and Extensive Damage Mitigation Guidelines has been also done by U.S. NRC.

Also in Slovenia, improvements which will be addressed during the Action plan implementation following extraordinary safety review performed due to Fukushima Dai-ichi accident include EDMG (aircraft crash, security events) [51].

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 [52] 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. 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 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 50.54(hh)(2) guides. The future EDMG may rely upon FSGs.

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 [49]. The study [49] 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 [50]. 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.

For external events HRA, there are three types of post-initiating event operator actions: internal events operator actions, preventive operator actions, and external event response operator actions. The internal events operator actions associated with these human failure events (HFEs) are actions required in response to a plant initiating event and/or reactor trip. Because internal events operator actions have been identified, their HFEs defined, and their HEPs quantified as part of the internal events HRA, it is not necessary to repeat the internal events HRA identification process. All that is required for the external events PSA identification process is to determine which of these HFEs could occur in external events scenarios.

Preventive actions would be plant and external event specific, and the identification of these actions would be performed by a review of procedures and discussions with plant operations. These actions would typically be included in the external events PSA on as-needed bases. Preventive operator actions are an area of ongoing study. Example of preventive actions could include:

- Closing doors or placing flood barriers, such as sand bags or drain plugs, prior to flood damage;

- Transporting additional diesel fuel on site prior to an expected prolonged loss of offsite power such as a hurricane;
- Staging portable equipment (e.g., preparing to implement FLEX options).

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 <u>CONCLUSION / RECOMMENDATIONS</u>

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. The following recommendations have been highlighted.

| Emergency team activation, rooms habitability, instrumentation, … | |
|---|---|
| Emergency Team | L2 PSA shall be able to identify scenarios:<br><br>• where the emergency teams can fail to manage the beginning of the severe accident due to context factors like time constraints, external hazards, extreme conditions …,<br>• 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 | |
|---|---|
| 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.<br><br>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:<br><br>• the dependencies between human errors before and after severe accident entry,<br>• the impact of context factor on human errors. This can be developed for the extended PSA approach (internal and external hazards …). |
| 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 state of the unit/ knowledge and L2 PSA assumptions; a dialog between L2 PSA teams and researchers is needed, in order to know how L2 PSA sequences can be modelled with physical codes (modelling of corium reflooding ...).

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 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. 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 bypass. 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 :<br><br>• reduce as far as occurrence probability of energetic phenomena (hydrogen and carbon monoxide explosion, steam explosion, HPME and DCH) able to threat the confinement of radioactivity,<br>• reduce as far as possible the risks of containment bypass,<br>• maximize the conditional probability of corium stabilization after severe accident entry.<br><br>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.<br><br>No undue conservativisms 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. |
| 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.<br><br>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). |

| 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.<br><br>L2 PSA can be used to analyze the possibility of late SRV closure, for example :<br><br>• closed in a SBO situation by depletion of batteries,<br>• closed manually by the operators (error or simply because the situation seems to improve – e.g. RCS flooding).<br><br>Several issues are of interest for L2 PSA :<br><br>• the RPV re-pressurization can causes the loss of the core coolability supported with low pressure injection systems,<br>• 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,<br>• during the late phase of accident, the condition can be beyond the qualification of the SRV ; the capacity of the SRV to be operable can be questioned,<br>• the coupling between containment heat removal, RCS pressure and water injection possibility can be of crucial importance :<br>    o for example, this has conducted to the loss of steam driven water injection pumps for the Fukushima unit 2 and 3,<br>    o the containment pressure increase has a direct impact on the RCS pressure and may make some low head pumps unavailable.<br><br>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. |

| **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.<br><br>In general, the efficiency of the flammable gas management system (recombiners, inerters, …) 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.<br><br>If some specific situations can lead to containment damage, then, depending on their frequency, improvement of SA strategies shall be considered. |
| 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:<br><br>• simple approach may be practical for NPPs with significant safety margin against flammable gas combustion effects ,<br>• 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 :<br><br>• kinetics of hydrogen or carbon monoxide release in the containment ,<br>• time of spray system activation and kinetics of steam condensation ,<br>• time of in or ex-vessel water injection and effect on flammable gas release in the containment.<br><br>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 : <br>• on hydrogen production during core degradation with or without reflooding, <br>• on PARs efficiency, <br>• on PARs ignition effect, <br>• on time of combustion (stochastics phenomena except if a controlled igniter is used). |
| 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: <br>• load (pressure and temperature peak) on the containment walls and impact on their integrity, <br>• local load (pressure and temperature peak) on some key equipment for the SAM, <br>• effect of hydrogen leakage into auxiliary buildings or secondary containment, <br>• effect of hydrogen for the containment venting system. <br><br>Comment : recombiners or partial combustion reduce the amplitude of the pressure peaks in the containment ; nevertheless, approximatively 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. |
| 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: <br>• 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), <br><br>• 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), <br><br>• 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, <br><br>• to examine consequences of any leakage (transfer of contaminated gas or liquid in auxiliary buildings, transfer of combustible gas …). |


| Containment pressure control | |
|---|---|
| FCVS – Design | The L2 PSA provides information on: <br>• scenario leading to containment venting; |

| | |
|---|---|
| | • safety improvement due to filtered containment venting; <br><br> • causes of FCVS failure, <br><br> • need, feasibility and inconvenience of multiple FCVS open/close cycles during severe accident phase. <br><br> 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. <br><br> At design phase, L2 PSA can be used to support decision in FCVS (or alternative solution) construction and define functional requirements. |
| FCVS – Additional use for severe accident prevention | PSA can be used to assess the benefit/inconvenience 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. <br><br> Considerable uncertainty is related to ignition probabilities. As shown by the Fukushima examples, ignition seems to be an almost random event. <br><br> In a L2 PSA for a PWR performed many years before Fukushima [9], 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. <br><br> 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. |
| Others 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 : <br><br> - the system availability for accidents coming from L1 PSA, <br> - robustness for the defense-in-depth concept (are such systems still available if the situation has already conducted to fuel melt), <br> - risks associated to flammable gas : how is the containment atmosphere flammability controlled(typically effect of the steam condensation by spray, by external wall spray …) ? <br> - risk associated to the leakage on the circuits (typically the risk of sump contaminated water release). <br><br> 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.<br><br>For example:<br><br>• reliability of the sodium hydroxide injection system,<br><br>• pH control with passive system,<br><br>• pH control during the latest phase of the accident.<br><br>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 :<br><br>• reactor containment leakage rate shall be as low as possible,<br>• 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,<br>• high pH reduces gaseous iodine,<br>• 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)),<br>• containment heat removal with no containment venting is the best solution to limit accident consequences,<br>• containment venting system shall include filtration device or limit cliff edge effects.<br><br>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 | |
|---|---|
| Risk assessment | It seems that "real" SAMs for mitigating SFP accidents are being developed just recently. However, even if no SAM strategies exist, L2 PSA can be used to discuss the risk contribution of SFP and justify or not further considerations. |
| 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. |

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