ASAMPSA2 project: Appliance of LWR PSA2 methodology to GEN IV reactors

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Abstract

The European project ASAMPSA2 (Advanced Safety Assessment Methodology: level 2 PSA) of the 7th Framework Program aims at writing practical guidelines for conducting PSA level 2 studies on Light Water Reactors (PWR and BWR). The project includes also a supplemental task dealing with GEN IV reactors. Two main objectives are assigned to this task: 1) the verification of the potential compliance of L2PSA guidelines based on PWR/BWR reactors with Generation IV concepts; 2) a brief survey of the modelling needs to describe the new features of GEN IV reactor concepts in terms of performing a level 2 PSA.
Taking into account the ASAMPSA2 partners knowledge, the project has focused on four concepts: SFR, LFR, GFR and VHTR. For each of those concepts, a conceptual design was selected as reference: the European Fast Reactor (EFR) for sodium cooled fast reactors, the ELSY project for lead cooled fast reactor, the CEA GFR2400 project and the ANTARES project for VHTR.
As a first stage, relevant data for each concept have been collected when available. These included: 1) basic general parameters and design characteristics relevant for safety studies with a specific attention given to passive devices; 2) information about former PSA2 studies on such concepts; 3) expert reviews about accident phenomenology knowledge (like PIRT); 4) list of computational tools developed or used for accident progression studies with, if possible, some basic information about the tools (availability, level of development, validation, documentation).
In a second stage, the collected data were used to evaluate the compliance of the LWR guideline chapters with GEN IV concepts. The LWR guidelines may be divided into two main sections: chapters dealing with a specific phenomenon induced by core degradation and chapters dealing with general PSA methodology (like interface between PSA1 and PSA2, human risk assessment, system modelling and the role of expert opinion). The overall conclusion is that methodology is not very much affected by the reactor type contrary to what is related to the accident phenomena.

Keywords: GEN IV, ASAMPSA2, LEVEL 2 PSA

1. Brief survey of the ASAMPSA2 project

The European project ASAMPSA2 (Advanced Safety Assessment Methodology: level 2 Probabilistic Safety Assessment – L2PSA) of the 7th European Framework Program (EU-FP7) aimed at developing best practice guidelines for the performance of Level 2 PSA on Light Water Reactor – LWR. Pressurized Water Reactor – PWR – as well as Boiling Water Reactor – BWR – were considered. Specific concerns of the project were L2PSA methodology harmonization at EU level and methodology for uncertainty evaluation in a Level-2 PSA. As a result guidelines for both limited scope and fuel scope L2PSA, based on each of the 22 partners' practical experience, were issued.

A small part of the project was devised as an extension to GEN IV reactors with two objectives:

- to determine how far the L2PSA methodology guidelines are relevant for GEN IV concepts,
- to provide a basis for the development of new models or extension of existing models to describe the GEN IV reactors specific "mechanisms".

Although this Work Package goal is ambitious, only restricted human and financial means have been allotted.

Work started in February 2008 and has been concluded in November 2010. A draft report has been sent to a great number of organisations all around the world with an extensive questionnaire. Received answers have been analyzed and discussed during an open meeting held in last March. Based on this analysis and available time, improvements to the report will be operated before end of year 2011. Possibilities to extend the project are under discussions.

2. Applied methodology for GEN IV

During the work package kick-off meeting, it was agreed among participants to work along three steps.

The first step of the project was to select a reference design for each of the six pre-selected GEN IV concepts and to collect relevant data for each of those designs. Data availability for participants resulted in keeping only four of the six nuclear reactor concepts elected by the GEN IV forum as of special interest (sodium cooled fast reactor, lead cooled fast reactor, gas cooled fast reactor and very high temperature reactor). The molten salt reactor and supercritical water cooled reactor were set aside as nobody in the team had any involvement with these reactor types. Data considered as significant for safety issues (core features, containment features) were then collected for each of the four selected designs.

The **second step** has been the collection of a wide range of information connected with the following issues:

1. the identification of specific degradation mechanisms as the so-called core disruptive accidents for fast neutron reactors,
2. the specific provisions for prevention and mitigation of severe accidents with a special concern for passive systems which should be widely used for GEN IV reactors. An example of such systems is the Japanese FAIDUS to relocate molten fuel out of the core to ensure the reactor remains sub-critical,
3. the parameters which should be of importance for the source term evaluation. One may quote the sodium chemistry for instance and the possibility to generate new physical species by combination between fission products and sodium. Those products may be liable to more important retention than the isolated fission product,
4. the R & D needs if such evaluation was available,
5. the specificity of shut-down states as for instance the rotating plug for SFR to prevent contact between sodium and atmosphere due to its strong reactivity with air,
6. to try to get information about previous PSA2 studies conducted on similar concepts,
7. to make a first survey of available codes for PSA2 studies on such reactors (and reciprocally needs for codes).

Then once all those information had been collected as much as possible in the project frame, the **third step** has been to evaluate the relevance of the guidelines written for LWR for those GEN IV reactors.

3. The four reference designs selected

For the four selected concepts, the reference projects have been: the EFR project for Sodium Fast Reactor (SFR), the ELSY project for Lead Fast Reactor (LFR), the GFR 2400 CEA project for Gas Fast Reactor (GFR) and the AREVA ANTARES project for Very High Temperature Reactor (VHTR).

The **European Fast Reactor** (EFR) project was selected as a representative for SFR designs. The EFR project was a European project stopped in 1998 and aiming at embodying all the Western Europe know-how about sodium cooled fast breeders gathered at the time. It had been quite an advanced project (with teams having worked on it for around 10 years) ultimately cancelled when it became obvious it would not be possible to build a new SFR anywhere in Western Europe before long. It’s a

3600 MWth, fast neutron spectrum sodium cooled pool-type (so different from the MONJU loop type reactor and similar to the Super PheniX - SPX concept) reactor.

The **ELSY** project (Alemberti and al., 2011) was chosen as a representative for lead cooled fast reactors. The ELSY project - developed in the frame of the EU-FP6 by a consortium of organizations - aims to demonstrate the possibility to design a competitive and safe fast critical reactor using simple engineered technical features. The ELSY power plant is a pool-type reactor concept, sized at 600 MWe, and uses lead as primary coolant. With a core outlet temperature close to 480°C, the primary side cycle is consistent with a secondary side water-supercritical steam at 200 bars and 450°C providing a thermal efficiency above 40%. Lead was preferred to the lead/bismuth eutectics (LBE) since it is less expensive, less corrosive and of lesser radiological concern than LBE.

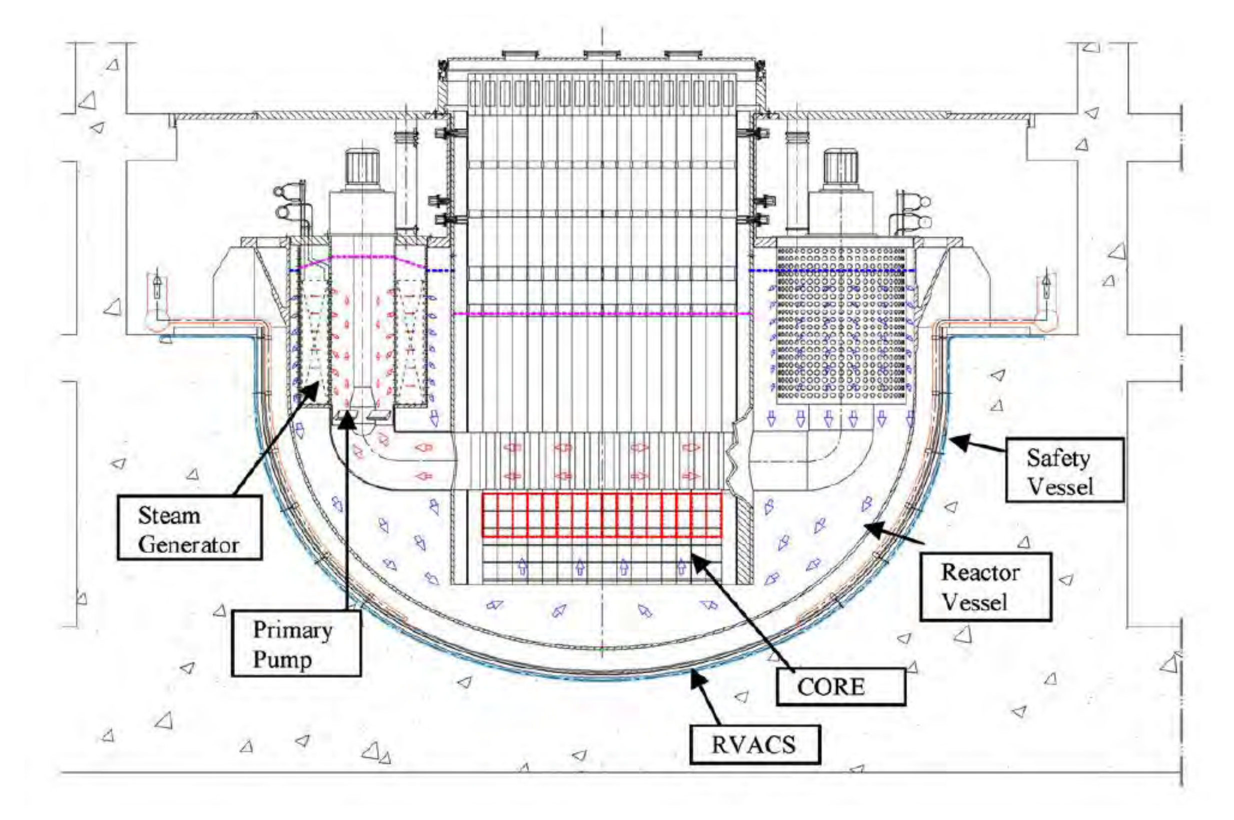

Figure 1 : sketch of the ELSY reactor concept

The **Gas cooled Fast Reactor** 2400 MWth (Dumaz and al., 2006) is a project developed by the CEA. As at least three papers presented during this OECD conference are dealing with this reactor, only limited information is provided here. No GFR prototype has ever been built although the idea to build such a reactor is rather old since it combines the advantages of the fast neutron reactors (high efficiency for electric production and possibilities of direct heat uses). As well the coolant is chemically inert (no corrosion or violent chemical reactions) and transparent which ease the monitoring, handling and repairing. However, since the helium density at low pressure is rather small, for core heat removal the reactor must be operated at high pressure contrary to the metal cooled reactors. It must be pointed out that loss of coolant accident doesn't lead to prompt criticity as in current SFRs. The reactor reference design in the project was a 2008 design with a very innovative fuel-plate concept displayed below with ceramic cladding. Some technological developments and component qualifications are still necessary to build an experimental reactor before building an industrial prototype.

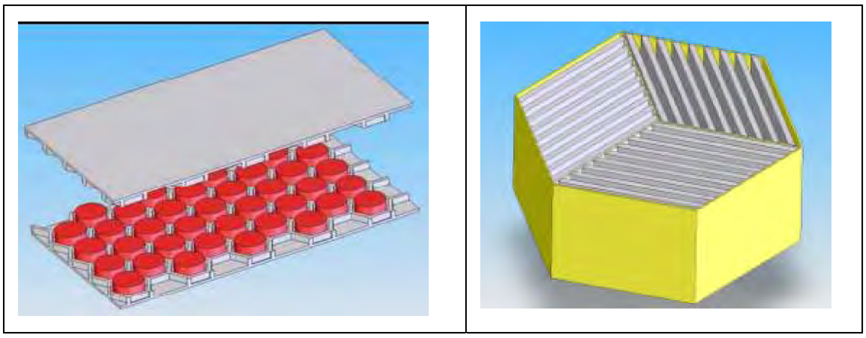

Figure 2: GFR reference fuel for the ASAMPSA2 project

The use of helium as a coolant means there are some similarities with the VHTR concept and a possibility to profit from some of the VHTR feedback.

For the AREVA **ANTARES** project (Gauthier and al., 2006) – a VHTR - some specificities of the concept may be summarized as follows:

1. it's a thermal reactor, the only one among the four projects, with graphite as moderator and helium as coolant,
2. it's an industrial project, so that access to the data remained quite limited,
3. it's a reactor design with a rather rich former history with at least five operational parent reactors having worked in the past and several projects being either built or having recently begun operation. This feature is a specificity shared with the SFR but which makes it distinct from the GFR and LFR.
4. it's a 600 MWth reactor, so rather a small reactor contrary to the three other projects.

For fuel packaging, the ANTARES project uses a prismatic bloc type core (UO2 fuel, TRISO coated particles) like the Fort Saint-Vrain reactor (USA) and different from the pebble bed concept built in Germany. The power cycle is an indirect cycle with the gas turbine located in the secondary circuit but still with a thermal efficiency of around 45 to 50% due to high temperature.

Examples of data we tried to collect for each of the four representative concepts are:

- for the primary circuit: nature of coolant, mass of the fluid, inertia of the circuit (fluid and structures), operating pressure, core inlet temperature, core outlet temperature (mean and maximum),
- for the secondary circuit: nature of coolant, mass or volume of the fluid, operating pressure and maximum temperature,
- for the containment: containment free volume, containment design pressure, maximum mass for H_2, CO and CO2 which is liable to be present in the containment,
- data needed for the assessment of the accident progression tree and related phases like main materials used for fuel, claddings, moderators and core structures and mass inventories for those materials.

Nota: in what follows, examples will be taken from one or the other design.

4. Specific provisions for prevention and mitigation of severe accidents consequences

In both PWRs and BWRs, several provisions are used in order to limit the consequences of Severe Accidents (SA). For PWR, such a provision is for instance the containment spray system, to reduce the containment pressure and remove the decay heat, or the use of igniters or catalytic recombiners for hydrogen control inside the containment.

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

1. a 3rd shutdown system. Such a system is implemented on some fast reactors and it could be self-actuated (a passive device not only for the rod insertion but also without the need of any signal: the actuation of the system is caused by effects induced by the transient like material dilatation in case of overheating of the coolant for instance) according to some GEN IV projects.
2. a specific design of the core assembly to promote the corium spreading and local recovery of cooling path. One may mention here the Japanese FAIDUS system which allows fuel ejection outside the core to prevent a Core Disruptive Accident (see below). Previous reactor designs (SNR 300, SPX, Monju and CRBR) have all designed some structures to resist the mechanical load due to a Core Disruptive Accident (CDA) in order to mitigate its short-term consequences.
3. a core catcher to collect the molten core materials is foreseen on several concepts. Both its location, inside or outside the core vessel and its composition are subjects under investigation among specialists. Collected material re-criticality is of specific concern. No core catcher is foreseen at the moment for ELSY and such a device is not relevant for VHTR (no core melting).
4. engineered safety features for containment like for instance specific filters before venting the containment to atmosphere in order to keep off-site doses within regulatory limits.

Severe accident management strategy will for sure play an important role but it needs a well-defined design to be developed.

5. Compliance and potential transposition of containment degradation modes

A short description of the major accidental transients liable to occur has been provided for each of the design based on the present knowledge. It is not possible to detail here those accident transients (and it was not the object of the work performed) but some features are useful to remind.

The Core Disruptive Accident, a accidental transient characterized by a prompt critical reactivity increase, was a central part in the safety analyses of previous SFRs and may occur in some other fast reactors. Such an accident is connected with coolant voiding effect or with fissile material compaction effects. It has a highly complex phenomenology with many possibilities in its development. The reactivity increase will end with material dispersion but may lead to fuel or steel vaporization and/or fuel-coolant interaction. A generally adopted solution is to design the primary vessel so that it can resist to a rather large amount of mechanical energy release. However it may be a challenge to demonstrate the mechanical load is well estimated.

On another hand, it must also be pointed out that for lead and sodium cooled fast reactors, the coolant choice induces a specific chemical risk absent with an inert gas as helium.

VHTR are very different reactors for which the reference accident scenario is a core heat-up accident, typically a loss of coolant flow without control rod fall. If all the active safety measures are failing for some reason, core temperature will rise but very slowly due to both the core huge thermal inertia (linked to the graphite weight) and the low power density. The reactor design is adapted so that maximum fuel particle temperature should (in theory) not exceed some reference temperature (for the moment 1600°C is currently considered).

In the WASH 1400 report about severe accidents of Light Water Reactor (LWR), representative containment failure mechanisms were depicted by the so-called α, ß, γ, δ et ε-modes. A tentative extension of this commonly used terminology to GEN IV reactors has been proposed to help creating a common language for discussions (see Table 1 below) based on the potential transients. For instance, Core Disruptive Accidents (CDA) for fast reactors have been assimilated to the α mode as have been dust explosions on VHTR.

Table 1: transposition of LWR containment degradation loss to GEN IV reactors

Mode	**SFR (EFR)**	**GFR (CEA design)**	**LFR (ELSY)**	**VHTR (ANTARES)**
α-mode	Mechanical energy release in case of Core Disruptive Accident (recriticality in case of core degradation, Fuel Coolant Interactions - FCI)	Energy release due to recriticality in case of core degradation	Steam explosion due to Steam Generator Tube Rupture	Dust explosion (or δ-mode ?)
ß-mode	IHX, DHX tube rupture Secondary containment failure	Identical to LWRs, even if containment and related systems are not well known, IS-LOCA (IHX, DHX tube rupture) combined with the containment isolation failure, HSS failure	Steam Generator Tube Rupture, Containment Isolation failure	Identical to LWRs because of the thermal loading of the IHX (failure of the isolation valves)
γ-mode	Na fire	H2 / CO emission (following steam ingress in the “carbide” core)	H2, CO/CO2 emission (following MCCI)	H2 / CO emission (following steam ingress in the graphite moderated core)
δ-mode	Na vaporization (in case of LDHR)	H2 or CO slow deflagration, failure of the guard vessel → pressurization of the Containment Building	Over pressurization in containment building	Dust explosion (or α-mode ?)
ε-mode	Corium / Concrete Interactions (MCCI)	Fuel Coolant Interaction (FCI)	Molten Core - Concrete Interaction (MCCI)	Not relevant

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

No evidence of any L2PSA for a GFR or LFR design has been found. Due to the "old history" of the concepts, such studies have been previously performed for SFR or HTR reactors and several documents are freely accessible.

No L2PSA was performed to our knowledge for the EFR concept. However, it should be emphasised that probabilistic studies were performed in the past for the US PRISM concept and the German SNR-300 reactor (GRS-51, 1982) whereas one is going on for the JSFR (and is the object of several papers in this meeting).

For HTR, evidence exist that PSA studies were formerly conducted for the American HTGR project by General Atomics, the German HTR-1160, the American MHTGR project (Everline, 1986) and the South-African PBMR.

7. Existing tools for severe accident analysis

Initially, it has been tried to identify those areas for which the phenomenology understanding is still too limited. For VHTR, PIRTs ordered by the NRC and in open access provide an up-to-date status of the art. Many information is also available for SFR and a summary has been given. No such survey is available to our knowledge for the two remaining concepts although some indications have been recorded. **Successively**, a tentative list of available codes for severe accident analysis is proposed. In many cases, the information collected is rather poor as participants may have only second-hand knowledge on several codes. Moreover, many of those codes are probably lost as they have not been used for years or have not been maintained. Anyway even a slight documentation about physical models coded may prove useful.

8. Screening of the compliance with L2PSA guidelines of LWR

8.1 Compliance with LWR phenomena and systems

Based on information collected a tentative scoring of LWR volume chapters with respect to their compliance with Gen IV reactors has been made. For each reactor and each chapter a score between 1 (high compliance) and 5 (no compliance) has been assigned depending on the evaluated relevance of the chapter for the reactor considered.

Not surprisingly, quite a large number of phenomena occurring in GEN IV reactors are not handled by the LWR guidelines (the CDA is a typical example) and, reciprocally, phenomena of importance in LWR are often absent in some or all GEN IV designs looked at. Even when similarities are present they may be quite limited. Two examples are given:

1) Molten Core Concrete Interaction is a phenomenon not to be expected for VHTR as the core should not melt. For fast reactors, it remains a possibility although due to the coolant and fuel specificities it should differ from what may occur on LWR. So scoring should be something as 5 i.e. "not relevant" for VHTR whereas it should be something average for other reactors (so a 3).

2) Hydrogen behaviour in the containment, risks associated to its detonation or explosion and the means to prevent such events to occur, the mechanical loads associated to such events and the containment answer to such loads are subjects much studied on LWR. A comparable problem is also present in GEN IV reactors even if the environment differs (containment volume, other chemical species present in the containment). So it may be assumed that past experience may prove useful for future.

What remains of interest in the ASAMPSA2-LWR work is mostly all the chapters dealing with non-phenomenological issues as human factor management, event-tree building techniques, how to make the binning between PSA1 and PSA2 etc.

8.2 L1PSA-L2PSA modelling structure

For Generation IV reactors the choice between performing a stand alone integrated L1/L2PSA model describing the accidental sequence from the initial event to the containment failure versus a L2PSA decoupled from the L1PSA should be discussed, knowing that Generation IV concept are not currently finalised. On the one hand an “integrated” model should be assimilated to a “simplified” model according the lack of knowledge and of operational feedback for these reactors but could lead to design improvement, especially for the containment building whose design is still subject to modifications. On the other hand, L1-L2 interface technique and building two decoupled models provide some advantages as:

- a capability of improvements and refinement of the models, thanks to the increase in the knowledge regarding physical situations or phenomena (through experiments, simulation...) for L2PSA.
- a decrease of the number of L2 representative initial states (and corollary the number of event trees in the L2PSA model) and therefore, a decrease of the amount of representative sequences that should be assessed by code calculations.

For the VHTR concept the question of making a distinction between L1PSA and L2PSA studies for VHTR may not be of concern:

- there is no core melting possibility with VHTR, the accident progression analysis is easier: a Level 2 analysis for a VHTR is straight forward and no change in methodology is needed.
- there is only a limited number of accident scenarios and safety systems so that it is worthwhile modelling the accident sequence up to external release using one event tree combining level-1 and 2 PSA and skip the plant damage state binning.
- some VHTR concepts as the South-African PBMR for instance have no containment which reduces the level 2 analysis significantly: no containment response analysis is needed.

8.3 Accident Progression Event Tree (APET) examples

Some attempts to build up simple APET trees have been committed. There should be seen as a preliminary step on the way to build up a L2PSA in the future.

For a **SFR**, the Level 2 PSA event tree might not be very large. The events that will be modelled will be the action of isolation of the containment and the reliability of the coolability of the corium spread on the core catcher.

For **ELSY** the reactivity increase accident implying the CDA (Core Disruptive Accident) conducting to lead boiling is not considered, given the high boiling point of the lead, with respect to sodium for example, that makes that kind of accident extremely unlikely. A potentially very severe accident is initiated by a Steam Generator Tube Rupture (SGTR), which can potentially lead to steam explosion, due to the interaction between hot molten lead and relatively cold water at high pressure. The violent expansion of this high-pressure steam bubble loads and deforms the reactor vessel and the internal structures, thus endangering the safety of the containment and the nuclear plant. The accident leads to radioactive releases into the containment due to failure of the top of the vessel. Missile emission due to the steam explosion can challenge the containment integrity (α mode). It has to be considered also the interaction of water/steam with materials potentially causing also the production of hydrogen, so that one can have early containment failure (γ mode), even if with a low likelihood. After rupture it’s possible to have a have failure of the containment due to MCCI (ε mode); γ mode failure results as combustion of H2 and other burnable gases as CO and CO2 resulting from Molten Core Concrete Interaction; finally we can have late containment failure due to over-pressurization.

For **VHTR**, typical accidental tree for a loss of coolant flow accident may be found in older sources as for instance results of the PSA studies on the German HTR-1160 (FASSBINDER). Such an event tree remains meaningful although some branches should be erased or added depending of the safety systems present on the design considered. If the safety heat removal devices do not work, core and fuel will heat up but at a rather slow pace due to the power density and huge graphite mass. At a certain time operators should depressurize the primary circuit which will enhance an important activity release inside the containment (a specificity of the VHTR is that a certain amount of contamination of the primary circuit has to be accepted so that depressurization will lead to a significant fission product transfer to the containment). Then the containment tightness and containment failure mode should be studied. The core is designed so that maximum core temperature should stabilize below a critical temperature above which fuel particle coating should fail. So the activity released inside the containment at depressurization time should remain the major contributor to the source term.

8.4 Miscellaneous

At the moment, no human reliability assessment is possible as no accident mitigation measures and procedures are defined on any of the reference concepts.

There is a clear will to use passive system on GEN IV reactors to a greater extent than was the case with LWRs. Failure assessment of such devices is a rather complex problem, combination of physics and human factors.

Several calculation tools exist. Their availability should be checked. In any case their level of validation and their applicability to the different concepts should be checked. The technical know-how to use those tools needs also to be rebuilt at least partly.

The role and extend of expert judgment will probably be significantly more important than with LWRs due to the limited feedback.

9. Conclusion

On a whole as projects on GEN IV reactors are just being restarted in the European environment, a lot of skills have to be rebuilt and designs to be more precisely defined before we can manage a complete L2PSA on any of the concepts. Simplified L2PSA may be performed at an early stage of design:

- to identify the major containment failure modes and the main phenomena contributing to containment failure,
- to estimate roughly the quantities of radioactive material released to the environment for different accident sequences,
- to help prioritising the R&D needs.

10. Glossary

BWR	Boiling Water Reactor
CRBR	Clinch River Brooder Reactor
FCI	Fuel Coolant Interaction
GFR	Gas Fast Reactor
IE	Initial Event
LBE	Lead Bismuth Eutectics
LFR	Lead Fast Reactor
LWR	Light Water Reactor
MCCI	Molten Core Concrete Interaction

PIRT	Phenomena Identification and Ranking Table
PWR	Pressurized Water Reactor
SA	Severe Accident
SFR	Sodium Fast Reactor
SPX	Super PheniX
VHTR	Very High Temperature Reactor

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IRSN

ASAMPSA2

SEVENTH FRAMEWORK PROGRAMME

ASAMPSA2 project - Appliance of LWR PSA2 methodology to GEN IV reactors

OCDE workshop, Paris

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06/20-22 /2011

PART 1 - Information on ASAMPSA2 project

PART 2 - Treatment of Gen IV reactors

ASAMPSA2 IRSN

PART 1 - The ASAMPSA2 project:

- *ASAMPSA2 stands for "Advanced Safety Assessment Methodology: level 2 PSA"*
- Project in the frame of the 7th European Framework Program (FP).
- FP = funding programmes created by the European Union in order to support and encourage research in the European Research Area.
- Goal of the project : developing best practice guidelines for the performance of Level 2 PSA for PWR and BWR

ASAMPSA2 IRSN

The ASAMPSA2 project:

- Specific concerns of the project were L2PSA methodology harmonization at EU level and methodology for uncertainty evaluation in a Level-2 PSA
- 22 partners
- Kick-off meeting in 2008
- Work was initiated with the writing and the sending of a questionnaire to end-users in Europe to clarify and crystallize their views about the performance of Level 2 PSAs
- Conclusion discussed during an open Workshop in October 2008 (Hamburg - Vattenfall)
- (e.g importance of some technical issues but also on decision-making process)
- First version of the guideline obtained at the end of November 2010.
- External review

ASAMPSA2 IRSN

December 2010 : Draft guideline sent to around 150 organisations all around the world with a questionnaire (with the help of CSNI=committee on the safety of nuclear installations and SARNET=severe accident research network)

Main objectives of the survey:

a. to determine as far as possible End Users satisfaction with the guidelines,
b. to determine to what extent the initial objectives of the projects have been fulfilled
c. to identify any need for a follow-up effort

24 answers to the questionnaire were received (more or less what was expected) and pre-analyzed by PSI (Switzerland). Among which answers 11 by project partners and 13 from outside the project (India, USA, Japan, Bulgaria, Lithuania, Germany, Belgium, Italy, France, Switzerland, Slovak Republic).

Specific comments were received, especially from organizations members of SARNET

ASAMPSA2 IRSN

- Thorough analysis of the survey performed by PSI.
- Three indicators have been defined and chapters scored according to these indicators.
- 2011, 7-9 March - Helsinki meeting, opened to non-participants of the project, to discuss the improvments to be done to the draft report (when possible within a short time)
- An improved version of the report is foreseen for next September although, as a consequency to Fukushima accident, work has rather been delayed since.
- Follows-up of the project still to be discussed

ASAMPSA2 IRSN

- Draft report is composed of 3 volumes (huge volumes 1 & 2 devoted to LWR and thiner volume 3 to GEN IV reactors)
- Chapter division in volumes 1 & 2 according to:
  - ➢ physical phenomena (ex : one chapter for core degradation, one for in-vessel steam-explosion etc.),
  - ➢ more specific PSA2 questions as: how to make the interface between L1 and L2 PSA, how to take into account the human factor etc.

ASAMPSA2 IRSN

- PART 2 - A small part (only 6 organizations, ENEA, NRG, RSE, CEA, AREVA, IRSN and with reduced time) of the project dedicated to GEN IV reactors with two goals:
  - ➢ To determine how far the L2PSA methodology guidelines are relevant for GenIV concepts
  - ➢ To provide a basis for new models development or extension to describe the GenIV reactors specific "mechanisms"

ASAMPSA2 IRSN

OUTLINE of the work performed

- **Review of the Main features of GEN IV Reactors**
  - Design Features of 4 different GEN IV Reactors
  - Degradation Mechanism and damage criteria
  - Passive safety systems
  - Calculation tools and uncertainties
  - Treatment of Hazards
- **Existing Tools for Accident Analyses**
- **Screening of the compliance with L2PSA guidelines for LWRs**
  - Compliance with LWR phenomena and systems for L2PSA
  - L2PSA Structure
  - Human reliability assessment
  - Role and extend of expert judgment
- **Conclusion and Prospect**

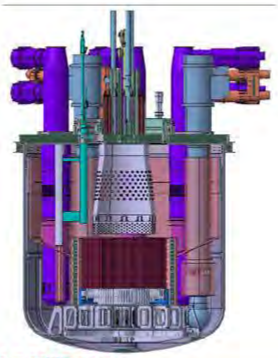
ASAMPSA2 IRSN

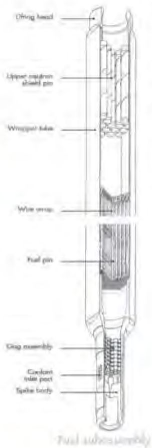
Four representative GEN IV concepts selected (choice based on data availability for project members):

- **Gas-cooled Fast Reactor (GFR)**
  - GFR project as designed end of 2007
  Developed by CEA
- **Lead -cooled Fast reactor (LFR)**
  - ELSY reactor
    Developed in the framework of the EU FP6
- **Sodium Fast Reactor (SFR)**
  - European Fast Reactor (EFR)
    Developed in the framework of a European collaboration 1988-1998
- **Very High Temperature Reactor (VHTR)**
  - ANTARES
    Commercial project designed by AREVA

ASAMPSA2 IRSN

Review of the main features of the EFR (SFR)

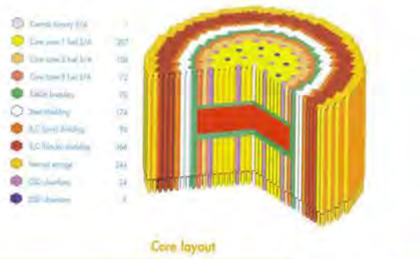




EFR: 3600 MWth

1. Fast neutron spectrum + closed fuel cycle for efficient conversion of U
2. Coolant: sodium, Tsc= 400 to 550°C
3. Pin-type core with high core power density and low coolant volume fraction
4. Pool-type reactor concept
5. Indirect Rankine thermodynamic cycle (Na filled intermediate cooling circuit and a steam-water at tertiary)

➔ thermal efficiency around 35%

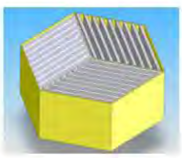


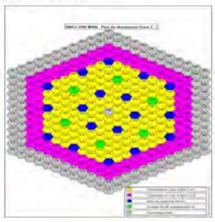
Review of the main features of the GFR

GFR: 2400 MWth

1. Fast neutron spectrum + closed fuel cycle for efficient conversion of U + management of Minor Actinides (up to 5% of MAs)
2. Coolant: helium (at 70 bar), Tsc=850°C
3. Plate-type core of (U,Pu,MAs)C + SiC coating (closing plates)
4. Combined thermodynamic cycle (indirect Brayton-cycle in the secondary circuit and indirect Rankine cycle in tertiary)

➔ thermal efficiency (45-50%).



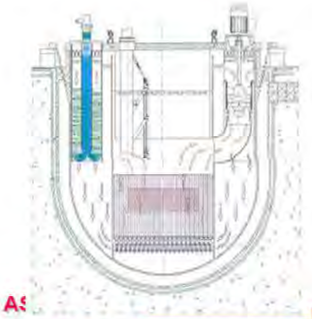


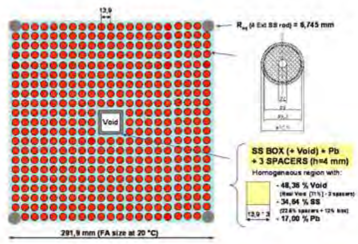
Review of the main features of ELSY (LFR)

LFR (ELSY 1500 MWth)

1. Fast neutron spectrum + closed fuel cycle for efficient conversion of U
2. Coolant: Lead (and not LB eutectics), Tsc=480°C,
3. Pool-type reactor concept and supercritical water at secondary side (240 bars, 450°C)

➔ thermal efficiency above 40%.



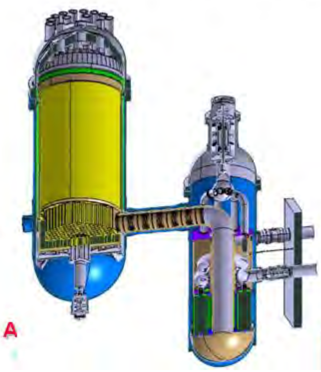


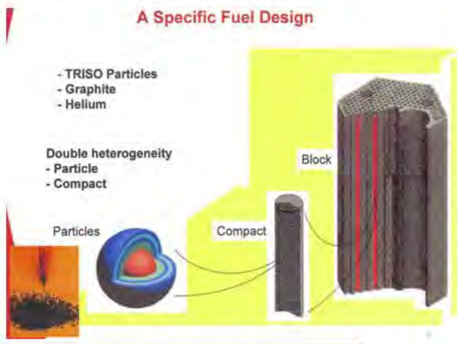
Review of the main features of ANTARES (VHTR)

ANTARES: 600MWth

1. thermal neutron spectrum (graphite as a moderator)
2. Coolant: helium, Tsc>900°C.
3. full passive decay heat removal
4. Prismatic bloc type core (UO2 fuel, TRISO coated particles).
5. Indirect Brayton cycle (i.e. gas turbine in the secondary circuit)

➔ thermal efficiency (45-50%).





Main features of the coolant circuits (1/3)

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Primary				
Nature of coolant	Sodium	Helium	Lead	Helium
Mass or volume of the fluid	~2500 m^3	8000 kg	6.3*10^6 kg	
Inertia (fluid+structure)	5 MJ/K			
Operating pressure (MPa)	0.1 (cover gas pressure)	7.0	0.1	6.0
Core inlet temperature (°C)	395	400	400	400
Mean core outlet temperature (°C)	545	850	480	850
Hottest core outlet temperature (°C)	570	900	500	

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Main features of the coolant circuits (2/3)

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Secondary				
Nature of coolant	Sodium	He/N2 (80/20 %vol) *Alternative He/Ar*	Water-superheated steam	He/N2 (80/20 %vol)
Mass or volume of the fluid	6 loops x ~200 m^3 (at 180°C)	6000 kg	25000 kg	
Operating pressure (MPa)	0.1 (cover gas pressure)	6.5	18.0	5.5
maximum temperature (°C)	525	820	450	800
Tertiary circuit (if relevant)				
Nature of coolant	Water	Water / steam	n/a	
Mass or volume of the fluid	n/a	n/a	n/a	
Operating pressure (MPa)	18.5	15.0	n/a	
Maximum temperature (°C)	490	535	n/a	550/250

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Main features of the coolant circuits (3/3)

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Decay Heat Removal secondary				
Nature of coolant	DRC : sodium DHRTV : water SGOSDHR : air	Water	Water	Water/air
Mass or volume of the fluid	6 loops ~15 m³ / loop (for DRC)		3400 Kg (cold water storage)	
Operating pressure (MPa)	0.1	1.0	0.1	
DHR Ultimate heat sink	DRC : air DHRTV : water SGOSDHR : air	Water	water	Water/air
Passive / active DHR system	DRC : FC+NC DHRTV : NC SGOSDHR : FC	Forced Convection + NC in He / pressurized water	NC	

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Next stage = Focussing on degradation phenomena

→Description of main degradation phenomena

→Classification of degradation phenomena

→Collection of parameters connected with degradation

→Looking for uncertainties

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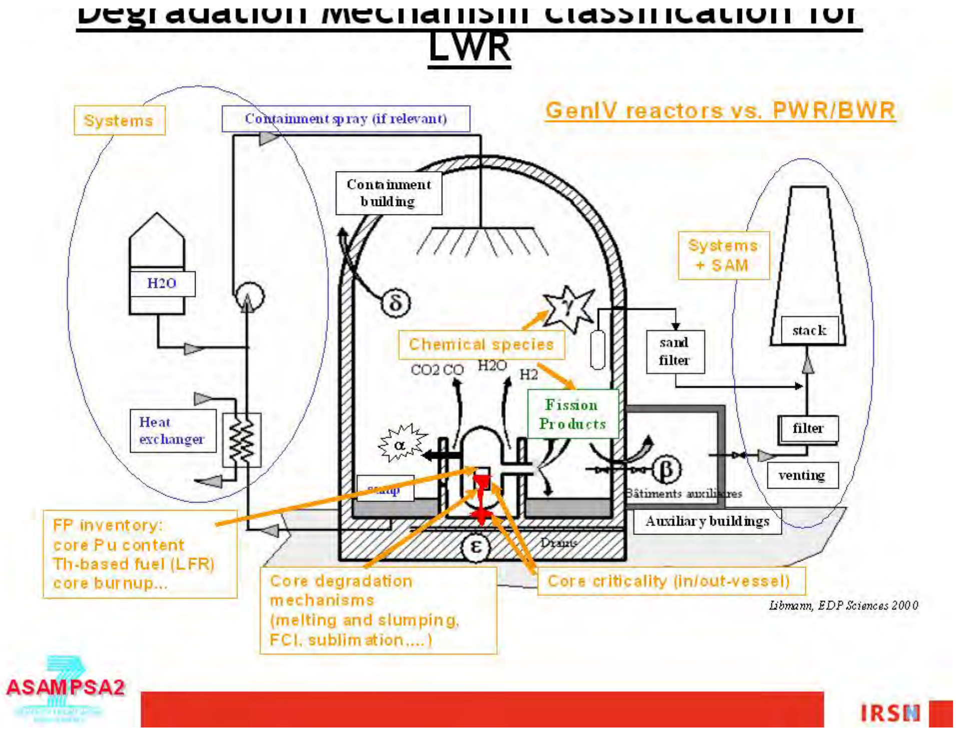
LWR
Systems
Containment spray (if relevant)
GenIV reactors vs. PWR/BWR
Containment building
H2O
Systems + SAM
stack
Chemical species
CO2 CO
H2O
H2
sand filter
Fission Products
Heat exchanger
filter
venting
Bâtiments auxiliaires
Auxiliary buildings
FP inventory:
core Pu content
Th-based fuel (LFR)
core burnup...
Drains
Core degradation mechanisms
(melting and slumping, FCI, sublimation....)
Core criticality (in/out-vessel)
Libmann, EDP Sciences 2000
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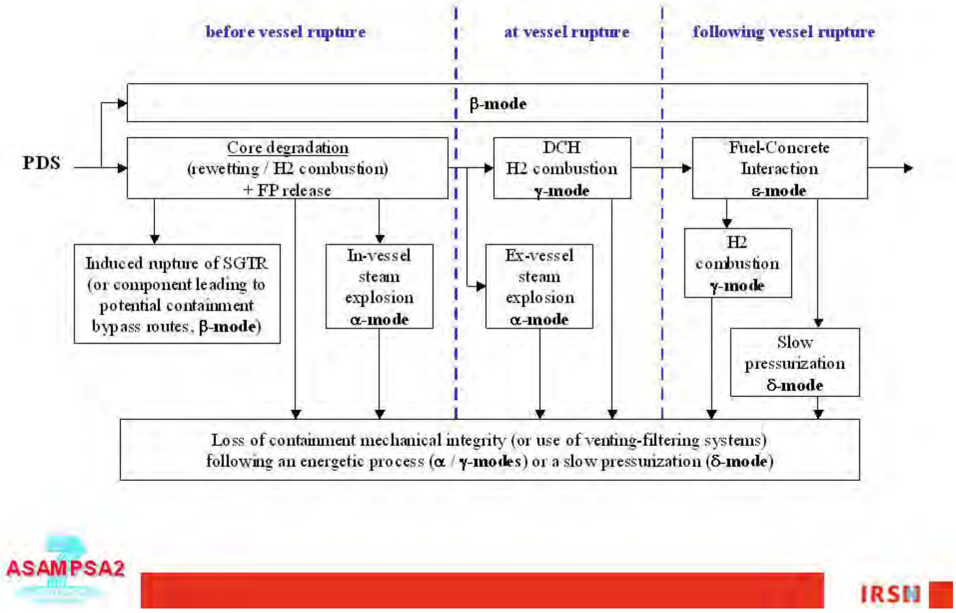
before vessel rupture
at vessel rupture
following vessel rupture
β-mode
PDS
Core degradation
(rewetting / H2 combustion)
+ FP release
DCH
H2 combustion
γ-mode
Fuel-Concrete
Interaction
ε-mode
H2
combustion
γ-mode
Induced rupture of SGTR
(or component leading to
potential containment
bypass routes, β-mode)
In-vessel
steam
explosion
α-mode
Ex-vessel
steam
explosion
α-mode
Slow
pressurization
δ-mode
Loss of containment mechanical integrity (or use of venting-filtering systems)
following an energetic process (α / γ-modes) or a slow pressurization (δ-mode)
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Compliance and potential transposition of containment degradation modes

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
α-mode	Mechanical energy release in case of CDA, recriticality in case of core degradation, FCI	Energy release due to recriticality in case of core degradation	Steam explosion due to Steam Generator Tube Rupture	Dust explosion (or δ-mode ?)
β-mode	IHX, DHX tube rupture Secondary containment failure	Similar to LWRs, IS-LOCA (IHX, DHX tube rupture) combined with the containment isolation failure, HSS failure	Steam Generator Tube Rupture, Containment Isolation failure	Identical to LWRs because of the thermal loading of the IHX (failure of the isolation valves)
γ-mode	Na fire	H2 / CO emission (following steam ingress in the "carbide" core)	H2, CO/CO2 emission (following MCCI)	H2 / CO emission (following steam ingress in the graphite moderated core)
δ-mode	Na vaporization (in case of LDHR)	H2 of CO slow deflagration, failure of the guard vessel → pressurization of the Containment	Over pressurization in containment building	Dust explosion (or α-mode ?)
ε-mode	Corium / Concrete Interactions	FCI	MCCI	Not relevant

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Collection of key parameters related to core degradation mechanisms for each concepts

- **Material inventories**
- **Material interaction at high temperature**
  - Core material behavior
  - Fuel coolant interaction
  - Interaction of core material with foreign fluids
- **Interaction between the primary fluids and others fluids**
  - SFR: Na interaction with water or air
  - LFR: No hazard
  - GFR and VHTR: not relevant
- **Key parameters for core disruptive accident**
- **Core criticality concerns**
  - Control rod(s) withdrawal
  - Coolant voiding
- **VHTR case (no gas change, low FP releases, …)**

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Identified needs of knowledge

- Sodium Fast Reactor (SFR)
  - Large efforts (codes, exp.) in the past and continuing
  - For CDA secondary phase, shortcomings identified
- Gas-cooled Fast Reactor (GFR)
  - Substantial lack of experiments and analysis tools
- Lead -cooled Fast reactor (LFR)
  - Known Limitations of the analysis tools: large uncertainties
- Very High Temperature Reactor (VHTR)
  - PIRT performed in US and available

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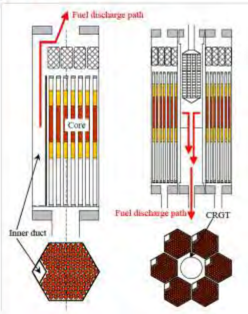
Specific provisions for prevention and mitigation of SA

- Supplementary shutdown system
  - SFR: passive system
  - GFR: as for SFR redundant monitoring system (CR withdrawal)
  - LFR: 3 independent diverse systems
- In-core corium spreading system
  - SFR: Faidus, CRGT systems
  - LFR: Reduced risk (large pin pitch)
- Core catcher
  - EFR: in-vessel core catcher
  - GFR: ex-vessel ceramic core catcher
- Specific Containment safety features
  - SFR: liner above the roof to prevent Na-Concrete interaction
  - GFR: Vented and filtered containment + pressure relief system
  - LFR: provisions against SGTR (steam explosion)
  - VHTR: primary coolant clean-up system + RCCS (heat sink)



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Important parameters for L2PSA source term evaluation

1. Inventory of radioactive materials in the fissile region (at EOL)
2. In-vessel radionuclide release and transport mechanisms
3. Retention and deposition of fission products inside RCS;
4. Chemical species (e.g. organic or non-organic iodine...)
   a. Iodine and caesium chemistry (Affinity of these isotopes with coolants involved);
   b. Chemistry of other isotopes (Te, Sr,...): knowledge regarding phenomenological trends in presence of helium, lead or sodium;
5. Activation and corrosion products;
6. Ex-vessel radionuclide release and transport (related to containment type)
7. Aerosols behaviour inside the containment;
   a. Deposition and re-suspension of aerosols mechanisms;
   b. Effect of energetic phenomena on in-containment fission product behaviour;
   c. Activation and corrosion products of concrete surrounding the core vessel (if any) and air of its cooling system;
8. Radionuclide release outside the containment (i.e. Source Term);
9. Tritium;
10. Potential for FPs scrubbing;
11. Additional barriers or structures (e.g. close containment for GFR is not considered as confinement barrier but could lead to a potential of FPs retention)

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Treatment of hazards

1. Internal missile,
2. Jet effects
3. Pipe whip
4. Leakage/LOCA
5. Internal flooding
6. Dropped loads
7. Internal fire
8. Asphyxiate and toxic gas release (dust)
9. Gas/chemical explosion
10. Hot and cold gas release
11. Sound, vibrations
12. Graphite dust explosion
13. Aircraft impact
14. Vehicular impact
15. Sabotage
16. Transport, industrial activities (fire, explosions, missiles, toxic & asphyxiant gases, corrosive gases)
17. Electromagnetic interference (EMI).

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Treatment of hazards: case of V-HTR

Hazard	Risk as compared to LWR
Internal missile Jet effects Pipe whip Leakage/LOCA Internal flooding	Less
Dropped loads Internal fire Asphyxiate and toxic gas release (dust) Gas/chemical explosion	Same
Hot and cold gas release Sound, vibrations	Higher
Core / fuel chemical reactions Graphite dust explosion	Higher, HTR specific hazard
Aircraft impact Vehicular impact Sabotage Transport, industrial activities (fire, explosions, missiles, toxic & asphyxiant gases, corrosive gases) Interference with water intake and Ultimate Heat Sink EMC	Same

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Specifics related to shutdown or refuelling states

- SFR: manufacture or loading errors: detected by instrumentation
- GFR: handling under pressure for fuel cooling
- VHTR with pebble bed: continuous refuelling

Existing L2PSA for Gen IV

- Sodium Fast Reactor (SFR)
  - SNR-300
  - US PRISM
  - JSFR
- Gas-cooled Fast Reactor (GFR): no
- Lead -cooled Fast reactor (LFR):No
- Very High Temperature Reactor (V-HTR))
  - HTGR of General Atomics
  - HTR-1160 (Germany)
  - MHTGR (US)
  - PBMR (South Africa)

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List of codes given for describing:

1. Core damage progression (initiating phase, transition phase, core disruption for FRs; slow core heat-up for VHTR)
2. Failure modes of the RCS (dynamic thermal-mechanical calculations and tools / assessment of conditional probabilities regarding the missile emission potentially challenging the containment integrity)
3. Failure modes of the containment (dynamic thermal-mechanical calculations and tool)
4. MCCI
5. Source term assessment (FPs release from the core, transportation & deposition in RCS, in retention tanks and transfer to the environment).

Compliance with LWR phenomena and systems (1=HC-5=LC)

Sections	Subsections	Items	SFR	GFR	LFR	V-HTR
Quantification of physical phenomena and containment loading	Definition and calculation of representative thermal-hydraulics sequences for each PDS		5	5		3
	In-vessel core degradation	a - Core degradation	5	4	3	5
		b - Induced-RCS rupture including induced-SGTR	5	3	4	5
		c - Hydrogen production	5	5	5	3
		d - Restoration of core-cooling	5	5	4	5
		e - Vessel cooling from outside	5	5	3	5
		d - Consequences of in-vessel water injection (coolability, hydrogen production, RCS pressurization ...)	5	5	5	5
		e - Containment atmosphere composition (recombiners/igniter effect) and containment pressurization	5	5	5	3
		f - Containment venting	2	2	1	3
		g - Hydrogen distribution/combustion	5			4
		h - Corium criticality	5	5	5	4
		i - In-vessel steam explosi on and consequences (leak in the RCS, vessel rupture, containment rupture)	3	5	5	4
		j - Vessel rupture (delay, break size ...)	3	3	3	5
	Vessel rupture phase	a - Direct Containment Heating, including H2 combustion and vessel uplift	5	5	2	5
		b - Ex-vessel steam explosion	5	5	2	5
		c - Corium criticality	5	5	2	5
	Ex-vessel phase (MCCI)	a - Corium coolability	3	3	2	5
		b - Basemat lateral and axial erosion	5	1	1	5
		c - Impact of water injection	5	5	2	5
		d - Production of steam and incondensable gases	4	5	2	5
		e - H2/CO combustion	5	3	1	3
		f - Evolution of containment atmosphere composition and long term pressurization	2	2	1	3
		g - Containment venting	2	2	1	3
		i - pool scrubbing		5		5
		h - Melt propagation into ducts and channels	5			5
Containment performance (tightness)	Initial containment performance (pre-existing leakage)		1	1	1	
	Failure of the isolation system		1	1	1	
	Evaluation of containment performance in severe accident conditions	a - Quasi- static loading / dynamic loading – Structural response, structural analyses, fragility curve (leak or	4	1	1	
		b - Specific issues : example the impact of a steam explosion in the vessel pit on the overall structure	5		2	
		c - drywell/supression pool performance			5	
	Containment penetrations performance (tightness) in severe accident conditions		2	2	1	
	Identification of specific containment bypass ways (example: case of existing pipes in the plant foundations, cavity door failure for VVER)		2		1	
Systems behaviour in severe accident conditions	Sump recirculation, CHRS, Spray system		5	5	5	5
	RCS safety valves		3	2	5	3
	Steam Generator		5	5	1	3
	Instrumentation		5	3	1	2
	Pedestall cavity flooding systems		5	5	5	5
	H2 recombiners/igniters		5		2	5
	Core catcher		3	3	5	5
	Reliability of passive systems		1	1	1	2
Source term assessment	Definition of release categories	a – identification of key parameters for source term assessment	2	2	1	3
		b – example of release categories		1	1	2
		c - screening frequency		1	1	2
	Group of fission products		3	1	1	2
	Source term assessment by integral codes		2	2	1	2
	Source term assessment by dedicated (fast-running) source term models				1	2
	Radiological consequences		1	1	1	2

Level 2 PSA Structure

- Common adopted approach for L2 (performed after L1PSA)
  - Definition of the initial conditions by binning of L1PSA end states into Plant Damage States (PDS);
  - Development, construction and quantification of event trees: Containment Event Trees (CET, i.e. small event trees) or Accident Progression Event Trees (APET, i.e. large event trees);
  - Definition of source term categories or release categories;
  - Binning of containment states related to specific containment failure modes
- L1-L2 interface parameters
  - The 2nd barrier integrity (e.g. intact RCS vs. LOCA) Primary pressure during core meltdown;
  - The core power (i.e. time after IE) at core damage onset;
  - The status of safety systems linked to the RCS;
  - The availability of power supplies (external, internal, AC and DC);
  - The integrity of the containment (intact/failed through isolation failure, bypass through heat exchangers or IS-LOCA);
  - The availability of containment protection systems (if any).

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Level 2 PSA Structure: APEC/CET

- GEN IV arising questions
  - The general structure of APET/CET assuming that time phases (i.e. before reactor vessel failure, at failure, and after) could be consistent if the vessel rupture “notion” (i.e. bottom head rupture in LWRs) is extended to the loss of integrity of the second barrier (e.g. cross-duct rupture for GFR and VHTR, rupture of the roof in SFR and LFR); then no major difference regarding time phases is expected.
  - The most important phenomena that should be considered for APET/CET building.
  - Mission time of system and definition of the reactor final state.
  - Common Cause Failures (containment penetrations and isolating devices).
  - Use of cut-off frequency (if any), compliance with the cut-off frequency of PWR/BWR.
  - Extend of feedback regarding the Generation IV reactors (data, level 2 PSA technical feedback...).

  Tentative ET for GEN IV reactors
  - SFR: small event tree (only the action of isolation of containment, the reliability of the corium coolability, and the criticity risk $\neq$ LWRs will be modelled)
  - LFR: No CDA; SGTR may lead to α (steam explosion), γ (H2) and ε (MCCI) modes
  - VHTR: example of event tree for the typical accident (loss of He flow) already available

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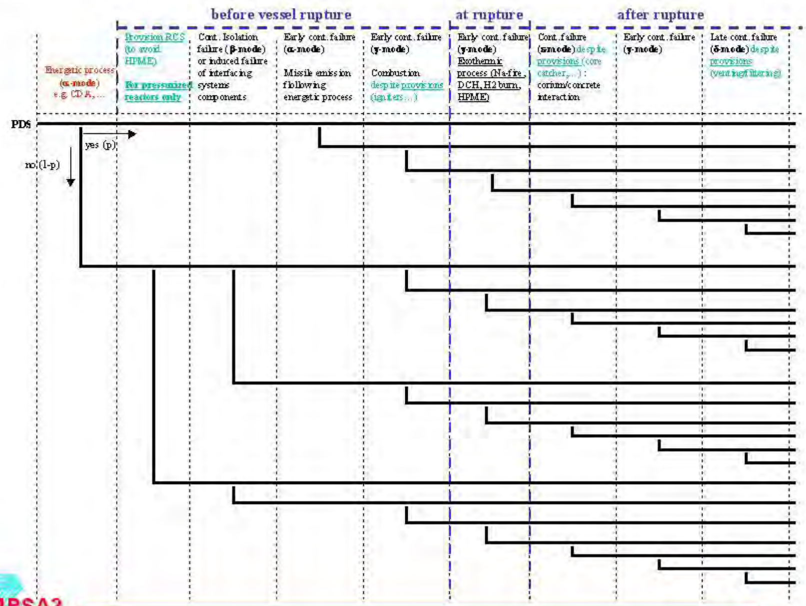
Generic Event Tree related to containment degradation modes



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Screening compliance (continued)

- Human reliability asssessment
  - For GEN IV no mitigation measures and procedures defined
- Quantification of physical phenomena and uncertainties
  - Sources of uncertainties:
    - Parameters (data) uncertainties;
    - Model uncertainties
    - Model Completeness
- Passive safety systems
  - Passive system used in a greater extend than with LWRs
  - Failure assessment of passive system is complex (will a natural circulation establish or not ? etc.)
- Calculation tools and uncertainties
- Role and extend of expert judgment
  - More important / LWRs due to limited experience

Compliance level with GenII/III L2PSA model building (from 1 to 5 : 1 = potentially highly compliant, 5 = not compliant)

Sections	Subsections	Items	SFR	GFR	LFR	V-HTR
L1-L2 PSA interface				2	1	
Accident Progression Event Tree (APET)			5	2	1	
Release Categories and result presentation			2	2	1	
L1-L2 interface			1	1	1	
Human Factors	Examples of human actions (from severe accident management guide, support of crisis organization, systems recovery...)		3	3	2	3
	Methods for the human factor quantification		4	3	1	3
APET-CET			2	2	1	3
List of plant data that should be available for the L2 PSA			4	5	1	
Severe accidents codes			5	5	1	
Event trees codes			1	1	1	1

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Conclusions (1/3)

- **GEN IV technology roadmap:**
  - "the design detail must allow use of simplified Probabilistic Risk Assessment (PRA) to identify design basis accidents and transients as well as the highly hypothetical sequences. The detail should be sufficient to identify and rank phenomena of importance to transient response and to specify experimental information required to validate transient models". In addition, it was recalled that "Generation IV nuclear energy systems will eliminate the need for offsite emergency response"
- **Issues to be addressed (end-users requirements)**
  - the determination of LERF (Large Early Release Frequency/LRF;
  - the identification of main containment failure modes and the related assessment of releases;
  - the plant vulnerabilities insights in Accident Progression and assessment of containment performance;
  - the insights to plant specific risk reduction option;
  - and finally, the insights to Severe Accident Management Guidelines (SAMG)

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Conclusions (2/3)

- **L2PSA may be performed at the early stage of design will give insights on:**
  - The identification of major containment failure modes and on how severe accident progress;
  - A rough estimation of the quantities of released radioactive material to the environment for different accident sequences;
  - The identification of particular important phenomena and processes, and especially those who are of importance for containment performance (i.e. the last barrier in order to avoid massive and long-term population displacement following an accident);
  - A useful help for the prioritization of R&D activities
- **Main differences of GEN IV reactors compared to LWRs**
  - Neutron Fast spectrum: Core Disruptive Accidents
  - Pu and Minor Actinide Inventories
  - Fire concerns (Na, graphite): have an impact on FP and chemical releases

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Conclusions (3/3) and perspectives

- **Too Early to perform study of compliance of LWR L2PSA guidelines to Gen IV reactors:**
  - GEN IV reactor still under design
  - No Emergency Operating Procedures (EOP) and Severe Accident Management Guidelines
  - Needs of probabilistics models
- **but**
  - Core damage prevention can be done (by design)

And later EOP will be defined to mitigate the consequences of severe accidents

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Thank you for your attention

Thanks to the ASAMPSA2 project partners and specially to F. Serre (CEA) for his help preparing this presentation.

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