ASAMPSA2 SEVENTH FRAMEWORK PROGRAMME	Advanced Safety Assessment Methodologies: Level 2 PSA	EURATOM

"NUCLEAR FISSION"

Safety of Existing Nuclear Installations

Contract 211594

ASAMPSA2
BEST-PRACTICES GUIDELINES
FOR L2 PSA DEVELOPMENT AND APPLICATIONS

Volume 3 - Extension to Gen IV reactors

Technical report ASAMPSA2/WP4/D3.3/2013-35

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Summary: The main objective assigned to the Work Package 4 (WP4) of the “ASAMPSA2” project (EC 7th FPRD) consist in the verification of the potential compliance of L2PSA guidelines based on PWR/BWR reactors (which are specific tasks of WP2 and WP3) with Generation IV representative concepts. Therefore, in order to exhibit potential discrepancies between LWRs and new reactor types, the following work was based on the up-to-date designs of:

- The European Fast Reactor (EFR) which will be considered as prototypical of a pool-type Sodium-cooled Fast Reactor (SFR);
- The ELSY design for the Lead-cooled Fast Reactor (LFR) technology;
- The ANTARES project which could be representative of a Very-High Temperature Reactor (VHTR);
- The CEA 2400 MWth Gas-cooled Fast Reactor (GFR).

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MODIFICATIONS OF THE DOCUMENT

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LIST OF DIFFUSION

Name	Organization
All Partners of ASAMPSA2 project. Specific list of organizations concerned by L2PSA development and applications for NPP. The guidelines is a public document	In particular, organizations in countries members of OECD/NEA/CSNI and observers. www.asampsa2.eu www.irsn.fr

ASAMPSA2 PROJECT SUMMARY

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

Specific relationships with communities in charge of nuclear reactor safety (utilities, safety authorities, vendors, and research or services companies) have been established in order to define the current needs in terms of guidelines for L2PSA development and application. An international workshop was organised in Hamburg, with the support of VATTENFALL, in November 2008.

The L2PSA experts from ASAMPSA2 project partners have proposed some guidance for the development and application of L2PSA based on their experience, open literature, and on information available from international cooperation (EC Severe Accident network of Excellence – SARNET, IAEA standards, OECD-NEA publications and workshop).
At the end of the ASAMPSA2 project, the guidelines have been submitted to an international external review open to European nuclear stakeholders and organizations associated to the OECD-CSNI working groups on risk and accident management. A second international workshop was organized in Espoo, in Finland, hosted by FORTUM, from 7 to 9th of March 2011 to discuss the conclusions of the external review. This final step for the ASAMPSA2 project occurred just before the Fukushima Daïchi disaster (11th of March 2011). All lessons from the Fukushima accident, in a severe accident risk analysis perspective, could not be developed in detail in this version of the ASAMPSA2 guideline.

The first version of the guidelines includes 3 volumes:

- *Volume 1 - General considerations on L2PSA.*
- *Volume 2 - Technical recommendations for Gen II and III reactors.*
- *Volume 3 - Specific considerations for future reactors (Gen IV).*

The recommendations formulated in these 3 volumes are intended to support L2PSA developers in achieving high quality studies and focussing time and resources on the factors that are most important for safety.

L2 PSA reviewers are another target group that will benefit from the state-of-the art information provided.

This first version of the guidelines is more a set of acceptable existing solutions to perform a L2PSA than a precise step-by-step procedure to perform a L2PSA. One important quality of this document is that is has been judged acceptable by organizations having different responsibilities in the nuclear safety activities (utilities, safety authorities or associated TSO, research organization, designer, nuclear service company …).

Hopefully it can contribute to the harmonization of the quality of risk assessments.

Most activities related to the development of the guidelines were performed before the Fukushima Daïchi accident. Some complementary guidance for the assessment of severe accident risks induced by extreme events will be developed in a follow-up European project (ASAMPSA_E).

ASAMPSA2 PARTNERS

The following table provides the list of the 21 ASAMPSA2 partners involved in the development of these guidelines.

1	Institute for Radiological Protection and Nuclear Safety	IRSN	France
2	Gesellschaft für Anlagen- und Reaktorsicherheit mbH	GRS	Germany
3	NUBIKI Nuclear Safety Research institute Ltd.	NUBIKI	Hungary
4	TRACTEBEL ENGINEERING S.A	TRACTEBEL	Belgium
5	IBERDROLA Ingeniería y Construcción S.A.U	IBERINCO	Spain
6	Nuclear Research Institute Rez pl	UJV	Czech
7	Technical Research Centre of Finland	VTT	Finland
8	ENEA - Ricerca sul Sistema Elettrico SpA	ERSE SpA	Italy
9	AREVA NP GmbH	AREVA NP GmbH	Germany
10	AMEC NNC Limited	AMEC NNC	United-Kingdom
11	Commissariat à l'Energie Atomique	CEA	France
12	Forsmark Kraftgrupp AB	FKA	Sweden
13	Cazzoli consulting	CCA	Switzerland
14	National Agency for New Technologies, Energy and the Environment	ENEA	Italy
15	Nuclear Research and consultancy Group	NRG	Nederland
16	VGB PowerTech e.V.	VGB	Germany
17	Paul Scherrer Institut	PSI	Switzerland
18	Fortum Nuclear Services Ltd	FORTUM	Finland
19	Radiation and Nuclear Safety Authority	STUK	Finland
20	AREVA NP SAS France	AREVA NP SAS	France
21	SCANDPOWER AB	SCANDPOWER	Sweden

ASAMPSA2 CONCEPT AND PROJECT OBJECTIVE(S)

Members of the European community who are responsible for fission reactor safety (i.e. plant operators, plant designers, Technical Safety Organisations (TSO), and Safety Authorities) have repeatedly expressed a need to develop best practice guidelines for the L2PSA methodology which would have the aim of both efficiently fulfilling the requirements of safety authorities, and also promoting harmonisation of practices in European countries so that results from L2PSAs can be used with greater confidence..

Existing guidelines, like those developed by the IAEA, propose a general stepwise procedural methodology, mainly based on US NUREG 1150 and high level requirements (for example on assessment of uncertainties). While it is clear that such a framework is necessary, comparisons of existing L2PSA which have been performed and discussed in (6th EC FP) SARNET L2PSA work packages, have shown that the detailed criteria and methodologies of current L2PSAs strongly differ from each other in some respects. In Europe the integration of probabilistic findings and insights into the overall safety assessment of Nuclear Power Plants (NPPs) is currently understood and implemented quite differently.

Within this general context, the project objectives were not to share L2PSA tools and resources among the partners, but to highlight common best practices, develop the appropriate scope and criteria for different L2PSA applications, and to promote optimal use of the available resources. Such a commonly used assessment framework should support a harmonised view on nuclear safety, and help formalise the role of Probabilistic Safety Assessment.

A common assessment framework requires that some underlying issues are clearly understood and well developed. Some important issues are:

- *the PSA tool should be fit for purpose in terms of the quality of models and input data;*
- *the scope should be appropriate to the life stage (e.g. preliminary safety report, pre-operational safety report, living PSA) and plant states (e.g. full power, shutdown, maintenance) considered;*
- *the objectives, assessment criteria, and presentation of results should facilitate the regulatory decision making process.*

The main feature of this coordination action was to bring together the different stakeholders (plant operators, plant designers, TSO, Safety Authorities, PSA developers), irrespective of their role in safety demonstration and analysis. This variety of skills should promote a common definition of the different types of L2PSA and so help develop common views.

The aim of the coordination action is to build a consensus on the L2PSA scope and on detailed methods deemed to be acceptable according to different potential applications. In any methodology, especially one developed from a wide range of contributing perspectives, there will be a range of outcomes that are considered acceptable. To represent this range, the project has initially considered a 'limited-scope' and a 'full-scope' methodology, based *on what is currently technically achievable in the performance of a L2PSA. In this respect it should be noted that what is technically achievable may not be cost effective, but for the purpose of this project it was taken to represent the upper bound of what may be considered 'reasonable'.*

- *'Limited-scope' methodology*

  A limited description of the main reactor systems, associated with standard data on the reactor materials, severe accident phenomenology and human actions reliability will lead to a simplified L2PSA. This 'limited-scope' PSA would include some indication of the main accident sequences that contribute to the risk of atmospheric releases due to a severe accident. For example, 'limited-scope' methods could apply to a L2PSA performed with a limited number of top events in the event-tree and mainly dedicated to identification of accident sequences which contribute to the Large Early Release Frequency (LERF). However such a L2PSA can include very detailed and complex supporting studies for the quantification of these top events. Engineering judgement may also help in the quantification of the top events of a limited scope L2PSA but the justification of this engineering judgement is considered as a key issue.

- *'Full-scope' methodology*

  This method can utilise sophisticated methods that consider the full range of reactor initial states and possible accidents together with detailed physical phenomena modelling and uncertainty analysis. As a consequence these L2PSAs allow identification of the most sensible sequences with their probabilities of occurrence (annual frequencies) and associated fission product release to the environment. These L2PSAs also allow identification of the uncertainty range of the results, weak points in the reactor system and operation, and the accident phenomena which would need further assessment to improve the relevance of the results. In such a wide ranging L2PSA, the quantification of sequences leading to large early release is not the only objective.

In reality, most current L2PSAs are at an intermediate level between these two approaches. However this representation was recognised as a pragmatic way to organise the coordination action because it allowed discussion on both simple and elaborated methodologies. It should be assumed that the need for application of an advanced method is established from the results obtained by an earlier simplified study in regard to specific requirements of the national safety authorities.

Evidently the second type of approach is time consuming and supposes a qualified dedicated team. Some applications do not warrant this level of detail and additionally some small stakeholders (especially utilities) cannot afford this level of commitment. The scope should be appropriate to the application and life stage under consideration and the detailed methods should represent an acceptable balance between best practice and available resources. L2PSA results obtained using differing approaches or for differing scopes should not be directly compared.

When developing the guideline it was found by the partners <u>that a clear distinction between limited-scope and full-scope was very difficult to formalize</u> and it has been decided to present in the report, for each issue, some *recommendations that may refer to simplified or detailed approaches. The guidelines users are then supposed to develop themselves a strategy to build a consistent set of L2 PSA event trees and supporting analysis.*

ASAMPSA2 CONTRIBUTION TO THE COORDINATION OF HIGH QUALITY RESEARCH

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

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

The project had strong links with SARNET (Severe Accident Network of Excellence) and took into account all harmonization activities performed in other framework (IAEA,OECD-CSNI, WENRA, EUR, ANS, ASME …).

ASAMPSA2 COORDINATION MECHANISMS

The ASAMPSA2 organisation of the coordination action was based on three working groups:

- *<u>A transverse group of End-Users, </u>consisting of representatives of plant operators, plant designers,TSOs, safety authorities, R&D organisations, and L2PSA developers. The objectives of this group were:*
    - *to define and/or validate the initial needs for practical L2PSA guidelines for both 'limited' and 'full-scope' methods according to the different potential applications and specific End-User needs at the beginning of the coordinated action;*
    - *to provide a continuous oversight of the work of the Technical Group;*
    - *to verify that any proposed L2PSA guidelines can fulfil the initial and evolving End-User needs if required at the end of the coordination action;*
    - *to propose any follow-up actions in collaboration with the Technical Group.*

    This group was coordinated by PSI and includes representatives from IRSN, NUBIKI, TRACTEBEL, IBERINCO, VTT, AREVA GmbH, AMEC-NNC, FKA, CCA, VGB, FORTUM, and STUK.

- *<u>A technical Group in charge for the development of a L2PSA guideline for Gen II and III reactors </u>;*

    This group was coordinated by IRSN and includes representatives from GRS, NUBIKI, TRACTEBEL, IBERINCO, UJV, VTT, ERSE, AREVA GmbH, AMEC-NNC, FKA, CCA, FORTUM, AREVA-SAS, and SCANDPOWER.

- *<u>A technical Group in charge of the development of a L2PSA guideline (or prospective considerations) for some specific Gen IV reactors.</u>*

This group was coordinated by CEA and includes representatives from IRSN, AREVA GmbH, ERSE, ENEA, AMEC-NNC, NRG, and AREVA SAS.

The overall coordination of the ASAMPSA2 project was assumed by IRSN, including all administrative tasks and relationship with EC services.

SOME LIMITS OF THE ASAMPSA2 PROJECT

The number of issues that were addressed in the ASAMPSA2 project and discussed in the guidelines is very large. Nevertheless, these best practice guidelines have to be considered as a set of acceptable existing solutions to perform a L2PSA and not as a precise step-by-step procedure to perform a L2PSA.

The reader should be aware that issues such as external events, fire hazard, and ageing are not in the scope of this first version of the guideline, consistently with the Grant Agreement with the European Commission. For these topics, it was identified a needed for further harmonization activities during the End-Users final review. The Fukushima accident has then further highlighted their importance. Additional developments are expected to be included in any future updates of these guidelines.

VOLUME 3 CONTENT

1 INTRODUCTION

The main objective of a Level 2 Probabilistic Safety Assessment (L2PSA) is the depiction and the quantification, in terms of probabilities and consequences, of challenges to the containment and of its possible response. In addition, it provides an assessment of the potential Fission Products (FPs) release into the environment. According to Light Water Reactors knowledge and studies (computations, experiments…), the containment challenges are in particular related to:

- Slow over-pressurisation of the containment (in particular due to a slow deflagration of hydrogen produced during core degradation and/or to the non-condensable gases produced during Molten Core Concrete Interaction – MCCI –);
- Fast pressurisation of the containment building mainly due to risks of internal explosions (caused by species produced during the core degradation e.g. hydrogen (fast deflagration or detonation), or non condensable gases like He);
- Potential containment isolation failures or bypasses;
- Late containment failure through the base mat, following the corium spreading.

For the source term evaluation, the inventory of the released material, its physical and chemical forms, and information on the time, the duration and the location of releases are foreseen.

As expressed in the Generation IV technology roadmap, "maintaining and enhancing the safe and reliable operation is an essential priority in the development of next generation systems" ([1-1], page 2). "For the viability and safety evaluations of the selected reactors, the deterministic concept of defence in depth needs to be integrated with simplified probabilistic considerations (e.g. systems reliability and probabilistic targets) to provide metrics for acceptability and a basis for additional requirements, and to ensure a well-balanced design" ([1-1], page 69).

The main objective assigned to the Work Package 4 (WP4) of the "ASAMPSA2" project (EC 7th FPRD) could be expressed as a verification of the potential compliance of L2PSA guidelines based on PWR/BWR reactors (which are specific tasks of WP2 and WP3) with Generation IV representative concepts. Therefore, in order to exhibit potential discrepancies between LWRs and new reactor types, the following work was based on the up-to-date designs of:

- The European Fast Reactor (EFR) which will be considered as prototypical of a pool-type Sodium-cooled Fast Reactor (SFR);
- The ELSY design for the Lead-cooled Fast Reactor (LFR) technology;
- The ANTARES project which could be representative of a Very-High Temperature Reactor (VHTR);
- The CEA 2400 MWth Gas-cooled Fast Reactor (GFR).

Recall of the WP4 schedule and proposed tasks:

In the first phase, it was proposed to build the most exhaustive list of mechanisms and provisions involved in the selected Generation IV concepts. In order to help doing that work (i.e. verification of compliance with Light Water Reactor phenomena or mechanisms), and according to the respective reactor designs, it is first proceeded in a review of their main features that could potentially impact the containment response and the source term. Then, the work is followed by a depiction of:

- Specific degradation mechanisms (for the containment, if relevant compared to PWR/BWR ones, and also for core degradation on the basis of final states resulting from the L1-PSA);
- Potential specific provisions (if defined) to face with the containment degradation mechanisms (including the specific core degradation mechanisms).

At this point, it seems interesting to notice that for the selected Generation IV concepts:

- Three of them are characterized by a fast neutron spectrum, i.e. the Sodium-cooled Fast Reactor (SFR), the Gas-cooled Fast Reactor (GFR) and Lead-cooled Fast Reactor (LFR);
- Two of these reactors have a gaseous coolant (helium) in the Reactor Coolant System (RCS), i.e. the GFR and Very-High Temperature Reactor (VHTR) while the two last operate with a liquid metal (Na for SFR and Lead for LFR);

As a consequence, coolant phase change and resulting threshold effects can affect the two late concepts as regards to:

- Thermal exchanges in the core region (that are reduced by several orders of magnitude when the coolant is vaporized and depends of the pressure in this case);
- Neutronic behaviour through the coupling between the coolant density and the reactivity.

With the present knowledge of L2PSA models building for LWRs (PWRs and BWRs), potential similarities or discrepancies could be exhibited between LWRs and Generation IV concepts. A review is performed for L2PSA models, which were developed in the past for nuclear reactors involving other coolants than water. In addition, assuming that for LWRs an important effort was made during the past decades to build and to maintain validated calculation tools for the consequence assessment, it appears crucial to draw an inventory of existing calculation tools (past and present) to evaluate the Severe Accident (SA) consequences for the selected concepts. With regards to the limited experimental support that enables the development of these tools, it was tried to exhibit their potential limitations for applying them for L2PSA quantification, in terms of applicability easiness (e.g. CPU cost) and deepness of depiction of the main phenomena that could be encountered. These items are developed in the chapter 2 of this document.

A glossary has been added at the end of the document. Some parts which were more developed for the VHTR reactor (as being a far more developed concept) have been put in a special square intending to mean it

provides some interesting supplementary information but the reader can drop them if he is not especially interested in the VHTR subject.

A second phase of the WP4 work consisted in a review of the potential compliance with the guidelines issued from WP2&3 and related to L2PSA models for LWR. It composes the main part of the chapter 3 of this document. In addition, some methodological points are discussed.

For easy reading and understanding of this document, it is assumed that the reader has knowledge of L2PSA models developed for LWRs.

References of chapter 1

[1-1] A technology roadmap for Generation IV nuclear energy systems. Document referenced GIF-002-00 available on line at www.gen-4.org

2 REVIEW OF THE MAIN FEATURES OF THE GENERATION IV REPRESENTATIVE CONCEPTS

2.1 MAIN OBJECTIVES AND FEATURES OF THESE CONCEPTS

As defined by the GIF, the main objectives with the development of Generation IV concepts are recalled hereafter:

- The SFR, GFR and LFR systems (i.e. those featuring a fast neutron spectrum) are top-ranked in sustainability because of their closed fuel cycle and excellent potential for actinide management, including resource extension; they are also rated good in safety, economics, and proliferation resistance and physical protection;
- SFR is primarily envisioned in electricity production and actinide management; the SFR system is the nearest term actinide management system; based on the experience with oxide fuel,
- GFR is primarily envisaged in electricity production and actinide management, although it may also support hydrogen production; given its R&D needs for fuel, the GFR is estimated to be deployable by 2040;
- The LFR system is specifically designed for distributed generation of electricity and other energy products and for actinide management, given its R&D needs for fuel, materials, and corrosion control, the LFR system is estimated to be deployable by 2025;
- The VHTR addresses advanced concepts for helium-cooled, graphite moderated thermal neutron spectrum reactors with a core outlet temperature higher than 900°C. The ANTARES concept features a thermal power of 600 MWth and allows a full passive decay heat removal. The core envisioned is based on prismatic bloc type assemblies that contain UO_2 fuel TRISO coated particles. The electric power

conversion unit operates in an indirect Brayton-type cycle (i.e. gas turbine mixture in the secondary circuit).

Four representative concepts have been selected as a basis for this work to have some clear data to base the discussion on. Choice of the concepts was only based on the data availability for this WP participants. The four selected concepts, retained as reference for the work to be performed in the WP4 of ASAMPSA2, are (see Figure 1):

- The EFR (European Fast Reactor) concept for SFR; an European engineering consortium (EFRA) developed the EFR project on behalf of a European utility consortium (EFRUG) from 1988 to 1998, aiming at pooling the experience and resource of several European design and construction companies, R&D organisations and electrical utilities. The result of this common work was embodied in a preliminary design.
- CEA 2400MWth GFR (as designed at the end of 2007);
- ELSY project of LFR; a European lead-cooled fast reactor developed in the framework of EU FP6,
- ANTARES project, a commercial project designed by the AREVA company, for VHTR.

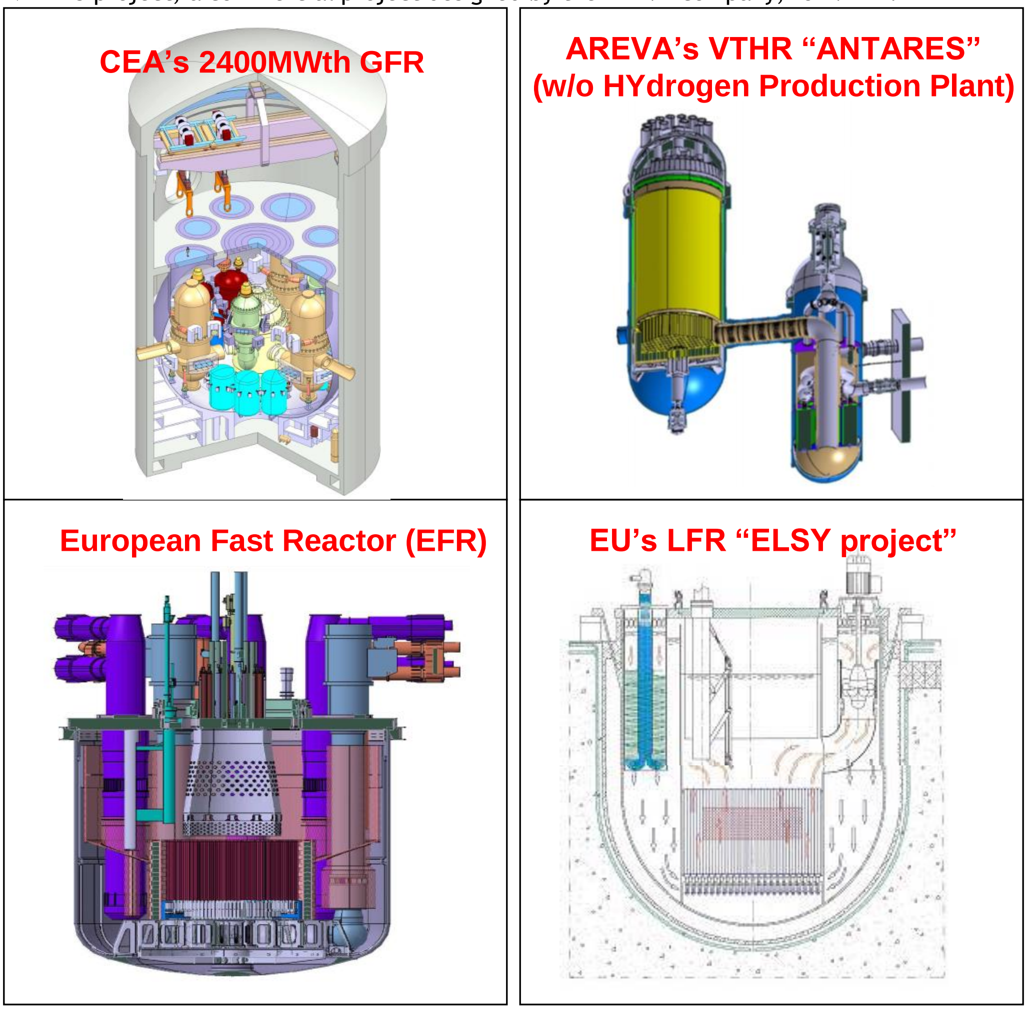

Figure 1: Overview of the four "representative" Generation IV concepts

SFR: The SFR features a fast spectrum reactor allowing an efficient management of high-level wastes and uranium resources. Using liquid sodium as the reactor primary coolant allows high power density with low coolant volume fraction. The primary system operates at near-atmospheric pressure with typical outlet temperatures ranging from 500 to 550°C. The EFR reactor, developed by a consortium of European utilities in the 90's, retains a 3600 MWth power, an intermediate cooling circuit (also filled with sodium) and a steam-water thermodynamic cycle.

GFR: The GFR features a fast neutron spectrum and a closed fuel cycle for efficient conversion of fertile material (uranium) and the management of Minor Actinides (MA). Actually, the reference version for the CEA is considering a 2400 MWth power and a combined thermodynamic cycle (Brayton-type gas turbine mixture in the secondary circuit and steam-water tertiary circuit); the helium-cooled system operating with a pressure of 70 bar and an outlet temperature of 850°C for high thermal efficiency (45-50%). Several fuel forms are being considered to ensure high FP retention capabilities: the reference core is actually based on plate-type fuel assemblies made of carbide fuel (with minor actinides) and ceramic clad elements.

LFR: The LFR features a fast neutron spectrum and use either lead or lead-bismuth eutectic as the liquid-metal coolant for the reactor. In the frame of the 6th FPRD, a consortium of organizations has been pursuing the development of the European Lead-cooled SYstem project (ELSY). The ELSY power plant is a pool-type reactor concept, sized at 600 MWe, and retains 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 240 bars, 450°C, and then providing a thermal efficiency above 40%.

VHTR: As an introduction, some of the VHTR features should be emphasised on. In contrast with the other GEN IV reactor concepts considered in the ASAMPSA2-WP4 project, the VHTR concept is a thermal reactor so that some safety issues specific to the three other projects are of no relevance for this concept. On the other hand, it shares some safety issues with the GFR concept as both are helium cooled gas reactors. It also shares some similarities with the SFR as both concepts are not so new so that it is possible to benefit a lot from former experiences. In fact, no less than five reactors have been operated in the past (1 in Great Britain, 2 in the U.S. and 2 in Germany). China is operating the HTR-10 and Japan the HTTR. The South African PBMR project has been cancelled.

The main conceptual difference between VHTR and former HTR lies in the core outlet temperature devised to be higher with VHTR; the V-HT stands for Very High Temperature as the objective is a core outlet temperature that is around 200 K higher than with the previous High Temperature Reactors. This high temperature issue is related to the use as energy source in the foreseen large scale production of hydrogen. This coupling will increase the economical interest. The most promising means of hydrogen production are the so-called hydrogen-cracking processes which require temperatures above 900°C to be efficient. However, such an industrial plant needs to be located in the vicinity of the nuclear plant as, contrary to electrical power, heat can not be transported efficiently on long distance. The drawbacks could be that the coupling with an industrial plant enhances specific hazards for the reactor with initiators as an hydrogen explosion in the hydrogen plant generating damages to the reactor containment or abnormal mass and/or heat exchanges through the coupling system. Nota : coupling between a HTR and an hydrogen production plant has been examined within the EUROPAIRS European project (FP7).

Gen IV project has scored VHTR concepts high from the safety point of view and it's true that they have been devised, ab initio, as inherently safe reactors. Modular HTR design is fundamentally ruled by the possibility to "exclude" severe fuel confinement damage, defined as degradation of the confinement capability of a large number of fuel particles. The justification that this accident is not plausible is expected to allow a considerable reduction in the requirements currently associated with the mitigation of severe accidents; in particular, it is expected that no pressure resistant containment is needed. Some VHTR safety features should be emphasised as they constitute major differences with LWR reactors:

- A certain degree of primary circuit radiological contamination will always exist due to some particles failure in operating conditions. This initial pollution, although limited, appears as a major contributor for source-terms;
- No-core melting is to be expected due to the combination of the high thermal inertia (large mass of core non fissile materials and large heat capacity, high core thermal conductivity), the low power density and the high graphite / fuel matrix melting temperature (large thermal margins);
- The negative temperature-reactivity coefficient for the entire fuel cycle and large fuel temperature margin (between operation and damages);
- The possibility to execute some safety tests on the reactor as has been done on the German AVR (stop of the blowers without control rod scram). Such tests are also planned on HTTR in Japan (project OECD HTTR LOFC).

One more point is worth mentioning: former and present VHTR cores may be built along two principles as the core may:

- Either be constituted of a pile made with hundreds of thousands of graphite pebbles (more or less the size of a tennis ball). The fuel particles are dispersed inside those pebbles. This is the pebble-bed concept;
- Or be constituted of hexagonal graphite assembly drilled with longitudinal holes filled with "compacts", a kind of long cylinder containing the fuel particles (especially for the ANTARES concept).

The ANTARES concept features a thermal power of 600 MWth that allows a full passive decay heat removal. The core envisioned is based on prismatic bloc type assemblies that contain UO_2 fuel TRISO coated particles. The electric power conversion unit operates in an indirect Brayton-type cycle (i.e. gas turbine mixture in the secondary circuit).

2.2 DESIGN FEATURES OF THE REPRESENTATIVE GENERATION IV REACTORS

Firstly, the main features regarding the core and the circuits of the representative Generation IV reactors are provided in the following paragraph. Then, it will be proceeded in a review of the specific degradation mechanisms to be accounted for in these various concepts, in such a manner that the final objective will be see the compliance with LWRs ones for L2PSA model building. In order to mitigate the consequences of a Severe Accident, several provisions of different natures and related to these specific risks are intended to be implemented in these Generation IV concepts. A paragraph is therefore consisting in a review of these provisions. Finally, for the source term assessment, some specific issues regarding the Fissions Products chemistry and phenomenological trends will be exhibited.

2.2.1 CORE FEATURES

The data have been provided by the different participants according to what was available or in open literature. For the ANTARES project, only a few data are allowed to be published which explains why a lot of cells remain empty.

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Power level (MWth)	3600	2400	1500	600
Core power density (MW/m^3)	300	91	160	≈ 6
Fissile height (m)	1	2.35	0.9	8
Core H/D ratio	0.25	0.62	0.2	
Nature of fuel	Oxides $(U,Pu)O_2$	Carbides (U,Pu)C + MAs	Oxides $(U,Pu)O_2$ at the first stage; MOX+MAs at the second	UO_2
TRU enrichment (%)	18 to 30% of Pu content	18.2 (Pu9 eq.)	15.7	≈20
Pu+MAs inventory (t/GWe)	6	11 (Efficiency 45%)	10.56	
Equiv. Pu9 mass BOC/EOC (kg)	6586 / 6610	8150 / 8284	4601/4625	
fuel / coolant vol. fraction (%)	36.01 / 32.94	22.4 / 40.0	32 / 58	

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Nature of cladding / coating	Stainless Steel	SiC/SiCf	9Cr-1Mo ferritic-martensitic (T91mod) steel/GESA	SiC
MAs inventory	< 5 %	From 1 to 2 %	1 %	
Core management (efpd)	residence time (fuel) : 1700	3 x 600	1460	
Neutron spectrum	Fast	Fast	Fast	Moderated (graphite)
Burn-up target	20 / 14% h.a. maximum / average (190/145 MWd/kg)	6.7 at% FIMA 100 GWd/t	100 GWd/t	
Delayed neutron fraction β_{eff} BOL/EOL (pcm)	350	355 / 342	340	460
Doppler constant BOL/EOL (pcm)	-900	-1283 / -837	-740	-2 pcm/K
Voiding BOL/EOL (pcm)	~ +2000 (6$)	+309 / +307 (0.85$)	+4040 (12$)	
Moderator constant BOL/EOL (pcm)				- 4 pcm/K

Table 1: Core features

For SFR (and EFR in particular), the core features a pin-type hexagonal arrangement with $(U,Pu)O_2$ fuel pellets surrounded by a Stainless-Steel cylindrical cladding. Figure 2 below illustrates the pin-fuel design and the core lay-out.

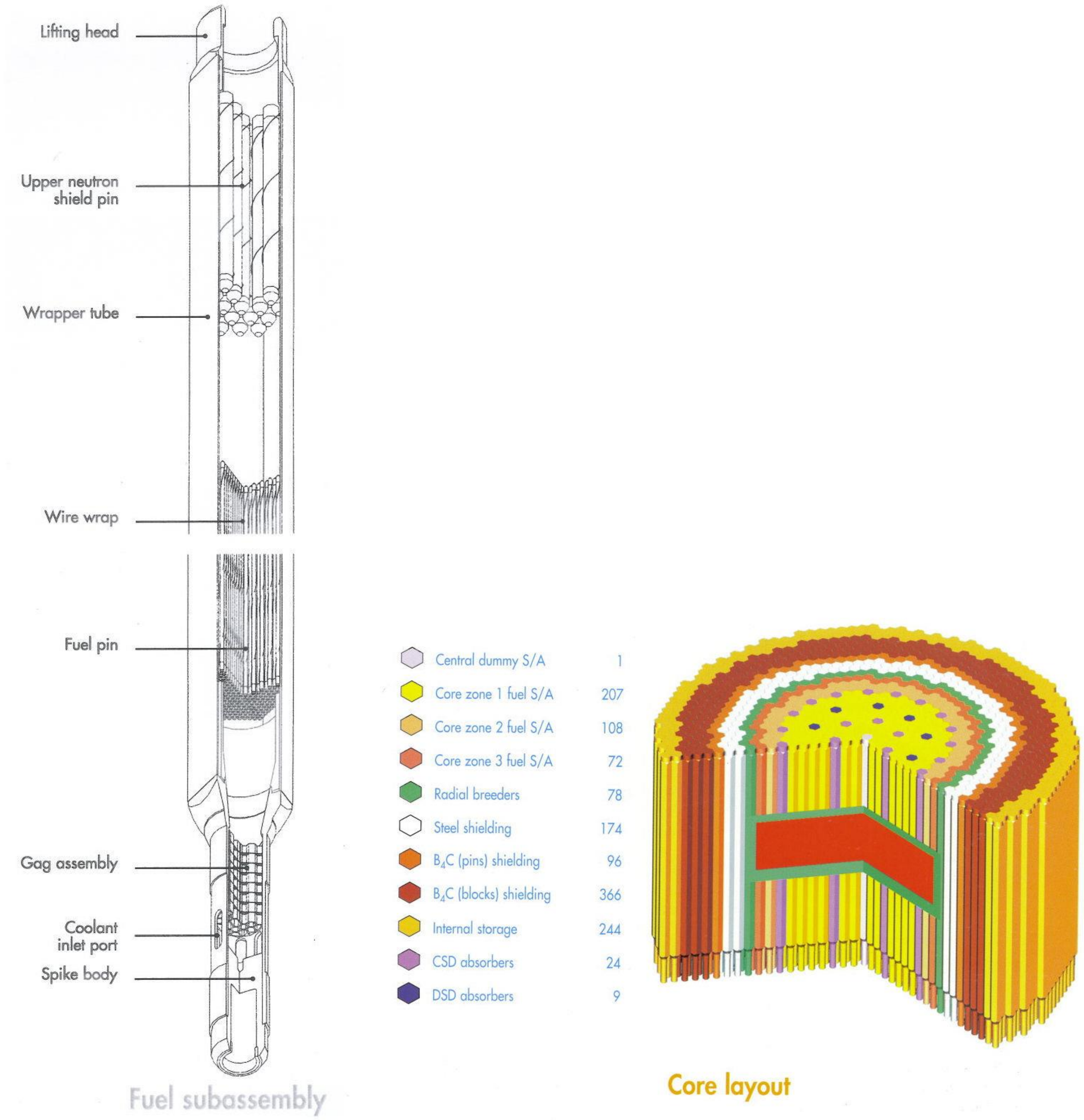

Figure 2: EFR pin fuel and core arrangement

The reference GFR core is a considered here is the plate-type core (as designed at the end of 2007; since this time the pin-type had been chosen as a reference) with ceramic cladding (SiC_f/SiC) with the following core operating temperatures: 400/850°C. In this concept, the fuel plates are made of a honeycomb structure (in grey in the following figure) containing cylindrical pellets made of mixed carbide (U,Pu)C (represented in red). The choice of this fuel design is essentially governed by the fact that this arrangement allows a micro-confinement in each hexagonal cell (expectation of lower radioactive releases in the case of an accidental scenario). These plates are arranged in a hexagonal SiC wrapper.

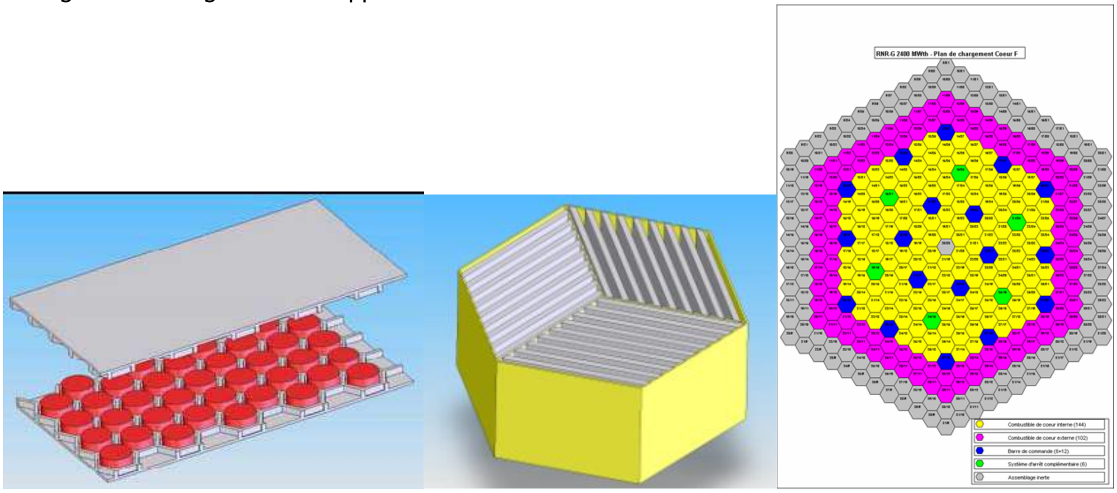

Figure 3: GFR plate-type fuel and core arrangement

For the ELSY core (LFR concept), the wrapper-less (“open”) square fuel assemblies is chosen as the reference design option, to be consistent with the available design of core support system and fuel handling components. A reference square fuel assembly (FA) consists of 428 fuel pins arranged in a 21 x 21 square lattice with a pitch of 13.9 mm. The fuel pins are supported along their lengths by six grid spacers, which maintain the lateral spacing between pins. Four structural tubes are located at the corners and a structural tube of square cross section (39 mm x 39 mm) is located at the centre of the fuel assembly replacing 9 fuel rods. Finger-type control rods moving inside of central structural tubes of FAs are also envisaged. The open square-lattice configuration, with fuel bundle details displayed in the figure below.

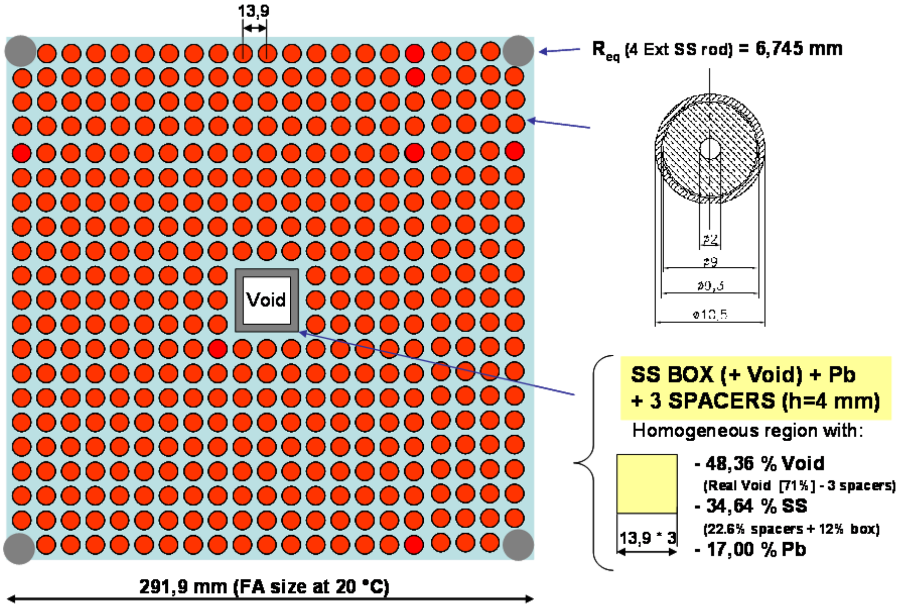

Figure 4: Schematic view of the square fuel rod lattice

The ANTARES core features fuel particles (TRISO) and fuel blocks. Fuel takes the form of a particle containing a core of fissile material (kernel of UO_2) surrounded by a buffer layer of carbon, a layer of pyrolitic carbon, a layer of silicon carbide and an outer layer of pyrolitic carbon (the overall diameter is about 1 mm – see the two figures below). The functions are differentiated:

- The inner porous layer of carbon serves as a buffer for the fission gases;
- The silicon carbide layer plays the role of a barrier to prevent the diffusion of solid fission products ;
- While the two dense pyrolytic carbon layers provide mechanical resistance to the internal pressure of the fission gases and help to retard the migration of solid fission products.

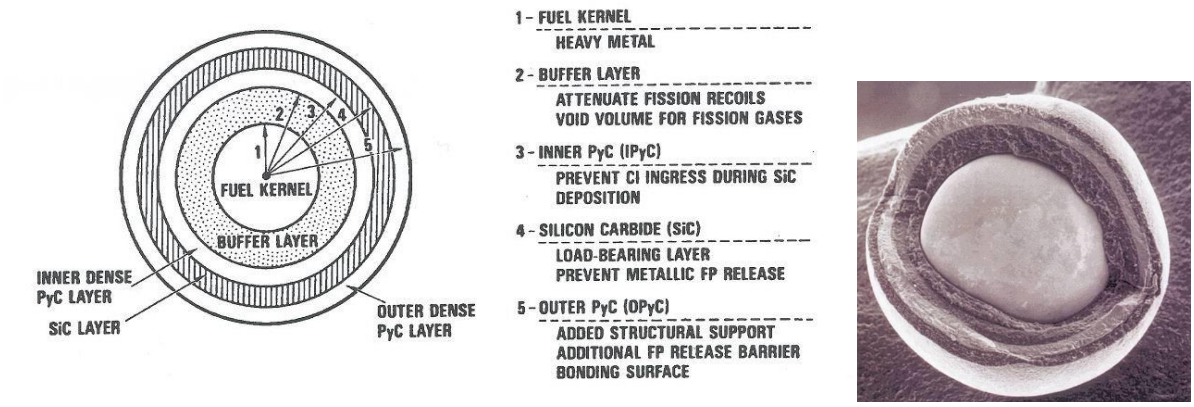

Figure 5: TRISO particles for VHTR

The fuel particles are agglomerated in a graphite matrix in the form of cylindrical rods called compacts. The compacts are inserted into prismatic graphite blocks, to constitute an organized core structure (see figure below).

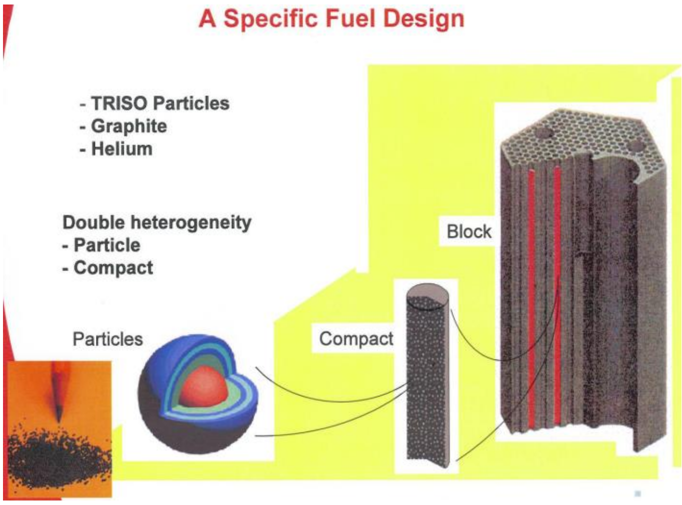

Figure 6: prismatic blocks for VHTR (ANTARES)

2.2.2 REACTOR COOLANT SYSTEM (RCS) AND CIRCUITS FOR DECAY HEAT REMOVAL (DHR)

SFR: The layout of normal and DHR systems as designed for EFR 98 is represented on the following figure.

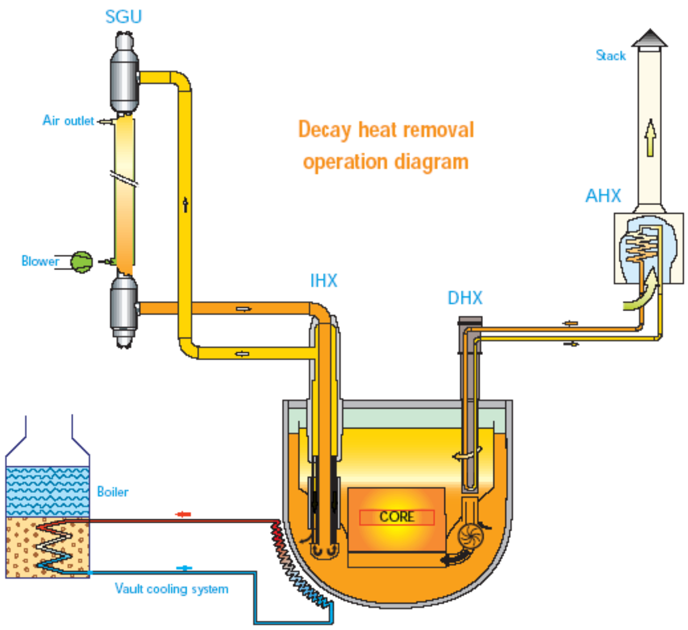

Figure 7: EFR 98 – DHR systems

As for all pool-concepts, the primary circuit is "immersed" inside the sodium pool. In normal function, fission heat is transferred to the secondary circuit through an intermediate heat-transport circuit using sodium as a coolant. It represents a barrier between the radioactive primary circuit and the non radioactive water/steam system. The primary circuit and the intermediate circuit are connected through Intermediate Heat Exchangers (IHX). Thermal exchange between the intermediate circuit and the secondary circuit is done through Steam Generator Units (SGU). There are six of such loops (which means six IHX and six SGU).Core decay heat is removed by the same route. However when this route is not available, dedicated decay heat removal systems using six sodium/sodium dip coolers (DHX) immersed in the hot pool, take the heat directly from the primary system. The heat is rejected to the environment using sodium/air exchangers (AHX). Those six loops are organised into two systems, each consisting of three loops. An additional safety system (SGOSDHR) is designed to cool the sodium in the SG by heat exchange with external wall of the SG. A specific system is foreseen to cool the reactor pit.

GFR: The main specifications of the 2400 MWth GFR concept were driven by the internationally agreed objectives, which led to the main features of the concept:

- A fast neutron core with a zero or positive breeding gain (without or with reduced fertile blankets) and characterised by an initial plutonium inventory allowing for the deployment of a GFR fleet near 2040 (for sustainability and proliferation resistance);
- A three loops helium-cooled primary circuit (7 MPa at full-power operating mode, around 850-900°C at core outlet) connected to a Brayton secondary circuit (Figure 8) allowing for a high thermodynamic efficency (for economics);
- A decay heat removal (DHR) system initially based on dedicated loops allowing for forced or natural circulation (passive features of systems for safety concern);
- A spherical close-containment that aimed at first providing low pumping power (and related electrical supplies) for FCDHR following a RCS rupture and also to keep a pressure level that is consistent with the expected performance of NCDHR.

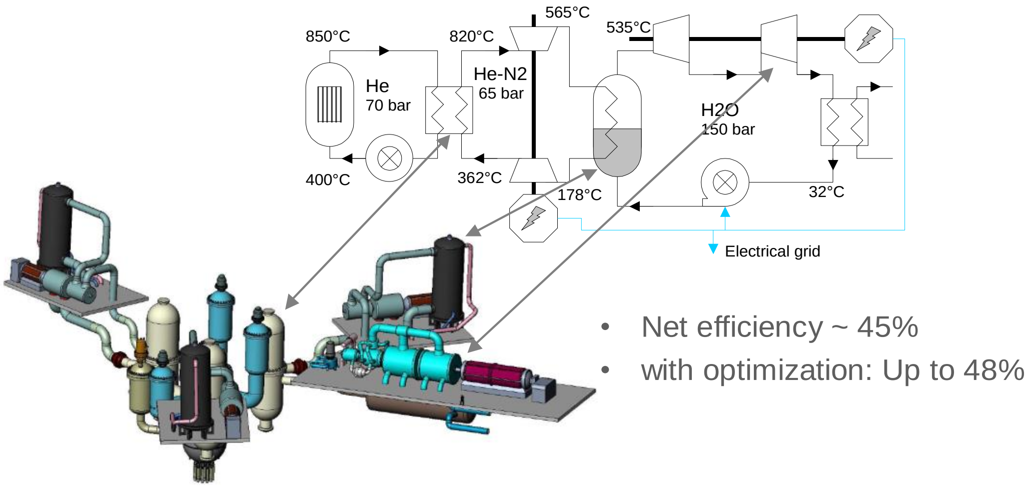

Figure 8: Layout of the GFR normal loops featuring a combined Brayton cycle

LFR: The primary system arrangement of ELSY can be seen in the following figure.

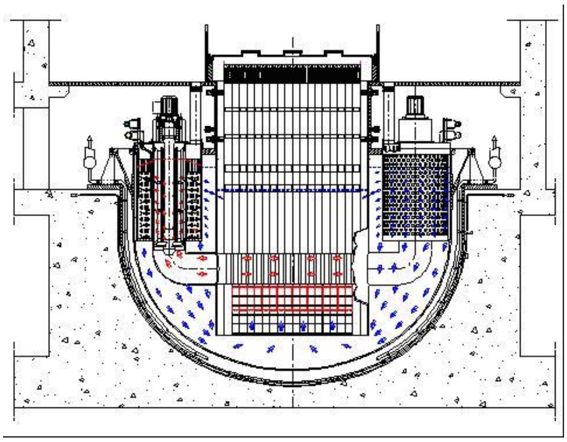

Figure 9 : sketch of ELSY primary circuit

There are eight flat-spiral Steam Generator Units (SGUs), each one of them encloses a coaxial Primary Pump (PP). The primary coolant moves upward through the pump impeller to the vertical shaft and then radially through SG tubes on the shell side out of the steam generator to the downcomer through the perforated double-wall outer shell. The coolant continues through the downcomer and at the bottom end of the RV it turns upwards through the core. Above the core the coolant turns to one of the SGU entrances, thus completing the full primary circuit path. The safety-grade DHR system of ELSY consists of the Reactor Vessel Air Cooling System (RVACS), the Direct Reactor Cooling (DRC) system, which is constituted by four water loops, and the Isolation Condenser (IC) sub-system, which branches off the feed water steam system. Thus, the overall DHR reliability and decay heat removal capability shall be achieved by a combination of the three systems, RVACS, DRC and IC.

VHTR: The ANTARES main features are:

- Reactor core thermal power 600 MWth;
- Primary coolant Helium;
- Core inlet temperature 400°C;
- Core outlet temperature 850°C;
- Indirect cycle arrangement;
- Two options for the primary loop: one loop with plate type Intermediate Heat eXchanger (IHX) or two loops with two tubular type IHXs;
- IHX secondary inlet temperature 350°C;
- IHX secondary outlet temperature 800°C;
- Primary loop(s) coupled to the process heat application through an intermediate heat transport loop that uses as coolant a mixture of 80% Nitrogen and 20% Helium;
- Power generating system combined cycle gas turbine with steam bottoming cycle.

The ANTARES plant includes the following key components/systems: the Vessel System (VS), which includes the Reactor Pressure Vessel (RPV, housing the reactor system, the reactor internals and the reactor support structures), the Intermediate Heat eXchanger (IHX) vessel (IHXV, housing the IHX and the Main Primary Gas Circulator (MPGC)), the cross vessel (housing the internally insulated hot duct and delimiting the surrounding annular cold duct), vessel supports, and lateral restraints (see figure below for a general sketch of the ANTARES concept).

The VS has the functions to confine the primary coolant and to maintain primary coolant boundary integrity.

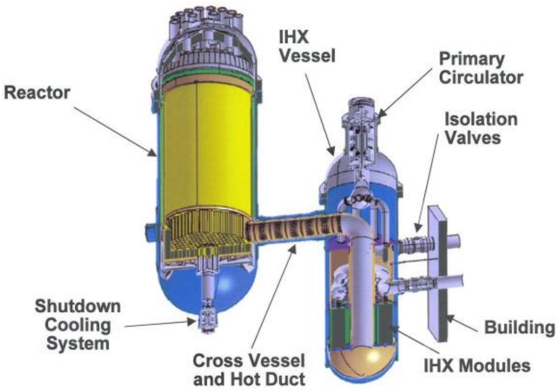

Figure 10: sketch of ANTARES primary circuit

- Reactor Core System (RCS), which includes the reactor core, the reactivity control assemblies (Normal Shutdown System (NSS) control rods (split in two groups: 36 operating control rods and 12 start-up control rods) and Reserve Shutdown System (RSS)), the core supports, the internal structures (permanent side reflector, replaceable reflectors, metallic core support (barrel), upper core restraint, upper plenum shroud), and the hot gas duct assembly. The RCS has the functions to generate heat from the release of energy of nuclear fission, to transfer heat to the primary coolant Helium and to confine radioactive products.
- Main Primary Gas Circulator (MPGC), which includes an electric motor and a compressor immersed in the primary Helium inside the IHXV. It is used to control the primary Helium flow rate modulating the rotational speed. The MPGC rotor is supported by active magnetic bearings and by catcher bearings. The MPGC includes also a shutoff valve to isolate the heat transport system from the RPV when the MPGC not operate
- Intermediate Heat eXchanger (IHX), which transfers the heat from the primary coolant to the secondary. Two types of IHX are envisaged: plate type and helical tube type. The number of IHX and IHXV depend from the IHX type selected. The IHX can be isolated from the Secondary System by means of the Secondary Separation Valves (SSV-H on the hot gas side and SSV-C on the cold gas side).
- Secondary System. The nuclear heat source is coupled via the IHX to a secondary system. The heat transferred to the secondary system can then be used to generate electricity either in a Brayton cycle or in a steam (i.e. Rankin) cycle. The secondary system can also be used as process heat including hydrogen production.

- Core Heat Generation Control System (CHGCS), which includes two reactivity control sub-systems, the Normal Shutdown System (NSS) control rods and the Reserve Shutdown System (RSS). NSS control rods are split into two groups: i) 36 operating control rods, ii) 12 start-up control rods. The NSS control rods have the same design and are made by boron carbide as neutron-absorbing material. NSS is used to maintain sub-criticality during cold shutdown conditions, to compensate for the xenon effect, to compensate reactivity effect in case of water ingress accident. The RSS is equipped by spherical neutron-absorbing elements that are dropped into the core fuel assembly channels by gravity. RSS provides reactor shutdown independently and diversely from NSS and it is designed for maintaining the core in sub-critical state if the NSS fails to operate.
- Secondary Decay Heat Removal System (SDHRS), a non safety-related loop implemented on the secondary system. The operation of this system requests the availability of the forced helium circulation in the primary circuit and the IHX integrity. SDHRS could also be used for normal start-up and shutdown of the plant.
- Shutdown Cooling System (SCS), a non safety-related system which include three heat transport circuits in series designed to remove heat from the RCS and transfer that heat to the ambient air. The first circuit is in parallel with the plant Primary Heat Transport System (PHTS) across the RCS and consists of a helium-to-water heat exchanger, an electrically powered gas circulator and a shutoff valve. The second circuit is a closed pressurized water heat transport loop that runs from the helium-to-water heat exchanger to a water-to-air heat exchanger. The water is circulated by conventional electrically powered pumps, and the ultimate heat sink (third circuit) is an air-blast type heat exchanger with electric fans. SCS can operate even if the secondary circuit and the primary forced helium circulation are not available. SCS is designed for achieving this function in pressurized and depressurized conditions.
- Reactor Cavity Cooling System (RCCS), a safety-related passive water cooling system for decay heat removal during emergency cool-down, for cavity heat removal during normal plant operation and for confining of radioactivity released into the reactor cavity during normal operation. The RCCS consists of two independent and redundant trains operating in natural circulation. Each train consists of the following four major components, plus associated pipes, headers and valves, all located inside the reactor building and the reactor auxiliary building:
  a) a panel wall cavity cooler, consisting of alternating vertical pipes around the periphery of the RPV (a compact air-to-water heat exchanger that surrounds the RPV);
  b) a water storage tank (a water-to-water heat exchanger is inside and integral to the pressure boundary of the water storage tank);
  c) a water-to-air heat exchanger (closed circuit cooling tower);
- Helium Processing System (HPS): System to transfer, to purify and to store the Helium.
- Fuel Handling and Storage System (FHSS): System to handle the fuel and reflector blocks, and to transport them between the receiving facility, the reactor core, and the fuel packaging and shipping facility;
- Reactor Control and Protection System (RCPS): System to provide the monitoring and control of the technological processes in all modes of plant operation, including emergencies.

The following table recalls the main features of coolant circuits for the 4 representative concepts.

	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	
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	

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Maximum temperature (°C)	490	535	n/a	550/250

DHR secondary				
Nature of coolant	DRC : sodium DHRTV : water SGOSDHR : air	Water	Water	Water/air
Mass or volume of the fluid	6 loops ~15 m^3 / 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	FC + NC in He / pressurized water	NC	

Table 2: Main circuits features for the four representative concepts

References of chapter 2.2.2

[VHTR-2.2.2_1] ANTARES : the HTR/VHTR project at Framatome ANP. Gauthier, Brinkmann, Copsey & Lecomte. Nuclear engineering and design 236, 2006.

[VHTR-2.2.2_2] Safety aspects of the modular high temperature gas cooled reactor (MHTGR),Silady, G.A., Millunzi, A.C., General Atomics, 1989

[VHTR-2.2.2_3] HTR coated particles and fuel elements – Present development. Languille. Eurocourse.

[LFR-2.2.2_1] ELSY Design Review of the Option Selection, DOC/08/043 Feb 2009

[SFR-2.2.2-1] EFR 98 – European Fast Reactor. Outcome of design studies. Commercial leaflet.

2.2.3 CONTAINMENT FEATURES FOR EFR

For the EFR reactor, the containment design has been quite well defined and information is provided in this chapter. Information for the other reactors are provided mainly in the following chapter. In the **EFR** project, the containment function is provided by three physical barriers implemented in series between the radioactive products and the environment. These barriers are:

- the clad;
- the primary containment;
- the secondary containment.

The boundary of the primary containment is formed by (cf. Figure 11):

- the primary vessel;
- the roof;
- the components seals;
- the external primary sodium purification loop;
- the primary cover gas circulation and purification system.

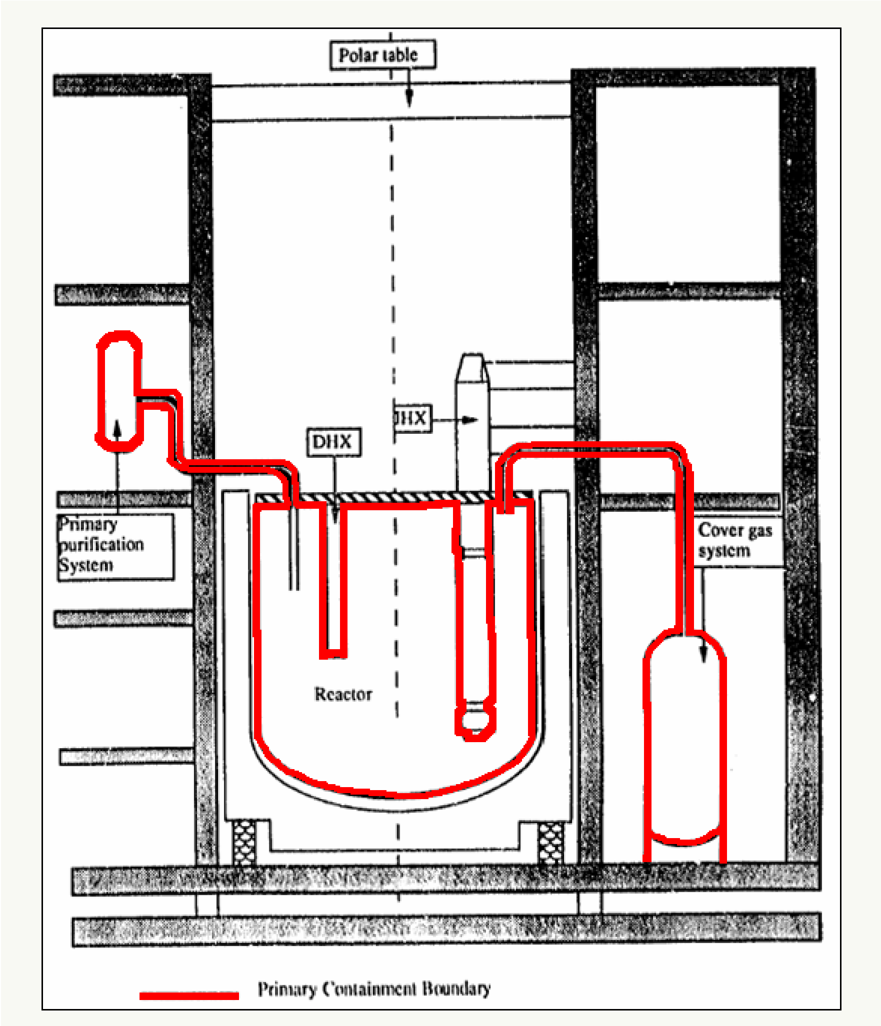

Figure 11: EFR - primary containment boundary

The boundary of the secondary containment is formed by (cf.):

- the reactor building (reinforced concrete) and its base mat;
- the walls of the secondary piping chamber inside the reactor building;
- the polar wall facing the secondary sodium pipe chambers;

- the leakjackets around top of integrated heat exchangers (Direct Heat eXchanger, Intermediate Heat eXchanger) and secondary sodium pipes above the roof and the connections of these leakjackets to the polar wall in order to ensure the continuity of the containment;
- the tubes of the IHX and DHX.

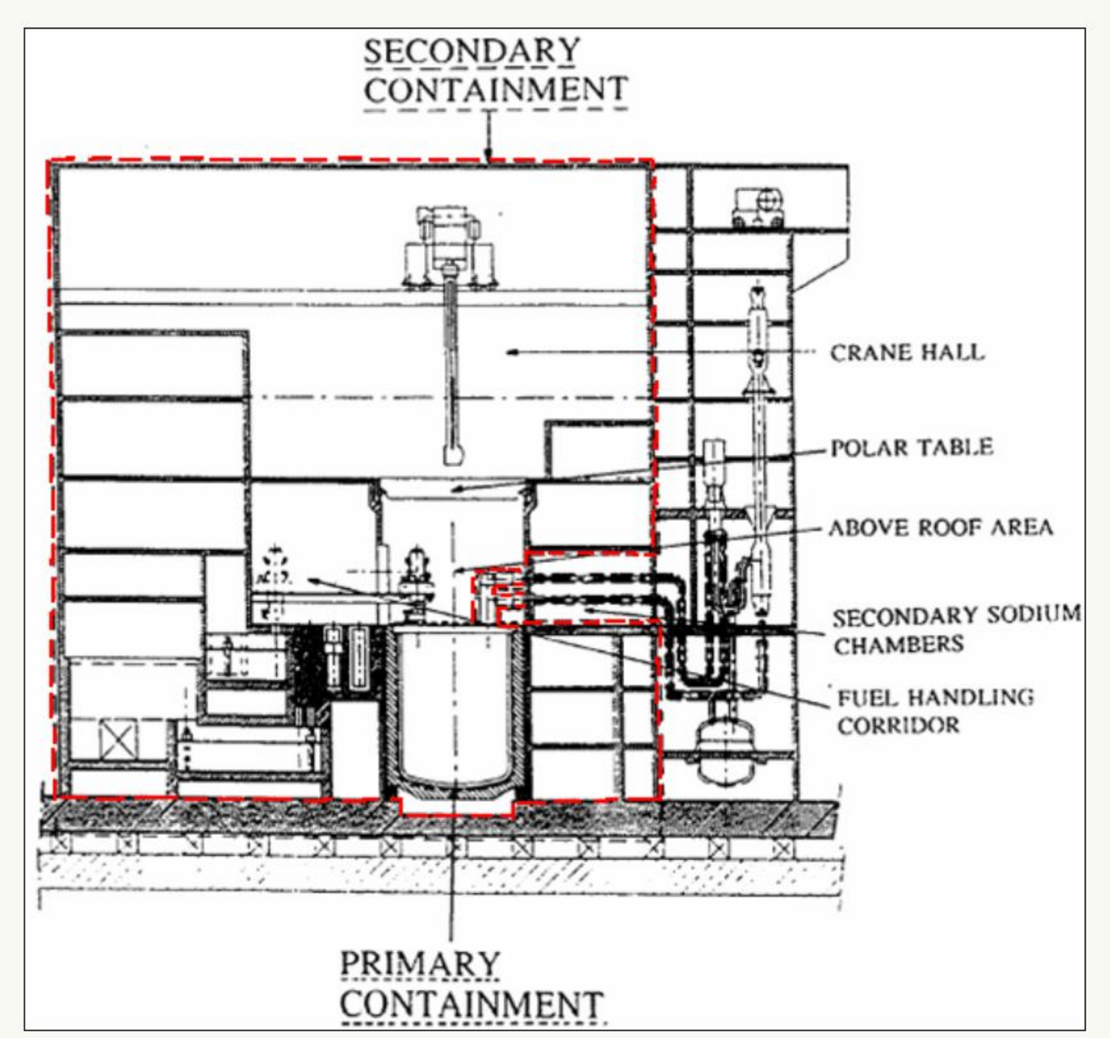

Figure 12: EFR – Secondary containment boundary

The containment has been designed to mitigate the consequences of the beyond design basis Plant State III which corresponds to a Core Disruptive Accident (CDA) leading to a large release of primary sodium through the roof.
The safety functions ensured by the secondary containment are the following:

- To limit the consequences of radiological releases following a leakage of the primary containment. In case of a hypothetical occurrence of a large radioactive source in the primary containment, the release ways for the radiological products from the primary circuit to the secondary containment and then to the environment are summarized in the following figure;
- As far as possible, to control the releases in the environment, except if this does not allow to minimise the radiological consequences;

- To protect the systems and components which ensure the safety functions (reactor shutdown, decay heat removal, primary containment) against externals hazards.

The secondary containment function is ensured by a dynamic mode: in normal operating conditions, and for all the conditions leading to radiological releases in the environment, except for CDA, the reactor building is maintained at sub-atmospheric pressure and the effluent is released in the environment through filters at the stack. The dynamic mode allows controlling the releases. In case of CDA, the reactor building is isolated.

In order to maintain the reactor building at sub-atmospheric pressure, the reactor building must have a small leak rate. For EFR, the proposed leak rate is 1% of the volume of the reactor building per day at 10 mbar overpressure. This leaktightness is maintained up to 250 mbar overpressure.

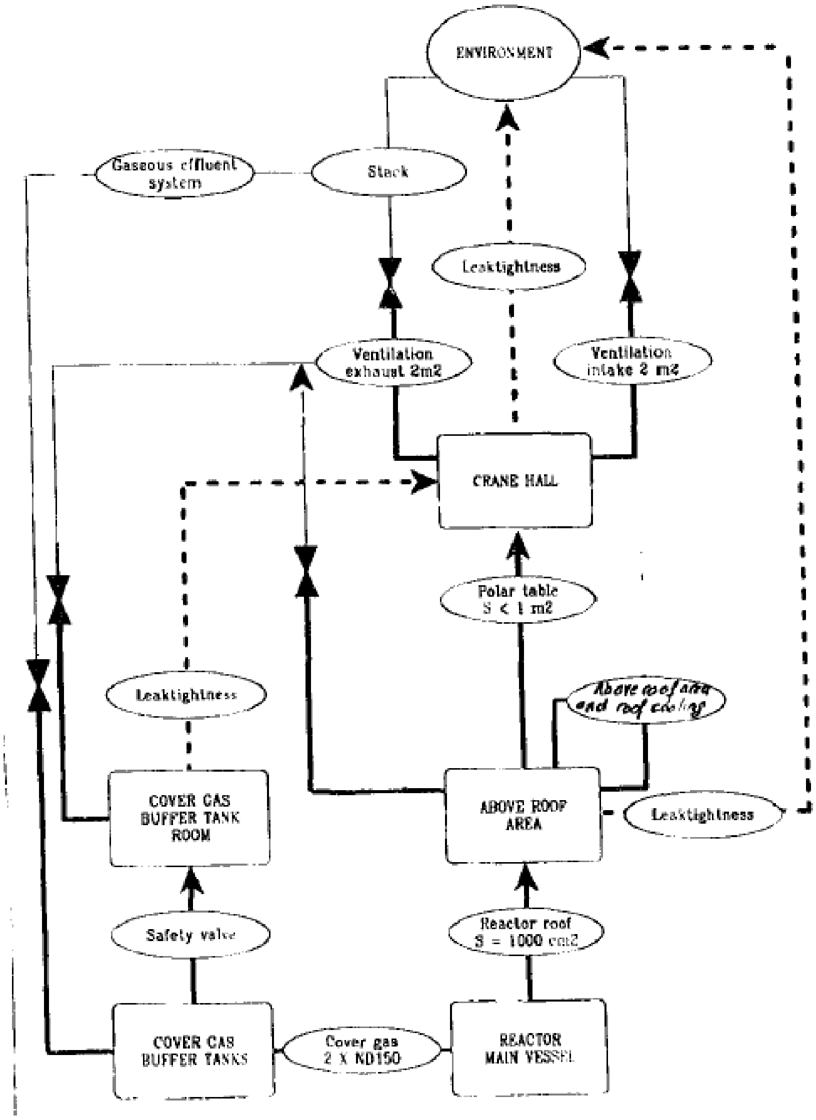

Figure 13: EFR – Release ways of radiological products

2.3 SPECIFIC DEGRADATION MECHANISMS AND DAMAGE CRITERIA RELATED TO THESE CONCEPTS

2.3.1 CLASSIFICATION OF PHENOMENA

In Figure 14 are exhibited the representative containment failure modes, as defined in the WASH-1400, and relative to PWRs.

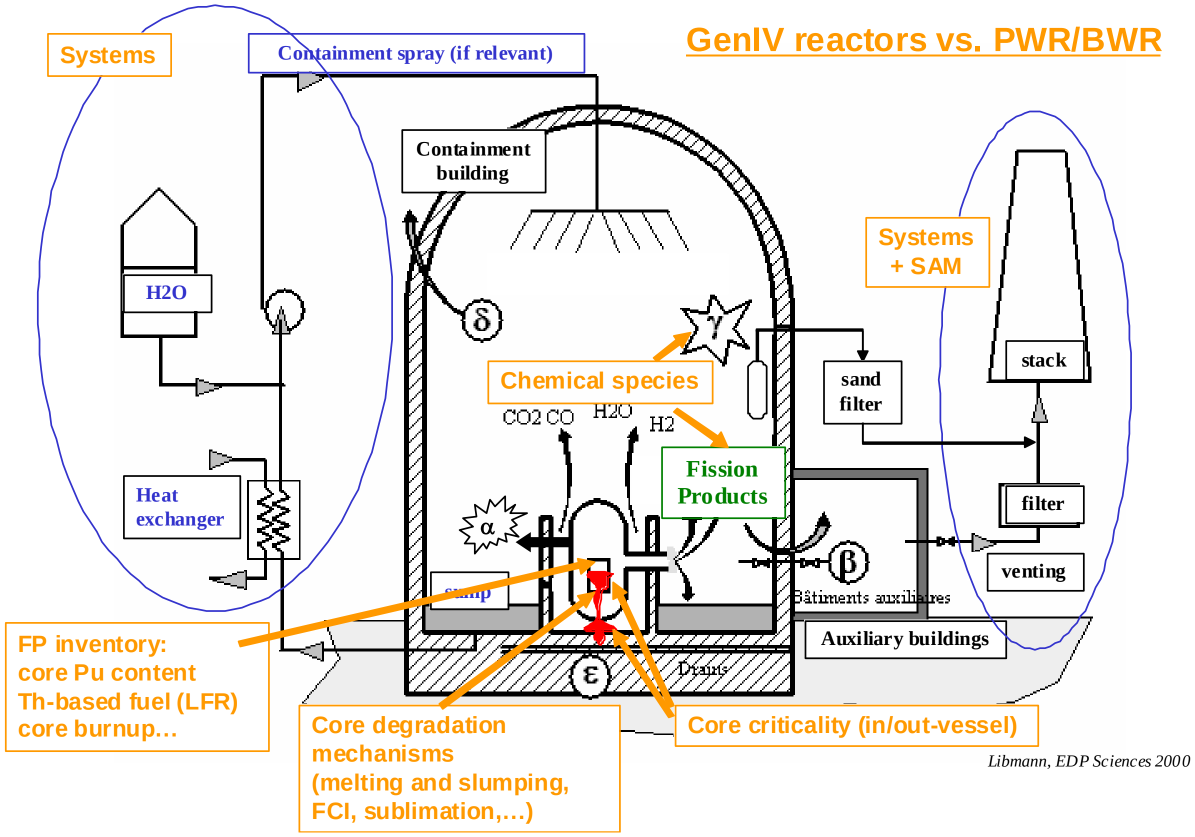

Figure 14: Containment degradation modes as defined in WASH-1400

Representative containment failure mechanisms are depicted by the so-called α, β, γ, δ, ε-modes:

- α-mode corresponds in general to the steam explosion mechanism in water-cooled reactors: in terms of consequences, the missile generation threatening the containment and SSC is dreaded;
- β-mode corresponds to the containment isolation failure:
  - Trough interfacing systems as a result of induced rupture of heat exchanger walls following the core degradation onset;
  - By containment penetrations to auxiliary buildings (failure to isolate the containment);

- γ-mode corresponds to combustion phenomena (mostly H_2 combustion in PWRs) in the containment building potentially leading to its early failure;
- δ-mode corresponds to the late failure of the containment due to slow over-pressurization;
- ε-mode corresponds to the containment failure through the base mat, generally following the corium spreading in the containment building.

According to the state-of-the-art for L2PSA modelling, the various containment degradation modes can be arranged in a flowchart depending on the time phase of the severe accident progression (see).

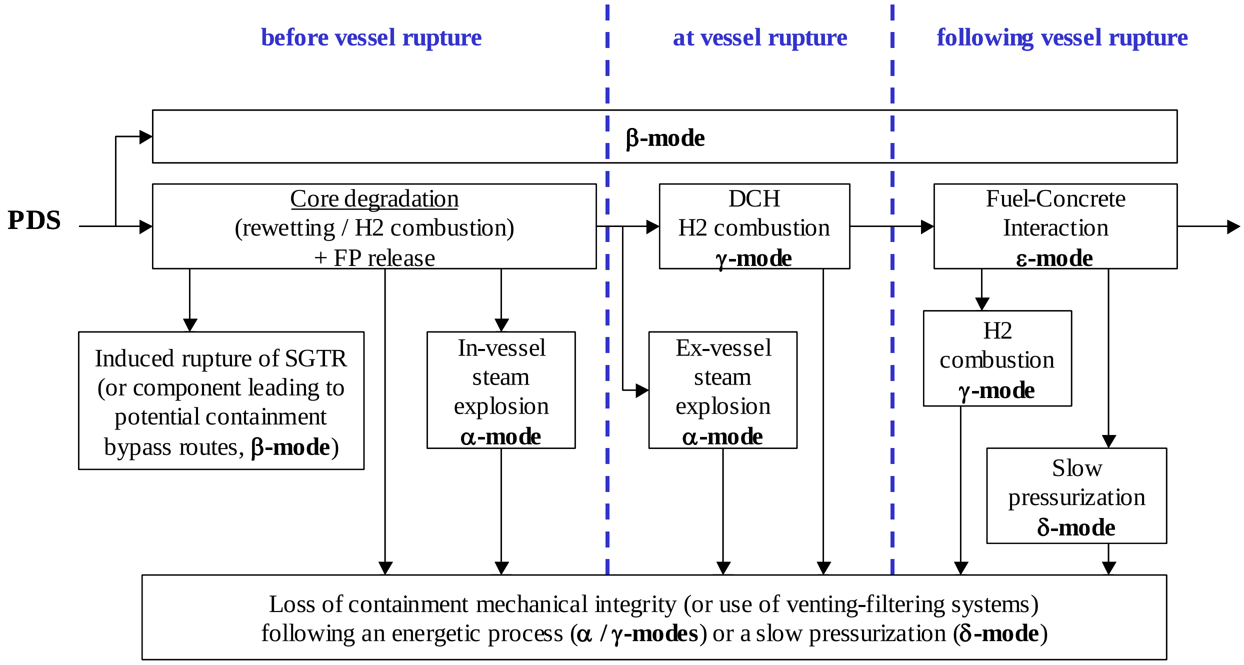

Figure 15: Flowchart of the containment degradation modes with the time frame of the severe accident progression

This depiction mean is used to organise the following paragraphs. For instance, the α-mode relates specifically to the steam explosion in LWRs. Therefore, it is not immediately obvious to look for compliance with this failure mode in Generation IV representative concepts. It is therefore proposed to “expand” the “α-mode” to all energetic processes (e.g. missile emission or rapid pressurisation) that could challenge the primary vessel integrity (thus causing a “lack” of radioactive materials retention within the RCS, i.e. the second barrier) and lead to an early failure of the containment building. Then, the HCDA for SFR and the potential fluid-structure interaction will be classified for simplification in the “α-mode”.

Based on these (expanded) failure modes to describe the associated risks related to the four selected concepts, it is intended hereafter to provide elements regarding:

- The “key parameters” and related phenomena associated with core degradation mechanisms (chapter 2.3.2);

- The containment building features and potential specific degradation processes (chapter 2.3.3);
- The specific provisions implemented in order to reduce the consequences of SAs (chapter 2.3.4);
- The main elements regarding FP thermo-chemistry for the source term assessment (chapter 2.3.5).

2.3.2 KEY PARAMETERS RELATED TO CORE DEGRADATION MECHANISMS

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

2.3.2.1 Material inventories

According to the depictions of the core and primary circuit materials and of coolant inventories nature involved in reactor circuits, some data are provided hereafter for the assessment of the accident progression tree and related phases (core degradation, vessel rupture and slumping in the containment building).

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
fuel	$(U,Pu)O_2$	(U,Pu)C	$(U,Pu)O_2$	UO_2
U inventory (kg)	32651 (+31174 : axial + radial blanket)		34028	
Pu inventory (kg)	8808		6338	
MAs inventory (kg)	177		0 at BOL, 400 kg at equilibrium	
“cladding”	Stainless Steel (SS)	SiC/SiCf	9Cr-1Mo T91	
inventory (kg)		13000	18830	
Control rods	B4C + EM10 + Na	B4C + SiCf	B4C	
inventory (kg)	7830 (24 CSD of 300kg & 9 DSD of 70kg)		1308	
Moderator	-	-	-	**graphite**
inventory (kg)	-	-	-	
CRs cladding	Stainless Steel (SS)	SiC/SiCf	9Cr-1Mo T91	
inventory (kg)	-		800	
Structural materials (in core region: grids, wrapper…)	Stainless Steel (SS)	SiC/SiCf	9Cr-1Mo T91	9Cr-1Mo

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
inventory (kg)	-			
Vessel internal structures (in lower head, diagrid…	Stainless Steel (SS)	9Cr	9Cr-1Mo T91	9Cr-1Mo
inventory (kg)	-			
Vessel lower heat	not applicable		9Cr-1Mo T91	
Inventory (kg)	-			

Table 3: nature and inventories of core and primary circuit materials

2.3.2.2 Core materials behaviour at high temperature (melting / slumping / sublimation) and potential interactions of fuel/cladding with primary coolant or foreign fluids (water/air/...):

SFR:

Behaviour at high T of fuel and cladding:

- At ~930°C: sodium ebullition;
- At ~1400°C: clad melting (corresponding to steel melting point);
- At ~2800°C: fuel melting.

Interactions of fuel with coolant (Fuel-Coolant Interaction FCI): At low temperature, there is a chemical reaction: the sodium reduces oxide and forms a compound which has a bigger volume than the oxide volume and so it could lead to clad rupture and fuel fragmentation which could scatter in the primary sodium. To avoid this event, the following means are foreseen:

- Detector of clad rupture (two delayed neutron detections);
- Automatic reactor shutdown initiated by two reactor trip systems;
- Location of cladding ruptures system.

When the sub-assembly where the cladding ruptures occurred is located, it is removed and placed in core periphery. The experience showed that cladding rupture is a frequent event (1/year). At high temperature, interactions between $U0_2$ and Na could lead to a steam explosion. Nevertheless, the phenomenon is limited in regard with a phenomenon with water, thanks to the thermal conductivity of sodium.

Interactions of fuel with foreign fluids:

Melting fuel reacts with the concrete. When melting fuel start to react with the concrete, the water contained in the concrete will have already react with the sodium.

In EFR concept, there were not water circuit nearby (roof is cooled by air). The melting fuel will be always recovered by sodium, so there will not have reaction with air.

GFR:

Behaviour at high T of fuel and cladding ((U,Pu)C + W-Re(liner) + SiC): For high heat-up rates, thermodynamic calculations were performed with a relevant chemical composition. Considering a homogeneous mixture of the various compounds, a liquid phase would appear around 1600°C. Above 2200°C, the only phases that exist at the equilibrium are solid SiC plus a liquid phase including a lot of fissile materials.

For lower heat-up rates (i.e. slow transients), experimental tests have been performed on a system made of the W-Re liner and the SiC cladding and for temperatures ranging from 1000°C to 1600°C (i.e. just below the temperature range related to severe accidents). Within this temperature range, the interactions between materials occur at the solid state. For higher temperatures, thermodynamic calculations are exhibiting stability up to 1845°C, temperature at which a liquid phase appears.

Regarding the interaction between the fuel and the liner, thermodynamic calculations at equilibrium indicate the formation of a liquid phase at around 1880°C. Finally, according to thermodynamic calculations, the heating of the fuel induces a liquid phase formation around 2200°C (i.e. solidus temperature) and is fully liquid at 2400°C (i.e. liquidus temperature). By now, the reaction kinetics of the fuel decomposition is not known.

Interactions of fuel with coolant (Fuel-Coolant Interaction FCI): This topic is not relevant for helium coolant.

Interactions of fuel with foreign fluids: For interactions with oxygen (in case of the unlikely event of air ingress into the core region, thanks to the presence of a nitrogen-filled close-containment) and owing to available bibliography, experimental test carried out between 1000°C and 1700°C studies and supplemented by thermodynamic calculations, have shown two oxidation features: a passive oxidation with the formation of a protective SiO_2 layer at low temperature / high oxygen partial pressure, and an active oxidation with the formation of an unstable SiO layer at a high temperature / low oxygen partial pressure. Considering that air ingress would also be associated to nitrogen ingress (also because of the presence of nitrogen in the close containment), preliminary results of experimental tests exhibit that the reaction rate obtained with nitrogen (higher than at 1730°C in helium due the larger amount of oxygen) suggest that the SiC is oxidized but not nitrided. More analytical tests without oxygen impurities in the reactant flow are required to further conclude on the influence of nitrogen.

Due to the formation of volatile species, the oxidation of the silicon by steam is governed by a linear kinetics (non limited by the diffusion of the species through a protective layer because of the volatilization of Si). As a result, the reaction rate is largely higher (by a factor of 50) than that resulting from a passive oxidation by air. At a high temperature, the reaction rate can be very high and this reaction would lead to an erosion of the claddings.

LFR:

Behaviour at high T of fuel and cladding: The pin cladding and structural material used in the lead fast reactor, is T91 ferritic-martensitic (modified 9%Cr-1%Mo VNb) steel, which combines good thermal mechanical and irradiation performance. Austenitic steel (15-15 Ti mod Si), which have an advantage of having lower radiation swelling at higher temperatures, is maintained as second option. Concerning the corrosion behaviour, at low temperature (below 550°C) the phenomena is sensibly reduced by the in-situ growth of surface oxide layer on steel with sufficient oxygen concentration (> 10^{-6} %w). Decreasing the oxygen level (less to ~ 10^{-7} %w) both austenitic and ferritic-martensitic steels may suffer dissolution attack even at ~ 400°C. At temperature above 550°C, up to 600°C, within the oxygen control band, the formation and quality of the oxide layer on martensitic steels are uncertain. For austenitic steels, at the same time, the oxides are thin and not completely protective. Moreover at high temperatures in lead, oxidation kinetics may be accelerated too much, so oxygen-free coolant technology must be implemented.

Interactions of fuel with coolant (Fuel-Coolant Interaction FCI): It has been shown that the fuel and clad maximum temperatures are respected with coolant velocity being 1.5 m/s, which limits the pressure drop through the core (90 cm active height) to about 1 bar. The fuel lifetime in the reactor has been fixed to 5 years: the pellet depletion requirement results more stringent than the max displacement per atom (dpa) on clad (100) and, maybe, the problems arising from clad corrosion in a lead environment. In fact, the relatively low core outlet temperature minimizes the risk of stainless steel creep and the moderate T across the core inlet-outlet temperatures (400 - 480°C) reduces the thermal stress during transients.

From the results thermal-hydraulic core design, the above lead-cooled ELSY design seems viable with particular attention to draw on the operational control of the oxygen content in the lead coolant, in order to limit chemical fouling and the build-up of the oxide layer.

Interactions of fuel with foreign fluids: The core air/water ingress accident is not relevant in ELSY safety analysis. Concerning the corrosion behaviour, at low temperature (below 550°C) the phenomena is sensibly reduced by the in-situ growth of surface oxide layer on steel with sufficient oxygen concentration (> 10^{-6} %w). Decreasing the oxygen level (less to ~ 10^{-7} %w) both austenitic and ferritic-martensitic steels may suffer dissolution attack even at ~ 400°C. At temperature above 550°C, up to 600°C, within the oxygen control band, the formation and quality of the oxide layer on martensitic steels are uncertain. For austenitic steels, at the same time, the oxides are thin and not completely protective. Moreover at high temperatures in lead, oxidation kinetics may be accelerated too much, so oxygen-free coolant technology must be implemented.

VHTR

Behaviour at high T of fuel and cladding (UC+SiC): The HTR fuel design is aimed on the very low probability of releasing a significant amount of radioactivity up to the safety temperature limit of 1600ºC.

Interactions of fuel with coolant (Fuel-Coolant Interaction FCI): This topic is not relevant for helium coolant.

Interactions of fuel with foreign fluids:

- Air ingress into the primary system is a safety concern because of the potential for oxidation damage (graphite fire) to graphite structures and components within the vessel, and to the fuel (TRISO particles). At the operating and accident temperatures following a depressurisation significant oxidation is a distinct possibility. The extent of air ingress is dependent on design features, initiating event factors and subsequent accident scenarios. Depressurisation of the primary system to atmospheric pressure is a necessary for (atmospheric) air to enter the system. Air flow will be by natural circulation. During normal shutdown the graphite temperatures are below those necessary for significant oxidation. Natural convection ingress flow rates are usually limited to relatively low values owing to the high core flow resistances and resistances in other parts of system. Location, break size and type (single, double) are the major variations that affect the time for starting air ingress and the rate of ingress. Scoping calculations have indicated that most of the graphite oxidation occurs, at least for the first several days of significant ingress, in the graphite core support blocks below the core and the graphite lower reflector, with relatively little oxygen reaching the active core. Also, any heat released from oxidation in the active core typically affects only the lower regions and does not add to the peak fuel temperatures that would be reached under non-air-ingress accident conditions.

- Water/steam – originating from the secondary system - ingress into a hot reactor core causes three major safety concerns, namely, a positive reactivity insertion, chemical attack and a breach in the radioactivity confinement.

VHTR cores are usually under-moderated by design, so a moisture ingress event can cause a positive reactivity insertion. This effect depends on the degree of under-moderation and the total amount of moisture. This positive reactivity insertion could cause large transient increases in reactor power.
Chemical attack by moisture causes oxidation and corrosion of the graphite material in the core and, if exposed, the FPs as well. It could also challenge the structural integrity of graphite reactor internals and fuel elements. The reaction of moisture with graphite causes an increase of primary pressure and produces gases, including carbon monoxide and hydrogen, which would present additional safety concerns. Unlike the chemical reactions between graphite and air, graphite–water reactions at high temperatures are endothermic.

2.3.2.3 Interactions between the primary coolant and others fluids:

SFR:

Sodium interaction with water/steam: Interaction between sodium and water is an exothermic and violent (explosive) reaction.

$Na + H_2O \rightarrow NaOH + ½ H_2$ (ΔH=-180kJ/mol).

The reaction between primary sodium and water is eliminated by the interposition of an intermediate sodium circuit between primary circuit and steam/water circuit, and there is no water system in the primary circuit.

Sodium interaction with air: Interaction between sodium and air is also an exothermic reaction.

$2Na + ½ O_2 \rightarrow Na_2O$ (ΔH=-435kJ/mol)

$2Na + O_2 \rightarrow Na_2O_2$ (ΔH=-518kJ/mol)

Several barriers are interposed between sodium and air to avoid these interactions.

Sodium interaction with oil: In EFR, there is a risk of interaction between sodium and the oil of primary pumps. This reaction produces gases (methane and hydrogen) and solids compounds. Nevertheless, the quantities of oil are such that it could not have consequences on the safety of the core. Inside the pumps, leak jackets are foreseen to recover the oil leak.

GFR: This topic is not considered as relevant for gas-cooled reactors in particular owing to the inert nature of helium, without potential interactions with other fluids. However, the control of impurities in the coolant is of major importance to avoid potential chemical interactions with others fluids (role devoted to the helium purification system).

LFR: Lead does not represent a hazard because it does not react with water and air. In the case of accidental air ingress, in particular during refuelling, any produced lead oxide can be reduced to lead by injection of hydrogen and the reactor operation safely resumed.

VHTR: This topic is not relevant for VHTR in particular owing to the inert nature of helium, without potential interactions with other fluids. As for the GFR, the control of impurities in the primary coolant is of major importance in order to keep the moisture within defined bounds (in particular due to the presence of a large amount of graphite in the core region).

2.3.2.4 Key parameters - Core disruptive accident

The behaviour of **SFR** is quite different in some respects from water moderated thermal reactors, with implications for safety. The main differences are their neutron dynamics and properties of the coolant. A particular concern with fast reactors is that they are susceptible to large and explosive energy releases and dispersal of radioactivity following a core meltdown, called “Core Disruptive Accident”. CDAs have been the distinguishing concern in safety studies of SFR.

The progression of a CDA is generally classified into two phases. The primary phase includes axial rearrangement of the core materials induced by an increase of reactivity (generally due to coolant voiding for LM reactors) leading to the axial ejection of a part of the fuel and of fission energy production. This is accompanied by increased internal pressure due to coolant and fuel vaporization (for liquid metal cooled reactors, with potential energetic FCI), ultimately leading to a transition towards the secondary phase: the fuel not being coolable but the core being sub-critical, the hexagonal cans melt and the molten materials are relocated with a large core axial compaction leading to a second power excursion with rapid expansion of the fuel and subsequent termination of the chain reaction. It is worth recalling that if an initiator for core disruption leads to a “non sufficient” energetic initiation phase, the reactor core is left in an unstable state (eventually disrupted, uncoolable or neutronically unstable). A further core meltdown, i.e. the transition phase, could take place with the potential of localized recriticalities and then leading to secondary energetic excursions.

The amount of the dispersed fuel during the so-called “primary excursion” is fundamental for the SA progression depiction (i.e. re-criticalities or formation of a corium pool) and for the assessment of the thermo-mechanical energy release potentially challenging the vessel or the containment integrities. If accident conditions cause the fuel bundles to melt and rearrange, reactivity could increase.

Besides the risks inherent to FR cores (see hereunder for more details), it should be first underlined that the amount of heat that can be stored in the primary coolant is important during transient conditions and therefore has an impact on potential core degradation kinetics. Indeed, cores having coolants with a great capability to absorb the decay heat are exhibiting softer transient kinetics and lower Peak Cladding Temperature (PCT). Therefore, all liquid cooled reactors have a large capacity to store the heat produced in the core region. For gas –cooled reactors, one should distinguish two different situations: the VHTR for which the huge amount of graphite play a major role in the core decay heat storage, like for liquid coolants, and the gas-cooled fast reactor for which the heat storage capacity is well below that of liquid metal coolants or of moderated cores.
Therefore, for SFR, LFR and VHTR, long durations (order of magnitude: several hours) to reach PCT are expected during accidents initiated by the loss of heat sink.

According to the experience feedback of Severe Accidents studies performed for SFRs (computations, in-pile experiments…), a work energy release caused by an uncontrolled reactivity insertion (e.g. for a Transient Over-Power, due to the voiding effect following an Unprotected Loss of Flow, or due to a gas ingress into the core); this phenomenological trend of fast reactors is potentially leading to the CDA.

SFR: The EFR safety approach requires the optimization of the design of the plant so that the consequences of HCDA are as low as reasonably possible (ALARP principle). In case of hypothetical core disruptive accidents (HCDA) able to lead to large radiological releases in the environment, the main safety functions which are requested are the containment function and the decay heat removal. There are a large numbers of possible initiators of a HCDA. The typical faults are the following unprotected events:

- Slow and fast loss of primary flow (LOF);
- Loss of main heat sink (LOHS);
- Slow and fast transient overpower (TOP);
- Subassembly accident propagation (SAP);
- Loss of decay heat removal systems (LDHR).

The primary containment is designed in order to minimize the consequences due to a HCDA. Otherwise general states which could have resulted from HCDA combined with postulated failures of sensitive parts of the primary containment have been foreseen. The aim is to assess the effectiveness of the secondary containment in terms of radiological releases to the environment. A limited number of beyond design basis plant states which represent starting conditions for the assessment of the secondary containment is considered. They are defined by judgment based on previous experience from analyses of HCDA taking into account the foreseeable cliff edge effects and using enveloping assumptions. Four Plant Damages States have been defined:

PLANT STATE	DEFINITION
PS I	It corresponds to core disruptive accidents without sodium impact under the roof : - fractions of fission products released into the cover gas : . noble gases : 100 % . volatile fission products : 10 % . solid fission products : 0.01 % - nominal roof leakage - no primary sodium onto the roof
PS II	It corresponds to core disruptive accidents leading to limited leakages of the roof : - fraction of fission products released into the cover gas : . noble gases : 100 % . volatile fission products : 1 % (*) . solid fission products : 0.01 % - the roof leak area corresponds to about 1/10 of the total roof seal area : about 100 cm² - 0 to 100 kg of primary sodium is discharged through the roof. It leads to a spray fire over 1 second
PS III	It corresponds to core disruptive accidents leading to large roof leakage : - fractions of fission products released into the cover gas : . noble gases : 100 % . volatile fission products : 1 % (*) . solid fission products : 0.01 % - the roof leak area corresponds to the total roof seal area : about 1 000 cm² - 100 kg to about 1 500 kg of primary sodium is discharged through the roof. It leads to a spray fire
PS IV	It is the catastrophic case with collapse of the primary circuit and the complete failure of decay heat removal systems

* Volatile fission products released in the cover gas are assumed 10 times lower in PS II and PS IIIthan PS I because PS II and PS III are the result of energy releases which lead to larger mixing of the primary sodium and the volatile fission products.

Table 4: Plant Damage States definition for EFR

The containment function is ensured by the primary containment and by the secondary containment. The analyses performed in the frame of the EFR have shown that it was possible to demonstrate the efficiency of the secondary containment to mitigate the consequences of the beyond design basis Plant State III. Indeed, the definition of the beyond design basis Plant State III corresponds to a HCDA leading to a large release of primary sodium through the roof. The amount of sodium released through the roof is defined as 1500 kg, corresponding to an envelope of the quantity able to be ejected without structural failure of the roof. In this case, due to the high velocity of the ejected sodium (the sodium flow through the roof is several 1000 kg/s), the occurrence of a large spray sodium fire cannot be excluded.

GFR: Hypothetical Core Disruptive Accidents (HCDA) have traditionally played a major role in liquid metal fast reactors safety evaluations. Because a generic feature of fast reactors is that the core material is not assembled in its most reactive configuration, this will also be a great concern for the GFR. Therefore, a substantial effort should be devoted to assess the consequences in such situations. By transposition of SFR depiction and knowledge regarding HCDA, two scenarios could be encountered:

- If an energetic initiation phase occurs, the fuel dispersal and the work potential that is related to this phenomenon could lead to prohibitive strain on internal structures of the reactor vessel or on the Helium Pressure Boundary (HPB) itself. However, it is worth noticing that a gaseous coolant (compressible) could help to limit the fluid to structure interaction compared to liquid metal coolant by instance. On the other hand, a sudden rupture of the HPB could be dreaded if safety relief valves are unavailable to limit the gaseous pressure increase;
- If a non-energetic phase occurs, a further core meltdown (called "transition phase") could be foreseen with "predictable" occurrences of recriticalities and power excursions, and by consequence to challenges on the barriers (especially the metallic HPB for GFR).

Specificities regarding the nature and arrangement of fuel/cladding/coolant in case of a CDA: If one assumes that the fission products pressure is absent (i.e. fresh fuel) or neglected, as the vapour pressure of carbide fuel is lower than that of oxide fuel (for a same temperature level), the core disruption will occur at slower rate and then the energy deposited in the fuel will be higher. This constitutes a major difference between SFR (generally with oxide fuels) and GFR for the HCDA depiction. On the other hand, owing to the macro-structured fuel concept (i.e. in a honeycomb like SiC structure) that is a specificity of the GFR design, one question is arising with regards to the thermo-mechanical potential of this structure to withstand a sharp vapour pressure increase compared to a pin-type structure involving a gas plenum.
Finally, it should be recalled that for Plant Damage State definition (and related course of the severe accident), as the vapour fraction in high-pressure condition would be smaller than in low-pressure one (knowing that the rate of change of vapour pressure with temperature is roughly proportional to pressure), a more rapid fuel relocation (and reactor shutdown) would therefore be expected in high-pressure conditions (e.g. following a loss of circulation capability) compared to a low-pressure one (i.e. following a depressurisation).
Effect of FPs and behaviour during HCDA: Up to date, there is a lack of knowledge regarding FPs effect (for fuel dispersal) and behaviour (instantaneous or delayed release) in carbide fuels.

LFR: The possibility of reactivity increase due to coolant voiding effect seems to be not a main concern in the lead fast reactor, because of the high boiling temperature of lead (1737°C); in addition the core design optimization would provide to inherently activated negative feedback in the core. In the case of core disruption, it can be envisaged a condition in which the fuel dispersion dominates over fuel compaction, thus reducing considerably the likelihood of severe re-criticality events in the late phase of core melt. In fact, lead density (slightly higher than that of fuel density) and convective streams make it difficult to consider scenarios leading to fuel aggregation with subsequent formation of a secondary critical mass (or corium pool).

VHTR: The core design features are such that the worst reactivity excursion possible will not result in damage to the core. The core peak temperatures will stay below the temperature that marks the onset of fuel damage.

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

SFR: One first could distinguish various initiating events that led to an uncontrolled reactivity insertion:

- Single control assembly withdrawal: During the plant lifetime, the runaway of one or a few Control and Shutdown (CSD) assembly may be expected to occur. This fault is expected to be a frequent event. For all reactor operating conditions, the withdrawal of a single CSD will cause a local increase in sub-assembly power generations around the affected control assembly. This event could lead to a local fuel damage which could propagate to the others sub-assemblies;
- Multiple control assembly withdrawal: This fault is thus expected to be a frequent event. Nevertheless this event will be prevented by limitation systems. CSD rod array withdrawal faults may be initiated by an operator error, or by a malfunction within the reactor control system. As a result the drive motors withdraw the CSD rods from the core. The initial operating conditions may be nominal full power and flow or any partial load normal operating condition. For all reactor operating conditions, the simultaneous withdrawal of all control rods will cause core and sub-assembly power generation to rise and sodium temperature to increase. Unless terminated such faults may initiate a core disruptive accident as a result of fuel melting and pin failure or coolant boiling and a rapid insertion of reactivity;
- Sodium voiding: Voiding of the sodium coolant within the core can lead to an increase in reactivity and a consequent power excursion. Such events can be initiated by a number of causes like sodium boiling, failure of components (including cladding ruptures) inducing the release of gas in the core, or entrainment of gas through the core. It has been shown that the minimum core voiding to cause excessive clad temperatures and fuel melting is at least 300 litres, i.e. two orders of magnitude greater than the credible volume of gas which may circulate through the core.

In addition, the core compaction is a dreaded event for Fast Reactors. The method of support of the sub-assemblies could give rise to the potential for reactivity excursions. They are free standing on the diagrid, and spacer pads located just above the fissile zone maintain their separation. Their high bending stiffness restricts their lateral displacement to applied loads. Radial bowing of the sub-assemblies will effect a change in reactivity. The reactivity perturbation is proportional to the radial displacement and dependent on the radial position of the sub-assemblies, those on the core periphery dominate the bowing reactivity behaviour of the core. A mean change of core radius of 1 mm gives a change in reactivity of about 7 cents

(25 pcm). The extent of free lateral movement of sub-assemblies depends on the clearance between the spike journals and the chandelle bushes and the clearance between the spacer pads of adjacent sub-assemblies. The core design aims to ensure that at nominal full power with nominal sub-assemblies dimensions there is little or no free movement between sub-assemblies. In reality with realistic tolerances a partial contact pattern will exist at part loads. Once in contact, little or no free movement is possible, subjecting the sub-assemblies to bowing and the spacer pads to compressive loads. The pads are designed to have a high stiffness, about 250 kN/mm in six faces loading. Pad loads in excess of 25 kN are required to subject the pads to plastic deformation. There is no definable loading event in which core loading might exceed this value.

GFR: Internal mechanisms by which accidental insertion of reactivity could occur in the GFR have been researched (for internal, one should appreciate that earthquake, potentially leading to a core shaking, is excluded). These mechanisms (and the maximum reactivity amount and rate) are mostly related to a control rod assembly withdrawal (inadvertent or forced), a coolant depressurisation or a steam/water inleakage. It should be first underlined that a full depressurisation of the primary circuit would induce a maximum reactivity less than 1$ (compared to LMFBRs, the use of gas allows excluding the classical large reactivity accidents, mainly due to voiding effects). Regarding steam inleakage, and owing to the DHR secondary circuit pressure (i.e. 1.0 MPa that is less than helium one), a DHR heat exchanger tube rupture will be protected first by the delay to obtain pressure equilibrium between these two circuits. In addition, an isolation of the affected DHR loop is foreseen thanks to dedicated valves implemented in the water-filled secondary circuit (and if moisture detectors able to detect this initiating event are available). If one assumes that the reactor scram is largely delayed (i.e. for a delay greater than that of pressure equilibrium depending on the leak size) and that a failure to isolate the secondary circuit occurs, the amount of steam potentially entering in the core region would be limited. In spite of this hypothetical situation (which is a combination of several failures), the neutron spectrum softened by the steam during the first seconds of the transient would cause a decrease of the multiplication factor and then a decrease in reactivity. Finally, the most severe accidental reactivity insertion is potentially due to a forced withdrawal of an inserted control rod assembly, which could result from a sudden and gross failure of CRA housing. It is worth noticing that CRA mechanisms will be implemented in the lower part of the GFR reactor vessel (for temperature constraints, and in order to enhance the natural circulation performance with the DHR loops), thus leading to the meaning that CRA would be inserted (and not ejected) into the core region in case of a sudden depressurization of its housing. Then, an additional event is requested to insert a large reactivity amount in this situation (i.e. failure of the mechanism implemented to stop the CRA at the lower part of the core).

In the course of the accident scenario, it should be mentioned that in core damage situations, the loss of core structures (e.g. following clad melting), which are neutron absorbers in normal operation, could lead to positive effects.

LFR: Safety assessment of ELSY didn't identify as safety concerns for core criticality air ingress and water ingress events.

2.3.2.6 Case of VHTR

VHTR: a short sketch of the postulated accidents and main lines of the accidental phenomenology is first required to answer properly to chapter 2.3.1 and 2.3.2 objectives. Starting point of the potential accidents is characterized by the plant operational state of the reactor, and the accident initiator. Here only accidents occurring on a full-power operating reactor are considered. Specific attention has to be devoted to the radiological pollution in the primary circuit at the moment of occurrence of the accident as the primary circuit is always contaminated to some extent (this subject is addressed below). Appendix B provides a complete set of accident initiators for a VHTR reactor (given the present state of knowledge) as well as some methodological elements on the way to establish such a list. Typical events to be analysed are:

Heat removal transients:

- P-LOFC (pressurised loss of forced cooling): There are two major safety related aspects to consider: the core heat up transient and the potential for delayed radioactive release. If all the active safety measures are failing for some reason, core temperature will rise but due both to the core high thermal inertia (along with the graphite weight) and to the low power density, this rise is quite slow. 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). Present manufacturing quality of the fuel particles enables them to preserve their fission products (FPs) retention qualities for hundreds of hours at this temperature. At thermal equilibrium (at reduced power), heat is evacuated through the core and then from the core to the vessel and from the vessel to the vessel cavity by passive heat exchanges. Only some means should be provided for to evacuate heat from the cavity to the ultimate heat sink which is the function of the Reactor Cavity Cooling System (see below). This gives rise to the second concern: the heat-up of metallic structures and equipment, such as control rod sleeves, core barrel and support and vessel. Depending on the stress-temperature combination, collaps by creep (-rupture) is a possibility.

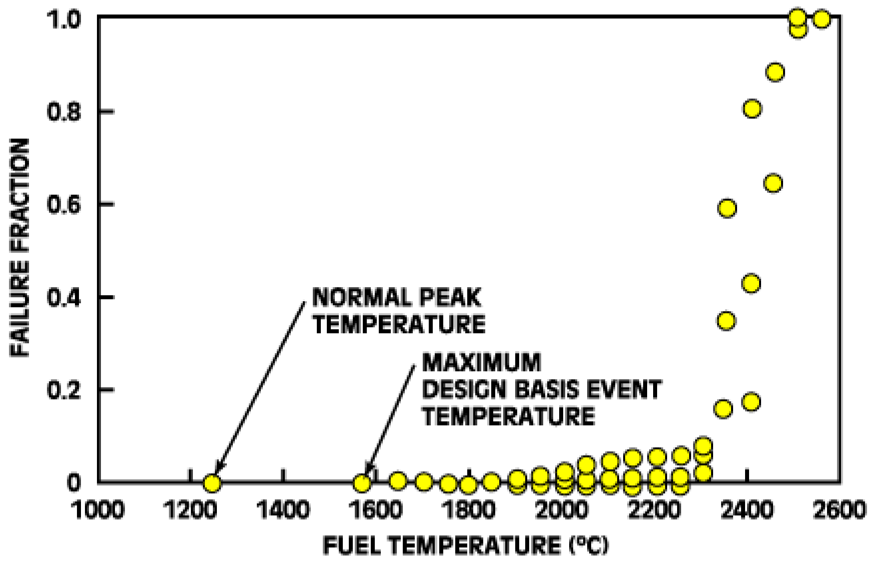

Figure 16: Integrity of MHTGR coated fuel particles as function of temperature [ref VHTR-2.3.1_2]

Primary system rupture transients:

- D-LOPC (depressurised loss of forced cooling): The major consequences are core heat-up and potential radio-active release (prompt as well as delayed) into the confinement and eventually into the environment (filtered and/or unfiltered). Because only permanent gases are involved, there is an efficient FP transport out of the circuit to the building and the environment only during the depressurization phase. Highest core temperatures in core heat-up events – and accordingly highest release from fuel elements – are achieved for a depressurised reactor, i.e. when the transport is already small. This leads to the fact that most relevant accident source terms of small, well designed HTRs are not necessarily related to the core heat-up, but to the depressurisation-induced release of FP stored in the circuit during long term normal operation (dust, plate-out).
- The rupture can cause pressure waves within the reactor that could threaten structural stability.
- Rupture with air ingress: Air ingress into the primary circuit can happen simultaneously to a loss of coolant through leaking. It will exert additional constraints, both chemical and mechanical, on the core materials (including the fuel particles), the structure materials and the FPs released in the circuit. Four main phenomena seem especially relevant for accident studies: graphite oxidation with air, air induced fuel particle damages, chemical interactions between FPs and air and FP transport by air. The damages are limited by the available air flow rate. Both air ingress and depressurization will lead to a contact between graphite dust and air with a potential risk of dust explosion.

- Rupture with water ingress: water ingress causes three major safety concerns that are a positive reactivity insertion (reactor power transients), a chemical attack (CO and H_2 production, potentially challenging the structural integrity) and breach of the confinement.

Accidents associated with reactivity control:

- Reactivity events: due to relatively large negative temperature reactivity feedback coefficient the safety consequences are negligible.
- ATWS: due to relatively large negative temperature reactivity feedback coefficient the reactor will stabilise at a low power level. The combination of the core high thermal inertia (along with the graphite weight), the low power density, and large thermal margin for fuel failure gives long delay times for manual actions.

Accidents associated with pressure transients:

- Turbine trip, loss of load, breaks etc. could cause pressure transients that could challenge the structural integrity of especially core structures.

Accidents with other sources of radioactivity (spent Fuel Pool loss of forced cooling).

- VHTR have sources of radioactivity outside the primary system boundary. See paragraph 2.3.4 for specifics.

Compared with LWRs, the confinement function of HTRs is mainly performed within the fuel particles; in consequence, different goals and requirements for other barriers such as the Containment structure/system in HTRs must be specified. Generally speaking, containment structures/systems usually provide the following accident prevention and mitigating functions that have specific performance requirements depending on each reactor technology and specific design concept:

1) Protection of risk significant Structures, Systems and Components (SSC) from internal and external events;
2) Physically support risk-significant SSCs;
3) Protect onsite workers from radiation;
4) Remove heat to prevent risk-significant SSCs from exceeding design or safety limits, if necessary;
5) Provide physical protection (i.e. security) for risk-significant SSCs;
6) Confine and reduce radionuclide releases to the environs (including accident and beyond design basis conditions);
7) Control the radiological releases (i.e., release at defined points and monitoring releases, providing adequate corrective actions).

Most of these functions, e.g. functions 1 to 4, are not subject to more discussion for HTR than for other reactor technologies and are only related to the state of the art and actual implementation of each specific design concept during the design and construction phases. The same holds true for Function 5, but it will be subject to improved and redefined security requirements for new reactors.

Functions 6 and 7 concerning the reduction of radionuclide releases into the environment are the only ones that may present certain aspects of specific HTR technology that could require developing significantly different HTR functional requirements compared with LWR containments.

The philosophy behind the functional requirements for LWR containments is that adequate time must be provided for fission product decay before allowing a release from the containment to the environment. After this time, based on the results of radiological analyses, when coping with long-term or gradual energy releases a design may use controlled venting to reduce the probability of catastrophic failure of the containment. In the meantime, a design may use diverse containment heat removal systems or rely on the restoration of normal containment heat removal capability. In this sense, it is convenient to remember that intrinsic to LWR technology is the fact that removal of decay heat from containment results directly in a significant reduction of containment pressure (steam condensation), that is the driving force for fission product releases.

The extrapolation of this philosophy to a HTR technology is not immediate for the following reasons:

- Reactor coolant in a HTR is not a condensable gas, so a large reduction of containment pressure cannot be expected by the long term removal of decay heat from the containment in scenarios with a breach of reactor coolant pressure boundary. As a consequence, the driving force of the containment pressure for fission product releases to the environment must be reduced in a totally different way than in a LWR.
- Potential radiological releases in scenarios with a breach of the reactor coolant pressure boundary are much less significant in HTR in the short term, since, as indicated before, the radiological inventories within circulating He coolant are very limited. Even lift–off of plated-out and dust embedded or attached in graphite components requires large openings in the reactor coolant pressure boundary, and their radiological inventories are also limited, although some uncertainties on actual reactor operational behaviour exist.
- A delayed FPs release is only possible for scenarios in which there is a loss of forced circulation cooling of the core and slow increase of core temperatures, where some fraction of the radiological inventory associated with failed or defective fuel particles or uranium contamination outside the fuel particles could result in delayed fuel releases. Potential long-term reactor coolant pressure boundary air ingress effects can be limited or avoided to minimize their potential impact on released radiological inventory.

Based on the behaviour of HTR accident scenarios, the functional requirements of the containment system to avoid radionuclide releases to the environment shall be tailored to fulfill the following strategy and requirements:

- The potential short term radiological releases as a result of a small breach of the reactor coolant pressure boundary will be limited to helium coolant circulating activity. The functional requirements of the HTR containment in these scenarios address the consequence of the limited radiological inventories that can be released. This shall be assessed radiologically but a direct release to the environment with a possible filtering process, based on ALARA criteria, is considered the optimal approach.
- The potential short term radiological releases as a result of a large breach of reactor coolant pressure boundary will also have to consider the potential lift–off of plated-out, the dust radionuclides embedded or attached in graphite components and, additionally, the activity of the He purification and storage system content. The functional requirements of the HTR containment in these scenarios will address the consequence of some higher but also limited radiological inventories that can be released. A direct release to the environment is considered the optimal approach to reduce the driving force (pressure) in the containment in the long term, when potential mechanisms of delayed fuel releases could appear. In the short term releases, the functional requirement of a filtering process would be determined by bounding the uncertainties on radiological dust inventories.
- An isolation of the containment after this initial release of a large breach of reactor coolant pressure boundary would ensure that the containment would provide the radiological line of defence function in case that such accidental scenario would result in potential delayed fuel release conditions. This isolation function can be supplemented by filtration provisions if active circulation is provided. This isolation function can also contribute to long term risks associated with air ingress within the reactor coolant pressure boundary. These can also be so prevented by the appropriate provisions as required.

Since the majority of the radiological inventory in a HTR is associated with intact coated particles that by design will remain inside the particles for all design accidental scenarios, some additional functional requirements could be imposed to the containment system based on principles of defence in-depth to prevent potential design or plant behaviour uncertainties in credible accident threats. These functional requirements should be based on the assessment of such uncertainties and the identification of the accident scenario characteristics with more likelihood to result in core fuel temperatures above the safety design limit of 1600ºC for sustained periods.

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2.3.3 COMPLIANCE AND POTENTIAL TRANSPOSITION OF CONTAINMENT DEGRADATION MODES

2.3.3.1 Potential transposition of containment degradation modes

Based on the phenomenology described above, it's proposed to make equivalences with the terminology used for LWR.

α-mode:

SFR-GFR : CDA may correspond to a α-mode (please refer to chapter 2.3.2.4).

LFR: α-mode is due to the SGTR (Steam Generator Tube Rupture), 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).

VHTR: The α-mode may correspond to dust explosion on VHTR.

<u>**β-mode:**</u>

Among the various fast or energetic degradation processes, the containment leak tightness may also be lost due to isolation failures. A containment isolation failure can be due either to an inadvertent pre-existing opening or to a failure of the containment isolation system. When an accident occurs, a number of valves must close to isolate the containment from the environment. If there is a non-isolable hole in the containment (like a spare penetration left unsealed) or if some containment isolation valves fail to close, the containment leakage rate will be much larger than expected during the whole course of the accident. As for LWRs, all the representative Generation IV reactors concepts will feature containment isolation devices (and their related failure modes) like equipment hatches, personnel airlocks, electrical penetrations, expansion bellows, containment isolation valves and the containment atmosphere will be controlled through the venting/filtering system (with failure modes. Therefore, as several modes of containment bypasses will be compliant with LWRs ones, the following items will be dedicated to specific containment bypass modes.

First, the control of the containment leak tightness is mainly carried out by a testing program, which objective is to guarantee that throughout the operating cycle, the containment leakage rate holds below the allowable risks. For atmospheric containments, methods exist to detect large openings in the containment during reactor operation. These methods do not have the same accuracy as the pressure tests of the containment, but they can be performed continuously. Accident management measures also exist to detect and isolate openings in the containment after an accident has taken place.

Of course priority should be given to the prevention of containment isolation failures, but, in line with the defence-in-depth concept, means should be available to the operator to take corrective actions after the onset of an accident. Finally, in some accidents, the containment building may be completely bypassed. Containment bypass arises with a fault sequence, which allows primary coolant and any fission products accompanying it to escape to the outside atmosphere without having been discharged into and mixed with the air in the containment volume. In Interfacing-System Loss Of Coolant Accidents (IS-LOCAs), check valves isolating low-pressure piping fail, and the piping connected to the reactor coolant system fails outside the containment. The radionuclides can escape to secondary buildings through the reactor coolant system piping without passing through the containment. A similar bypass can occur in a core meltdown sequence initiated by the rupture of a steam generator tube (SGTR) in which release is through relief valves on the steam line from the failed steam generator. The two paths, SGTR and IS-LOCA, only become important in the event of multiple faults, so they are designed to be of very low probability.

Because containment bypass sequences are potentially high consequence accidents, and because mitigation by accident management is questionable, it is important that those sequences are of a very low probability, and design provisions best achieve this. On the consequence side, large uncertainties still remain in the prediction of the retention factors in the auxiliary buildings (for IS-LOCAs) and in the secondary side of the steam generators (for SGTR).

<u>Identification of containment penetrations "specific to these concepts" for bypass concern:</u>

SFR: Penetrations of the secondary containment are isolable except for sodium circuits. A way of bypass is a leak between primary and secondary sodium. Nevertheless, to lead to a release to the environment it requires also a leak of the secondary sodium circuit. Another way of bypass of the containment is possible in case of HCDA if there is a leak on the leakjackets around the secondary and DRC sodium pipes.

GFR: Regarding the specific containment bypass routes, it is worth recalling that for the gas-cooled reactor an Helium Supply System is aimed at ensuring the coolant inventory in the RCS and also extracting continuously primary coolant for purification concern. Therefore and as far as this system is designed, two potential release path for FPs could be defined for failures affecting this purification system through the containment filtered ventilation system (δ-mode) if a break occur inside the containment building, and "directly" (β-mode) if the failure occurs on a gaseous tank for radioactive waste treatment, and located in an auxiliary building for instance (for intervention easiness).

Other containment bypass routes are related to components at interface of circuits (main IHX and SG). Compared to LWRs, it is recalled that the up-to-date GFR design features a gaseous intermediate circuit then leading to consider a combination of failures in order to lead to a containment bypass. In addition, the close-containment (even if this structure is not considered as a confinement barrier) has a potential for FP retention.

LFR: ß-mode may concern the bypass containment following SGTR (Stem Generator Tube Rupture) accident and failures of containment isolation.

VHTR: ß-mode may correspond either to some failure at the IHX level or to some containment leakage or to some defect in the containment filter.

<u>δ and γ modes:</u>

First, it seems important to provide the following features of the containment building related to the four representative concepts in order to look at the compliance with PWRs:

- Containment type (steel, concrete) and shape (cylindrical, spherical), presence of liner walls (if any);
- Free volume / geometry and compartmentalization (if defined);

- Operating & design (limit) pressure / temperature;
- Containment engineered systems (cooler capacity / spray…);
- Strategy of containment gaseous content relief (before filtering) vs. pressure peak (e.g. VHTR).

By now, three selected designs have focussed on feasibility key-points as regards to core features, RCS structural materials or to specific components (e.g. heat exchangers or blowers). It appears that the EFR concept is the only reactor that benefit from a design of the containment building. Therefore, the following table could exhibit some lacks or discrepancies between the four representative reactors of the generation IV concepts. However, some design choices will be listed as far as the design studies have been carried out.

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Containment free volume (m^3)	Around 150 000 m^3 including ~5 000 m^3 for the above the roof area and ~90 000 m^3 for the crane hall	Around 60000 m^3	40770 m^3	Around 9000 m3
Maximum mass of H_2 (kg)	0	1500 kg (conservative)	n/a	
Maximum mass of CO (kg)	0	6800 kg (conservative)	n/a	
Maximum mass of CO_2 (kg)	0	n/a	n/a	
Containment design pressure (MPa or bar)	250 mbar	About 3 bar	Around 3 bar	Not designed to cope with LOCA
Maximum pressure reached after deflagration in adiabatic conditions (MPa)	(Mechanical energy release) 250 mbar in the secondary containment (during sodium fire) 700 MJ in the primary containment in case of HCDA	Detonation / deflagration limits not reached according to flammable products obtained		

Table 5: Containment features

SFR: Over pressurisation could be the result of a sodium fire. In that way, a polar table has been design to limit the overpressure and pipes have been design to resist to the mechanical energetic release genreted by the sodium fire. Towards the hydrogen risk, a description is given below. Towards the LDHR risk, which is a slow sequence, diversified and redundant systems are foreseen that allows to practically excluded this situation ($f<10^{-7}$/y.r).

GFR: γ-mode should correspond to the potential combustion of gases in the containment building. According to the various materials involved in the GFR design, one should paid attention for CO combustion (coming from SiC oxidation for instance) and less likely of hydrogen in case of water ingress. However, this event is considered as limited thanks to the close-containment avoiding oxygen ingress in the main vessel.

LFR: One can have early containment failure (γ mode), due to the production of hydrogen even if with a low likelihood; γ mode failure results as combustion of H2 and other burnable gases as CO and CO2 coming from MCCI. Finally one can have late containment failure due to over-pressurization (δ mode).

VHTR: γ-mode corresponds to combustion phenomena in the containment building potentially leading to its early failure. Potential combustion phenomena for VHTR are connected either to graphite fire or to hydrogen combustion. For both cases, the phenomenon leading to γ-mode failure of the containment might however be rather slow due to the air availability in which case failure might take time and would rather correspond to a so-called δ-mode.

<u>ε-mode (Cavity / sumps / core catcher if any):</u>

- Base mat (chemical content) / corium flooding potential (?) / sump volume (if any);
- Presence of a core catcher (+ potential sacrificial materials: type and mass) in the case of corium-concrete interaction, water and CO_2 will be released thus increasing the amount of potential explosive species.

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Base mat	not considered*	concrete	concrete	n/a
compounds	not defined	n/a		n/a
inventory (kg)	not defined	n/a	n/a	n/a
Core catcher	internal core catcher : steel grades Z2 CND 17 12 Az Co	Yes (ceramic crucible)	no	n/a

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
Sacrificial materials	no	yes (to be confirmed)	n/a	n/a
inventory (kg)		n/a		n/a

Table 6: basemat and core catcher features

* Not considered in the PSA, if the core catcher fails, the base mat too.

SFR: A core catcher is foreseen in the vessel, which facilitates the cooling of itself and of the melting fuel (with sodium). The core catcher had a large capacity (practically the entire core) which allow to ensure the under criticality of the melting fuel.

GFR: A core catcher is foreseen in the GFR preliminary design in order to increase the level of prevention of containment failure for severe accidents. To date, the design features and materials involved in the core catcher (especially for the crucible material) are under investigation in the CEA.

LFR: After vessel rupture we can have failure of the containment due to MCCI (ε mode)

VHTR: ε-mode doesn't seem to be relevant here as no core melt is to be expected although some damage to the basement should surely be caused by heating.

The table below summarizes this discussion.

	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)	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	(identical as LWRs, even if containment and related systems are not well known), IS-	Steam Generator Tube Rupture, Containment Isolation failure	Identical to LWRs because of the thermal loading of the IHX (failure of the isolation valves)

	SFR (EFR)	GFR (CEA design)	LFR (ELSY)	VHTR (ANTARES)
		LOCA (IHX, DHX tube rupture) combined with the containment isolation failure, HSS failure		
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 Building	Over pressurization in containment building	Dust explosion (or α-mode ?)
ε-mode	Corium / Concrete Interactions	FCI	Molten Core Concrete Interaction	Not relevant

Table 7: main containment degradation modes

2.3.3.2 Identification of specific and important containment degradation processes (phenomena + damage criteria) related to the aforementioned reactor concepts:

If corium coolability is not achieved, the containment will eventually fail by base-mat melt-through. For existing plants, accident management measures are taken to make this outcome as unlikely as possible. For future plant, design provisions are taken to avoid or at least minimize MCCI (Molten Core Concrete Interaction).

- **SFR:** As described in § 2.2.3, in EFR, the primary containment is formed by the primary vessel (steel), the roof (steel), the components seals, the external primary sodium purification loop, the primary cover gas circulation and purification system. The secondary containment is formed by the reactor building (reinforced concrete) and its base mat, the walls of the secondary piping chamber inside the reactor building (reinforced concrete), the polar wall facing the secondary sodium pipe chambers (reinforced concrete), the leakjackets around top of integrated heat exchangers (Direct Heat eXchanger, Intermediate Heat eXchanger) and secondary sodium pipes above the roof and the connections of these leakjackets to the polar wall in order to ensure the continuity of the containment, the tubes of the IHX and DHX. The above roof area is protected by a liner which avoids interaction between sodium and concrete. The volume of this area is about 5000 m3. The volume allows limiting the O_2 inventory in case of sodium fire in this area. The shape of the secondary containment is rectangular.

- ✓ *Free volume / geometry and compartmentalization (if defined):* The release ways of radiological products are described on Figure 5;
- ✓ *Operating & design (limit) pressure / temperature:* The primary containment is designed to resist to a mechanical energy release of 700MJ. The temperature limit on the reactor vessel is about 530°C. The temperature limit on the concrete is about 100°C and the ultimate pressure to ensure leaktightness of the secondary containment is 250mbar;
- ✓ *Containment engineered systems (cooler capacity / spray…):* There are only containment isolation systems;
- ✓ *Presence of specific devices to avoid explosive atmosphere in the containment building (e.g. recombiners, igniters):* There are no recombiners or igniters;
- ✓ *Containment penetrations (piping, electrical) and components or systems located in auxiliary buildings: please refer to the* β-mode section;
- ✓ *Strategy of containment gaseous content relief (before filtering) vs. pressure peak (e.g. VHTR):* The strategy consists in releasing gases to buffer tanks, then in the buffer tanks rooms and above the roof, and finally to the crane hall. For the long term, it is envisaged a retention chamber.

Accidents which could impact the containment (except hazards developed in § 2.4) considered in EFR are:

- ✓ Sodium fires;
- ✓ Hydrogen risk.

The risk of hydrogen production in the above roof area is practically eliminated by the thermal insulation of the concrete protected by a steel liner. Nevertheless, in case of a large sodium fire on the reactor roof, high temperatures are likely to be reached in the crane hall concrete which are not insulated. Therefore, this accident could lead to water release from the concrete and thus could lead to a sodium/water reaction which could itself lead to hydrogen and soda (sodium hydroxide) production. However, the hydrogen would be consumed in the flame as fast as it is produced.

Sodium Interaction with Concrete: At low temperature, liquid sodium reacts with the water of concrete. At high temperature, liquid sodium reacts directly with the concrete. Ordinary limestone concrete is traditionally used in civil works but is not suitable for the anchored safety vessel option because there is a possibility of a severe sodium reaction with it. Sodium resistant concrete is therefore used to make an interface between the safety vessel and the rest of the vault made of ordinary limestone concrete. It is also required that the material should be easy to pour in site.

- **GFR:** Even if the containment building is not yet designed for the 2400 MWth GFR, some features could nevertheless be exhibited:
  - ✓ For containment integrity, one of the design requirements is that it could be able of withstanding the overall helium volume (and nitrogen one initially present in the close-containment) and the energy egress associated with the depressurization accident following a concomitant rupture of RCS and of the close-containment;
  - ✓ For FPs behaviour, a containment building comprising an inner steel liner and outer concrete shell would provide a leak tight barrier for activity release and maintenance of a maximum pressure at equilibrium ranging from 2 to 3 bars. A vented and filtered containment building is proposed to limit the potential release under severe accidents;
  - ✓ The implementation of a core catcher in the containment building (or in the close-containment) is required to avoid the Molten Core – Concrete Interaction (see specific provisions detailed here-under).

    Then, *Free volume / geometry and compartmentalization (if defined):* The release ways of radiological products are described on Figure 5.
  - ✓ *Operating & design (limit) pressure / temperature:* studies are underway in order to assess the design limits of the containment;
  - ✓ *Containment engineered systems (cooler capacity / spray…):* There are "classical" containment isolation systems, supplemented by a venting/filtering system. Regarding the implementation of a spray system, this option should be investigated if its effect for source term decrease is demonstrated (especially for fuel aerosols);
  - ✓ *Presence of specific devices to avoid explosive atmosphere in the containment building (e.g. recombiners, igniters):* to date, according to the preliminary design and related studies for GFR, it is not intended to implement recombiners or igniters in the containment building;
  - ✓ *Containment penetrations (piping, electrical) and components or systems located in auxiliary buildings: refer to the* β-mode section.
  - ✓ *Strategy of containment gaseous content relief (before filtering) vs. pressure peak:* The containment atmosphere should be filtered before any relief into the environment.

LFR: Although, up to date, it is not yet well defined, the containment shall be designed to perform the following safety functions:

- Protection against external hazards.
- Confinement and control of radioactive products.
- Biological shielding.

The reactor building envelope forms the primary containment. The secondary containment, within the primary containment, covers the area in which a certain number of penetrations exist.

The primary containment shall be made of reinforced concrete and a metallic liner shall cover the 100% of the internal surface. It is designed to withstand the double-ended rupture of one main steam manifold.
The primary containment is connected to:

- Control and service buildings by means of electric and ventilation penetrations.
- Radwaste building by means of the equipment and spent fuel outlets.
- Fuel building by means of the fresh fuel inlet.
- Turbine building by means of the steam tunnel.
- Auxiliary building by means of piping penetrations.

The secondary containment envelope shall be made of reinforced concrete.
Containment systems enclosing the reactor are provided for the retention of radioactive products within a circumscribed envelope, and support the safety-grade DHR (i.e. through a passive or active reactor vessel air cooling system). The containment systems perform their specified functions in concurrence with the most critical accident, which is anticipated to occur. Active parts of the containment systems (for example, containment isolation system) are redundant and are sufficiently independent from the systems whose malfunction belongs to the initiating failure sequence.

- **VHTR:** this item is not relevant for VHTR

2.3.4 SPECIFIC PROVISIONS FOR PREVENTION AND MITIGATION OF SEVERE ACCIDENT CONSEQUENCES

In both PWRs and BWRs, several provisions are used in order to limit the consequences of Severe Accidents. One might list for PWRs as an illustration: the Containment Spray System (which is aimed at reducing the containment pressure and removing the decay heat and also to enhance the FPs aerosols deposition in the containment building) and the Hydrogen Control by the use of igniters or catalytic recombiners.

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

- A supplementary shutdown system (chapter 2.3.4.1),
- a specific design of core assembly to promote the corium spreading and local recovery of cooling path (chapter 2.3.4.2),
- A core catcher (chapter 2.3.4.3),
- Some containment’s engineered safety features (chapter 2.3.4.4),
- Means / systems of ultimate heat sink (chapter 2.3.4.5),
- A severe Accident Management strategy (chapter 2.3.4.6).

2.3.4.1 Supplementary shutdown system

A 3rd shutdown system is implemented on some fast reactors and it could be self-actuated (passive device not only for the rod insertion but also without any signal elaboration ; the actuation of the system is triggered by the effects induced by the transient, like material dilatation in case of overheating of the coolant for instance as described below for the CREED) according to some GEN IV projects.

SFR: This group of elements of the third shutdown level consists of passive and active measures being capable of bringing the reactor to a safe condition in case of postulated failures of the two basic shutdown systems and of other features. Two different types of failure of the basic shutdown function have been considered:

- failure to de-energise the scram magnets,
- failure of rods to drop into the core.

Failure to de-energise all electromagnets (CSD and DSD) is minimised by diversity and redundancy of the trip systems. Because the magnets of the DSD rods are located under sodium, the concern of mechanical failure due to blockage by sodium aerosols is definitely ruled out. Principally, only rod jamming in the core remains as a common cause of mechanical rod failure. Regarding the high degree of diversity of CSD and DSD rods and the extreme misalignment and tube deflections which can be tolerated in particular for DSD, this type of failure is judged to be of minor relevance compared to failure of magnet de-energisation. Each failure type comprises different detailed failure modes, but these details do not affect the two principle third shutdown functions:

- to disengage the absorber rods so that they may fall into the core (CREED),
- to mechanically assist the insertion of the absorber rods (BRI).

SADE system: The SADE system would passively terminate the power supply to the DSD electromagnets after a loss of primary pump electrical power supply, if the trip signals had failed to initiate rod drop. The DSD electromagnets are electrically fed by a generator which is driven by a motor provided with a flywheel. The design of the flywheel is such that the gravity drop of the DSD rods occurs in less than about 10 seconds. This value is consistent with the halving time of the primary pump coast down and is small enough to avoid boiling in case of loss of station service power (LOSSP) combined with a failure of both shutdown systems. The SADE system is not effective in case of primary pump coast down by causes other than LOSSP.

Delatching by Control Rod Enhanced Expansion Device (CREED): CREED is a passive mechanism allowing to the control rods insertion in response to core outlet temperature increase. It essentially consists of an element which provides an increased thermal expansion on the rod driveline. At a certain threshold of expansion a delatching mechanism initiates passively rod release. This mode of shutdown is fully independent from the reactor trip systems.

CREED characteristics are specified such that coolant boiling is prevented for slow ULOF events. The most important data of CREED are:

- The threshold for delatching of about 590 - 600°C;

- A time delay of the enhanced expansion related to the core outlet temperature of about 14 s at nominal flow and increasing with decreasing flow.

CREED is implemented on each DSD rod. If the core outlet temperature increases a thermal expansion device comes into contact with the lower part of the hot electromagnet, and leads to the mechanical disconnection of both parts of the hot electromagnet. Then, the DSD rod falls.

Bulk Rod Insertion (BRI): This active measure provides shutdown by the motorised insertion of absorber rods. There are two types of BRI initiation, each activating CSD and DSD motors of both rod groups (RG) which have separated electrical supplies:

- BRI1 is triggered by the plant protection systems,
- BRI2 is triggered by the reactor trip systems.

In case of jamming of the rods, BRI provides drive-in forces that are much higher than gravity and only limited by the strength of the drive mechanisms. The maximum insertion speed of the CSD rods is the same as for control actions, namely 1 mm/s. There is ample flexibility to choose the DSD speed as large as required to prevent boiling in case of ULOF for each type of failure of the basic shutdown function.

Rod Disconnection Initiated by the Mechanical Stroke Limitation Device: The primary purpose of the Stroke Limitation Device (SLD) is to mechanically terminate the withdrawal of the faulted rods and so reduce the risk of fuel melting to an acceptably low level. In this regard, SLD is a design feature to alleviate unprotected transients. Conditions in the affected fuel pins may further deteriorate after termination of the rod withdrawal. SLD has, therefore, the additional function of rod release should the stroke limit be reached. A means of repositioning the stop for compensation of burn-up reactivity is necessary.

At least, one should mention the vessel design which (at least in the past) was taking into account a mechanical load due to the CDA. Similar approach has been developed for DHR function. This leads to implement additional DHR systems (DHRTV and SGOSDHR) in complement of the systems required by the safety analysis. For instance, different devices are designed to cope for DHR. For SPX, the situation is such:

- In case of reactor normal shut-down, through the steam generators,
- If steam-generators are not available, through sodium-air exchangers,
- If the four secondary loops are lost, through four sodium circuits (called RUR). If the 4 circuits are operative in natural convection conditions, they should be sufficient for residual power evacuation,
- Two water loops installed inside the reactor pit (i.e. RUS, in the SPX terminology).

- **GFR:** As for SFRs, and because of the risk of CRA removal (withdrawal or ejection), this event will be prevented by redundant and diversified monitoring systems complemented, by instance, by mechanical devices limiting the movement of individual CRAs (i.e. Stroke Limitation Device).

If required, by the light of L1PSA results and in particular owing to the contribution of the loss of reactivity control for CDF, a 3rd shutdown system or device could be requested to act in a passive way (low-temperature metallic fuse for instance). For "CRA ejection" concern, the prevention is mostly insured by the implementation of CRAs mechanisms (under the fissile region), and systems able to stop the CRA in case of sudden depressurization of its housing. In general, for scenarios potentially leading to reactivity insertion (water ingress), and according to the contribution of such events determined by the L1PSA results, a third shutdown level could be required (e.g. passive systems).

- **LFR:** the shut-down system shall meet diversity, reliability, and performance requirements (shutdown margin, drop time). On the whole the absorbers have been organized according to two different concepts, arranged into three independent systems, for a triple redundancy. Every set moves in empty (Argon) box beams and almost satisfies the postulated anti-reactivity. The worth of the first system, made by 8 massive conventional absorbers devoted to the scram (by gravity) and refuelling, is some 5800 pcm. The second system, made by 32 Finger Rod Absorbers (FRAs) used only for scram, yields about 2700 pcm. The third system, made by 38 FRAs has a double scope since a subset has the regulation and in addition also compensation duties. They have been mainly positioned between the intermediate and the outer fuel zones to maximize their effectiveness: for cycle swing compensation they provide the needed anti-reactivity by a 30 cm insertion into the active zone. Their complete insertion has been evaluated in further 2500 pcm for the reactor shutdown.

- **VHTR:** not relevant

2.3.4.2 Specific design of core assembly to promote the corium spreading and local recovery of cooling path

Specific design of core assembly to promote the corium spreading and local recovery of cooling path (especially for SFR, for criticality and heat removal concerns in order to limit the core degradation): localized in the core region, this mitigation device is determinant for the assessment of energetic core degradation and induced effect on containment structure.

- **SFR:** One should mention here the Japanese FAIDUS system. Previous reactor designs (SNR 300, SPX, Monju, CRBR) have all taken into account some mechanical energy load due to a HCDA in order to mitigate its short-term consequences. The phenomenology of the so-called transition phase is however so complicated and subject to so many uncertainties that a safer way would be to "practically eliminate" the possibility of a transition phase. Theoretically, there are two possibilities to ensure this: one is through “core dilution” and one through “core relocation”. Japanese scientists are working a lot about this last solution. The principle would be to create inside the core some ducts acting as escape paths for molten core material relocation (called “FAIDUS” for Fuel Assembly with Inner Duct Structure).

- **GFR:** By now, any specific provision was intended to promote the molten fuel spreading in the reference design of the GFR core.

- **LFR:** As far as the core relocation process is concerned, lead fast reactor presents a reduced risk of blockage formation by the adoption of a large pin pitch. In fact it is possible to design fuel assemblies with fuel pins spaced further apart than in the case of sodium and hence with a large coolant fraction as in the case of the water reactor, with associated improved heat removal by natural circulation.

- **VHTR:** The two shut-down systems implemented are considered sufficient for this kind of reactors.

2.3.4.3 Core catcher

- **The core catcher implementation** (a core catcher is foreseen to collect the core materials. Both its location inside or outside the core vessel and its composition are subjects under investigation among specialists. Collected material recriticality are of specific concern.
  - **SFR:** The internal core catcher was designed to cool, contain (mechanically) the corium and ensure its under-criticality. It is cooled by the DHR systems. In order to have a good coolability of the corium, corium needs to be correctly spread. The surface of the core catcher is designed in this aim. The spreading of the corium allows ensuring under-criticality.

  - **GFR:** For the GFR, the implementation a core catcher in the containment building (or in the close-containment) is foreseen in order to ease the corium freezing and to avoid potential recriticalities of the molten fuel after the main vessel rupture. One of the principal constraints for the design of this component is related to the absence of a liquid coolant, which can absorb the upward-flowing heat from a molten hot pool. By now, the work is in progress to evaluate alternate concepts derived for former studies of fast-breeder reactors (and gas-cooled ones in particular). Among these alternate concepts, the main candidates that could provide a high stored heat capacity are based on a ceramic crucible that utilizes a build-up of either refractory materials, steel boxes filed with borax ($Na_2B_4O_7$), heavy metal (e.g. depleted uranium) or steel.

The potential criticality of the corium pool is managed by a sufficient spreading of the amount of discharged materials.

- **LFR:** No core - catcher foreseen.

- **VHTR:** No core-catcher foreseen.

2.3.4.4 Specific containment engineered safety features

Containment's engineered safety features:

Within this topic, systems involved for containment heat removal (i.e. fan coolers or spray system), devoted to avoid explosive atmosphere (i.e. recombiners/igniters) or those for venting / filtering the containment atmosphere are described.

- **SFR:** The liner of the above the roof area protects against sodium/concrete reactions. In addition, a retention chamber was provided to be used in case of CDA. Its function was to decrease the pressure of the crane hall in order to limit radiological releases through containment leaks.

- **GFR:** As aforementioned, a vented and filtered containment building is proposed to limit the potential release under severe accidents. As a result of an accident sequence including core meltdown, there may be a continuous rise in the containment pressure (e.g. at vessel rupture, at successive corium spreading in the containment, by heat-up of the containment atmosphere by corium radiative exchange), which could finally lead to its failure after a certain period. To protect the containment against over-pressurization, a controlled pressure relief system would be designed. This system should also be able to filter the containment atmosphere in order to keep off-site doses within regulatory limits (in the frame of Design Basis Accidents). Several filters technologies (e.g. high efficiency particulate absorbing filters, iodine absorption units, charcoal filters…) should be "tested" according to their capacity limits in terms of allowable gas flow rates and performance for GFR proto-typical FPs and aerosols retention in representative severe accident conditions. To date, the work related to this topic was not addressed for the GFR. Regarding systems devoted to avoid explosive atmosphere (i.e. recombiners/igniters), preliminary (but roughly conservative) evaluations showed that it would be very unlikely to form detonable and even flammable mixtures by hydrogen or carbide monoxide formation. If more refined calculations would exhibit other trends, mitigation provisions should be foreseen.

- **LFR:** As regards the containment failure modes and degradation, specific provisions against the consequences of steam generator tube rupture are foreseen, with the aim to prevent the effects due to steam explosion and wave propagation. In addition, in the case of design basis event of steam generator rupture, the following measures are foreseen:

- safety valves to prevent the reactor vessel from exceeding the reactor vessel design pressure (< 3.5 bar);
- double wall perforated casing of the steam generator or excess-flow valves on steam feeding tubes of the SG.

- **VHTR:** The whole VHTR concept is by itself a mitigating (or preventing) system. Now, two additional specific devices should be considered as playing a fundamental role in severe accident mitigation:
  - the primary coolant clean-up system (or helium purifying system) is a device devoted to purify the cooling gas during reactor operation in order to limit the primary circuit radioactive inventory (which is a major contributor to source term). Due to absorption phenomena, a FP clean primary circuit remains out-of-reach;
  - the Reactor Cavity Cooling System (RCCS) is a water-cooled circuit used as the ultimate heat sink to evacuate the core heat in severe accident conditions.

  Two other elements may play a central role in accident mitigation although it's not sure such systems would be present on ANTARES concept. These are:
  - the CACS (as it was named for the HTR-1160 project) or Core Auxiliary Cooling System, a safety system to cool the primary circuit in case the coolant flow is stopped;
  - the existence of some device (as a liner) to make the containment a confinement (that's to say to reduce the containment leakage as much as possible).

2.3.4.5 Means / systems of ultimate heat sink

The ultimate heat sink is generally speaking the sea, river, lake or outside atmosphere.
Their operating mode could be passive (by natural convection of air e.g RCCS for V_HTR, by radiative heat exchange with containment atmosphere or by conduction/convection processes like in a core catcher) or active (e.g. by convection with a fluid system in the core catcher);

- **SFR:** The normal DHR systems are designed to resist to the HCDA and to cool the corium when it is on the core catcher.

- **GFR:** For the early phase of an accident scenario, the heat removal will mostly rely on dedicated reactor systems that were designed to avoid or limit the core damage. The ultimate heat sink is therefore made of sufficient water capacities which autonomy is around 24 hours after IE. A refilling procedure should be put in place in order to ensure the long-term heat removal. After a potential corium discharge in the containment building, the heat removal in the ceramic crucible will rely on convection (inside the corium bath) and radiative exchange processes, with the supply of sacrificial materials able to increase the performance of these phenomena. For the containment building, a controlled relief is aimed at reducing the temperature level of the gaseous atmosphere.

- **LFR** The core is designed in such a way that sensible fuel heat and decay heat, following reactor shutdown, is transferred to the ultimate heat sink by means of passive systems through reactor coolant natural circulation (no active component is credited).

- **VHTR:** The VHTR design features three cooling systems. The Shutdown Cooling System (SCS), a non safety-related system designed to remove heat from the (Reactor Core System) RCS and transfer that heat to the ambient air. Its first circuit is in parallel with the plant normal Primary Heat Transport System (PHTS) across the RCS and consists of a helium-to-water heat exchanger, an electrically powered gas circulator and a shutoff valve. Its second circuit is a closed pressurized water heat transport loop that runs from the helium-to-water heat exchanger to a water-to-air heat exchanger. The water is circulated by conventional electrically powered pumps, and the ultimate heat sink (third circuit) is an air-blast type heat exchanger with electric fans. The SCS can operate even if the secondary circuit and the primary forced helium circulation are not available. SCS is designed for achieving this function in pressurized and depressurized conditions. The Reactor Cavity Cooling System (RCCS) is a safety-related passive water cooling system for decay heat removal during emergency cool-down, for cavity heat removal during normal plant operation and for confining of radioactivity released into the reactor cavity during normal operation. The RCCS consists of two independent and redundant trains operating in natural circulation. Each train consists of the following four major components, plus associated pipes, headers and valves, all located inside the reactor building and the reactor auxiliary building:
    i) a panel wall cavity cooler, consisting of alternating vertical pipes around the periphery of the RPV (a compact air-to-water heat exchanger that surrounds the RPV);
    ii) a water storage tank (a water-to-water heat exchanger is inside and integral to the pressure boundary of the water storage tank);
    iii) a water-to-air heat exchanger (closed circuit cooling tower); and iv) a circulating water pump.

2.3.4.6 Severe Accident Management strategy

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

References chapter 2.3.4

[LFR 2.3.4_1]: Design Objectives, requirements and general specifications on ELSY, ELSY Project, DOC/07/04. January 2009.
[SFR 2.3.4_1]: The result of a wall failure in-pile experiment under the EAGLE project. Konishi & al. Nuclear engineering and design 237, 2007

2.3.5 IMPORTANT PARAMETERS FOR L2PSA RELATED TO THE SOURCE TERM EVALUATION

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

- Inventory of radioactive materials in the fissile region (at EOL) according to the nature of fuel, maximum burn-up, content of MAs… (i.e. isotopic vector);
- In-vessel radionuclide release and transport mechanisms (related to fuel/cladding and coolant natures);
- Retention and deposition of fission products inside RCS;
- Chemical species (e.g. organic or non-organic iodine…)
    - Iodine and caesium chemistry (Affinity of these isotopes with coolants involved);
    - Chemistry of other isotopes (Te, Sr…): knowledge regarding phenomenological trends in presence of helium, lead or sodium;
- Activation and corrosion products;
- Ex-vessel radionuclide release and transport (related to containment type);
- Aerosols behaviour inside the containment;
    - Deposition and re-suspension of aerosols mechanisms;
    - Effect of energetic phenomena on in-containment fission product behaviour;
    - Activation and corrosion products of concrete surrounding the core vessel (if any) and air of its cooling system;
- Radionuclide release outside the containment (i.e. Source Term);
- Tritium;
- Potential for FPs scrubbing;
- Additional barriers or structures (retention tanks, close-containment) for radionuclide (e.g. close containment for GFR is not considered as confinement barrier but could lead to a potential of FPs retention).

SFR: Sodium is very reactive, particularly with iodine which will be retained in the sodium. In case of HCDA, the fraction of fission products released into the cover gas as defined in the Plant state III (see Table 4) is :

- noble gases : 100%
- volatile fission products : 1 %
- solid fission products : 0.01%

GFR: One of the main concerns regarding the source term for reactors involving fast neutron spectrum is related to the high Pu content (i.e. around 20% for the GFR core, nearly constant through the fuel lifetime due to the targeted zero breeding gain). If one assumes that a maximum of 1% of plutonium aerosols will be released in case of whole core meltdown, a substantial increase of the source term (in the containment building and by consequence in the environment) could be expected. The following tables are providing a representative inventory of radioactive materials that could be encountered for a GFR core at EOL (100 GWj/t) with a mean initial Pu enrichment of 20% and 2% of MAs.

		inner fissile zone	outer fissile zone
0 GWj/t	**Pu**	**18.20%**	**21.40%**
	MAs	**2%**	**2%**
100 GWj/t	Pu8	4.35%	4.29%
	Pu9	49.53%	47.71%
	Pu40	31.19%	31.93%
	Pu41	5.63%	5.89%
	Pu42	9.31%	10.19%
	enr. Pu	**20.65%**	**22.28%**
	Am	68.09%	74.34%
	Cm	20.55%	14.47%
	Np	11.36%	11.19%
	MAs	**1.91%**	**2.10%**
	mFP/mtot	**11.50%**	**8.70%**

FP	**core average content at 100 GWj/t**	
	in ppm	**in %**
Xe	13172	1.38%
Kr	648	
I	833	0.09%
Br	42	
Cs	11064	1.16%
Rb	581	
Te	1780	0.22%
Sn	267	
Se	129	
Ba	4356	0.57%
Sr	1382	
La	3341	5.39%
Ce	6538	
Pr	2886	
Nd	10150	
Y	716	
Zr	7701	
Nb	38	
Mo	8854	
Tc	2194	
Ru	9020	
Rh	2470	
Pd	6789	
Ag	609	
Cd	512	
In	31	
Sb	86	
Pm	507	
Sm	2785	
Eu	375	
Gd	297	
Tb	15	
Dy	18	
total	**100186**	**10.02%**

Table 8: Preliminary assessment of FP inventory for the 2400 MWth GFR at EOC

In the frame of the GIF targeted safety goals, it was expected that new nuclear systems would eliminate the need for offsite emergency response, through design and application of advanced technology. Therefore, it will be required to accommodate this assumed plutonium release through specific dispositions as increasing the site boundary distances, providing larger filtration systems or eventually using spray systems. As the first two would either limit the potential sites available or increasing the investment cost (e.g. power requirements for filtering systems, component sizes…), a sizeable dose reduction could be obtained by the use of a spray system.

Compared to LWRs for which the spray system plays several roles (i.e. to condensate the vapour in the containment building, to control for containment pressure avoiding its mechanical rupture, to refill the RCS for water inventory), one of the main aim for implementing a spray system in the GFR will be to increase the FPs deposition phenomenon and to limit the pressure into the close containment by cooling its outside walls (and automatically decrease the gaseous release of FPs into the environment). However, the main drawback of this solution is that it introduces large quantities of water into a "dry" containment (option b for GFR). If a substantial R&D work is performed to demonstrate the adequacy of spray in "removing" fuel aerosols (especially Pu and MAs), this system could easily be implemented without risk of potential re-criticalities with the discharged corium, as far as a core catcher is implemented in the close containment (option a). This variant could be envisioned thanks to the presence of a close-containment in the GFR concept (which primary goal is to keep a back-up pressure for forced and natural convection of gas in case of LOCA).

By consequence for the GFR, the FPs chemistry would be very dependent with the choice of the containment concept:

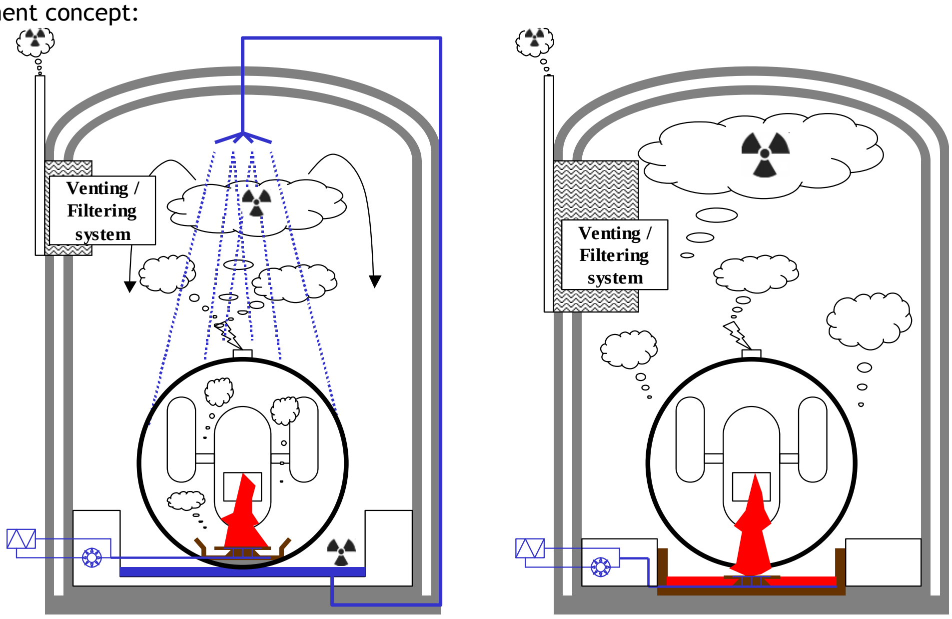

Figure 17: tentative sketches of GFR containment features for FP retention enhancement

Another specific point for helium-cooled reactors (VHTR and GFR in particular) and source term assessment could be associated to the production of tritium by helium-3 neutronic capture (especially due to its high absorption cross-section), which is however at very low concentration. In addition, high-energy neutrons irradiating boron-10 (boron carbide being the reference material for CRAs) will also occasionally produce tritium. One major feature of the gas-cooled reactors is that a helium purification system is intended to remove the major part of impurities or activation products that are present in the coolant, and therefore account for the tritium question (in particular for the potential β-mode).

LFR: In case of severe accident, source term and likelihood of radionuclide release from the containment, should be positively affected by the capability of lead of trapping fission products and high shielding of gamma radiation (low dose).

The pool scrubbing is not expected within the plant configuration here considered, as no direct pathways of airborne radionuclides through water reservoirs are foreseen.

VHTR: The HTR has the following potential sources of radioactive materials:

a) Sources within the main power system pressure boundary:

- The fuel in the core within intact coated particles (fuel coated by several Pyrolitic Carbon (PyC) and Silicon Carbide (SiC) layers)
- The radio-nuclides issuing from failed fuel particles, natural uranium contamination of the graphite components and activation:
  - Embedded in the graphite or graphite dust (circulated or plated out),
  - The plate out on reactor coolant pressure boundary surface,
  - The circulating activity by core coolant.

b) Sources outside the reactor coolant pressure boundary:

- The spent fuel in fuel handling and storage systems,
- Coolant purification and storage system and connecting piping,
- Solid and liquid radwaste systems.

The main barriers of each of these radiological sources are indicated below:

1. Fuel in the core:
   - Fuel particles;
   - Partial retention in graphite matrix;
   - Reactor coolant pressure boundary (RCPB);
   - Containment structure-system;
2. Other radiological sources outside the core within the reactor coolant pressure boundary:
   - Reactor coolant pressure boundary;
   - Containment structure-system;
3. Spent fuel in fuel handling/storage systems:
   - Fuel particles;
   - Partial retention in graphite matrix;
   - Storage or transport systems;
   - Containment structure and spent fuel storage building;
4. Other radiological sources:
   - Various tanks and transport systems;
   - Containment and radwaste buildings.

The total radioactive material inventory within each of the these four categories is very different, so the required lines of defence or barriers are different in each case to maintain operational or accidental releases to the environment within specified limits. In the case of the HTR, the main radiological material inventories that can contribute to potential risks as a consequence of scenarios with potential releases of source term outside the reactor coolant pressure boundary are:

1. Circulating and stored helium coolant radioactivity including elemental and dust-borne activity;
2. Elemental and dust-borne radioactivity plated out on RCPB surfaces and embedded or attached dust in graphite components;
3. Radioactivity from uranium contamination outside fuel particles;
4. Radioactivity in failed and defective fuel particles;
5. Radioactivity in intact fuel particles.

The document [VHTR-2.3.4_1] provides an order of magnitude of the inventories of one key radionuclide ^{131}I inside the reactor coolant pressure boundary of a HTR (cf. Table 9). Inventories of ^{131}I can be considered representative of the radioactive material inventory as determined in previous HTR PSAs, although some uncertainties have arisen regarding the amount of other solid radiological inventories, e.g. Cs, Ag, Sr in dust, highly dependent on operational behaviour.

Component of Inventory	^{131}I Curies
Circulating Activity	<<1
Plate-out on internal RCPB surfaces	<1
Uranium contamination outside coated particles	~100
Failed and defective coated particles	~500
Intact coated particles	~1 x 10^7

Table 9: Comparison of ^{131}I inventories

The significant part of radioactive material inventory is located within the intact coated particles of the fuel. The main difference of HTR fuel in comparison with fuel used in the Light Water Reactors is their intrinsic robustness and strength in normal as well as accident conditions.

The HTR fuel design is aimed on the very low probability of releasing a significant amount of radioactivity up to the safety temperature limit of 1600°C. Below the temperature of 1200°C the main reasons for the emission of radioactivity are coating particles with defective layers or contamination during the fabrication process. Up to 1600°C the probability of release increases due to diffusion processes. Above these temperatures the layers become more porous and can be breached as a result of pressure build-up of gases.

Based operational experience and tests with similarly manufactured fuels, the main uncertainties associated with the potential leak of intact coated particles are related to:

1 To the reliability/quality of the fuel manufacture process, and;

2 The avoidance of environmental conditions that could potentially damage the intact coated particles (e.g. air ingress).

The HTR design is focused on prevention, by means of design measures, the achievement of any feasible accidental scenario where such potential process or environmental conditions would be able to challenge the integrity of coated particles. These design measures are inherent features to the technology (e.g. strong negative temperature coefficient of reactivity, large thermal heat capacity) or design options (e.g. air or water ingress avoidance). Moreover, the safety philosophy in HTR technology is strongly supported in the extensive use of passive safety features more than active safety features.

Based on these two design principles, in any scenario the following considerations should be taken into account for the characterization of potential accidental radioactive releases:

- a breach of reactor coolant pressure boundary permits in short term the release of the circulating activity and certain inventory of helium purification system into containment structure (no lift-off, no dust).
- a large break in the reactor coolant pressure boundary is capable of producing large shear force ratios during the blow down and consequently cause the release into containment structure of plated-out radionuclides and dust embedded or attached in graphite components, and the inventory of helium purification system.
- for scenarios with a loss of forced circulation of the core coolant; the core temperature increases slowly. The thermal transient could lead to a slight decrease of confinement capability of fuel particles and therefore to delayed fuel radiological releases into the containment structure.

The delayed fuel release is associated with the slow release of part of the inventory in any failed or uranium-contaminated fuel particles in regions of the core that experience an increasing temperature transient several days after the initiating event. This is a condition that can be met only for small regions of the core and only when there is a sustained loss of forced core cooling. Peak core temperatures decrease with time for any pressurized or depressurized condition with continued forced circulation cooling.

Since the main contribution to radiological releases in accident scenarios is the result of delayed fuel releases and these can take place only after large time periods, possible mitigating strategies can be planned and performed without the typical time constrains of LWRs scenarios.

Potential long-term reactor coolant pressure boundary air ingress effects can be limited or avoided to minimize their potential impact on released radiological inventory.

In the short term, radiological releases from accidental scenarios are limited to circulating coolant activity and in the mid term to lift-off of plated-out and dust radionuclides. Despite some uncertainties in source term definition these short-term radiological inventory releases are much lower than the long-term delayed fuel releases.

References of chapter 2.3.4

[VHTR-2.3.4_1]: PBMR white paper entitled "Probabilistic Risk Assessment Approach for the Pebble Bed Modular Reactor", Revision 1.

2.4 TREATMENT OF HAZARDS

A number of hazards can cause loss of containment integrity among which one may list:

- Internal missile,
- Jet effects
- Pipe whip
- Leakage/LOCA
- Internal flooding
- Dropped loads
- Internal fire
- Asphyxiate and toxic gas release (dust)
- Gas/chemical explosion
- Hot and cold gas release
- Sound, vibrations
- Graphite dust explosion
- Aircraft impact
- Vehicular impact
- Sabotage
- Transport, industrial activities (fire, explosions, missiles, toxic & asphyxiant gases, corrosive gases)
- Electromagnetic interference (EMI).

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

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

- The seismic event: an earthquake can potentially cause structural failure of the containment or its penetration. Earthquake is also a great concern for fast neutron spectrum reactors in particular due to the core compaction and the resulting criticality risks.

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

2.4.1 RADIOLOGICAL RISKS

2.4.1.1 Treatment of hazards - case of SFR

- **Internal hazards:**

1) Fire: two types of fires are generally considered, i.e. conventional fires and sodium fires.

- For conventional fires, prevention, separation/segregation and detection arrangement were classically foreseen.
- For sodium fires, it depends of the sodium nature (primary sodium, secondary sodium in the reactor building and secondary sodium in the SG building). For secondary sodium leaks, fires in the SGB are within the design basis. The main goal is to ensure DHR since it is the safety function most vulnerable to the fire consequences by direct failure or AHX clogging. Otherwise the safety goals for sodium fires are similar to those for conventional fires. Primary or secondary sodium fires within the reactor building are beyond design basis, and large secondary sodium fires will be studied as limiting events to demonstrate that there are no cliff edge effects. The main safety goals for sodium fires on the reactor roof is the maintenance of the shutdown and DHR safety functions and the maintenance of the structural integrity of the reactor building.
- Applying the principles of fire protection to sodium fires requires some special considerations. The strategy consists of firstly preventing sodium leaks, which is a matter of quality assurance during design, construction, fabrication and operation and the exclusion of any damaging impact. Secondly, if a leak occurs, the strategy is to contain the leakage and prevent oxygen supply. This may be achieved by double envelopes or inert gas filled rooms around the sodium filled components. Complete segregation is difficult to achieve for systems like the secondary or DRC loops, whose main task is to transport thermal energy and where sodium pipes must pass through the fire barrier and must come together in the above roof area. Common auxiliaries between the secondary and DRC system also present potential segregation difficulties.
- Chemical and radiological effects of a sodium fire must be considered.

2) Hydrogen explosion: As described in § 2.3.2, hydrogen risk, in case of sodium fire on the above roof area, is avoided as the concrete is protected by a liner. The most likely scenario is a hydrogen explosion during the washing phase of components which are extracted of the primary vessel. In fact, components are washed with a water spray. Hydrogen production is monitored and extracted.

- **External hazards:**

1) Earthquakes: No site has been selected for EFR. Therefore, an envelope of all national practices was considered. Preliminary analysis considered that the effects of a Safe Shutdown Earthquake (SSE) on the sub-assemblies are acceptable in terms of differential lateral movement and reactivity insertion. The reactivity introduced by pellet tamping as a result of a SSE is expected to be acceptable. Based on Superphenix analysis it is expected to show that no clad failures will occur as a result of SSE in the EFR core. The shutdown safety function is assured during SSE by seismically qualified I&C systems. The DHR function is ensured mainly by passive design of the DRC loops to remain functional in event of SSE. The I&C equipment to open AHX flaps is seismically qualified.

2) Exceptional meteorological conditions: It includes abnormal humidity, wind, precipitations, snow and icing, fog and extreme ambient temperatures. They are considered as category 2 events. Consequences other than on building design and on some specific systems being in contact with the environment are not seen. For some buildings the design against other external events like aircraft crash will cover the requirements due to some exceptional meteorological conditions.

3) Inland or marine flooding: analysis should be performed as soon as a site is specified.

4) Aircraft crash: reactor building and reactor auxiliary building were reinforced to withstand an aircraft crash. For the Steam Generators building, R&D was identified (as for Water/sodium/air reaction) to clarify the need of reinforcement of these buildings.

5) Gas cloud explosion: for most of the buildings the design requirements are clearly specified. For the SG buildings, the decision on the need to protect against aircraft crash will be to determine the need for protection against gas cloud explosion if relevant considering the industrial surrounding.

6) Turbo-alternator missiles: Missiles arising from turbine disintegration are considered to be a category 3 event. They may be of high or low trajectory. The former will tend to fall near the turbine with a degree of dispersal: the latter are ejected close to the horizontal and will impact on structures in their flight path. To minimise the probability of a building containing equipment and structures necessary for the performance of safety functions suffering impact by low trajectory missiles, the turbo alternator axis is oriented so that any such missiles released are not naturally projected towards these buildings. Additionally, the buildings themselves are situated sufficiently far from the turbo alternator to ensure they lie outside the more probable landing area of this category of missile.

7) Lightening discharge: analysis should be performed during the detailed design in particular regarding the risk of jeopardizing the I&C system.

Particular case:

Water/sodium/air reaction in the steam generator building: In a SG building, the sodium and water areas are separated by the vertical and horizontal partitions of the concrete structure with the exception of the

SG bunker where the outer SG casing constitutes the separation between the water entry and steam exit loops. The two potential initiators of a Water/Sodium/Air Reaction in the Steam Generator Building are:

- external hazards such as aircraft crash (cf. before point 4);
- internal events in particular a large SG accident where wastage and penetration of the shell could possibly occur.

2.4.1.2 Treatment of hazards - case of GFR

Even if GFR studies are up-to-now non-site specific, several hazards are particularly feared owing to the potential they could have for a potential core disruptive accident (e.g. earthquake) and to the multiple failures of required systems they could engendered. As a basis for further studies, one should envision seismic events, fire in the containment building, and flooding (potentially caused by a climatic event).

2.4.1.3 Treatment of hazards - case of LFR

LFR: The potential impact of internal and external hazards on containment integrity as well as the dependent failures they could cause on systems needed for severe accident mitigation, including those supporting operator actions, should be taken into account: these include for instance containment structural damage due to seismic, flooding events for external hazards. Internal hazards as internal fires and flooding could challenge the integrity and functionality of safety systems and their support systems, as the hydrogen control system (igniters or catalytic recombiners), the containment venting system and the containment isolation system.

- Internal event (flood and fire) PRA generally utilizes the models generated for random internal initiators modified to include consideration of the type of flood/fire initiator, the potential for flood/fire and smoke propagation, and the impact of flooding environments/fire on both the equipment located in the flooded areas and on the operator actions. For certain new reactor designs, the flooding mediums of concern may include other fluids (e.g. liquid metal) in addition to water and steam. Internal hazard PRA includes internal floods/fires initiated during all modes of plant operation. Internal flooding /fire initiators that can adversely affect sources of radioactivity other than the core (e.g., waste, spent fuel pool) are also analyzed.
- An important aspect of flooding and other spatial-related accidents (e.g., fire, seismic, and other external event analysis) is the determination of whether failure of equipment in one or more locations can result in core damage. The evaluation of these types of initiators provides critical information on the adequacy of the spatial separation and redundancy of equipment necessary to prevent and mitigate these initiators.
- Seismic hazard analysis estimates the frequency of different intensities of earthquakes based on a site-specific evaluation reflecting recent data and site-specific information. As is the case for internal initiators, a seismic PRA includes analysis of seismic events that occur during all modes of plant operation and that can affect different sources of radioactive material at the plant site.

- The external event other than earthquakes (e.g., high winds, hurricanes, aircraft impacts, and external flooding). PRA includes consideration of random failures and the impact of the external events on SSCs and on operator actions. As is the case for internal initiators, external events are evaluated for all modes of plant operation.

2.4.1.4 Treatment of hazards - case of VHTR

In reference [VHTR-2.4_1], a comparison between hazards for LWR and VHTR is made. An overview is given in the next table. The general conclusion is that the risk impact of hazards is comparable with LWR. Two hazards have a higher risk and five hazards a lower risk. There is one additional hazard: Core / fuel chemical hazards, including dust explosions and combustible gas generation (H_2 and CO_2).

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

Table 10: Hazards overview and comparison of risk impact versus LWR

The hazards having less risk compared to LWR are connected to the cooling medium that is used (Helium instead of water) and the lower process pressure. The VHTR primary circuit is at a lower energy level which reduces or excludes the impact from internal missiles, jets and pipe whip. In case a gas turbine is present in the primary circuit the internal missile hazard could be higher, but this depends heavily on the mutual orientation of the SSC. The lesser impact of loss of coolant depends strongly on the power density and size of the core in relation to the (passive) cooling capabilities of the Safety systems. Given the nature of the primary coolant it can be excluded as source for internal flooding.

Hazards for which a higher risk is expected are also related to the primary cooling medium. The higher gas temperature will cause damage to structures (steel as well as concrete) in case of leakage. The high velocity helium flow could cause vibrations in the core and high cycle fatigue in structures.

HTR specifics to consider from a hazard point of view: Reviewing the VHTR designs (block and pebble bed reactors) numerous initiating events differ from the initiating events in an LWR. The same is true however between block and pebble bed designs. The continuous feed of pebbles introduces a spectrum of possibilities to change core geometry and (local) composition.

There are however very few new hazards. They are related to the fuel / core design used:

- Graphite fires are possible in case of air or water ingress in the primary system. Although: rapid Zirconium oxidation in steam could be type casted as fuel fire
- Dust is a potential source of a release (Although: LWR primary circuit also contains radioactive sources)
- Dust can be erosive causing failure of piping, ducting, heat exchangers, rotating equipment etc. The impact and rate depends on process conditions and design.
- The hot graphite dust released in case of leakage could result in a dust explosion outside the primary boundary.

Coupling of a reactor as heat source, source of electricity etc. to another (chemical) plant poses no additional hazards to those already evaluated as the normal existing external hazards originating from large industrial sites, shipping lanes, railways and high ways: toxic clouds, explosions, dust clouds, BLEVE's, heat fluxes, etc in LWR[1]. The possible transients that could be induced on the reactor process by the coupled external processes are part of the design process.

The possible "new" initiating event to consider in case of a direct link between nuclear plant and chemical plant is the "(partial) loss of off electrical load", "(partial) loss of normal heat sink (process heat)", "off site explosion/on site explosion" at the same time. The extend of the possible combinations of the afore mentioned events is of course site specific

[1] Coupling is in fact not an HTR specific issue. It applies to all reactor types with process links to other industrial activities.

Having more reactors on one site does not introduce new hazards. It's a common practice. One reactor is a source of already identified external hazards for the other and vice versa.

In the European Utility Requirements multi unit sites are addressed: interdependency - with a focus on not sharing safety facilities – should be avoided. Sharing of facilities should always be justified on a case by case basis.

2.4.2 OTHER RISKS

PSAs for reactors are dealing first with the radiological risk. However, substantial and specific chemical risk should be accounted for these new generation reactors (e.g. toxic gas cloud). Some elements regarding this item are provided hereafter.

Chemical risk: For all industrial activities in the Netherlands, when there is a possible risk to the public, a quantitative risk assessment (QRA) has to be performed, showing that no vulnerable objects (as defined by law) are within the 10^{-6} risk contour and the group risk target is not exceeded. The same targets (more or less) are used for Nuclear Reactors; the CDF is in fact a subrogate value. The risk is the decisive factor in the nuclear licensing process. The lethal – non nuclear – risk assessment (QRA) is performed in the same way as a PSA level 3 (identify initiators, mitigating systems, calculate release frequency and perform dispersion calculation), although in most cases a far more simple level 2 model (or non as most often the process piping is the only boundary between process and the public) is used. The whole process is put forward in extensive guidance documents and the software to use is prescribed. Incorporating a toxic cloud in the L2 seems no problem. As for nuclear source terms the release paths, energy and timing analyses are essentially the same. For most toxics the lethality functions are (well) known.

Other hazards than radiological ones have to be taken into account. For the SFR, chemical releases of sodium aerosols (created by a sodium fire) in the environment could be toxic.

We believe that the PSA study should encompass all the relevant risks related to plant operation, so that all the risk paths (chemical, toxic) should be identified and addressed. For example there is a risk of toxicity related to lead vapour, should a release happen, for LFRs.

References of chapter 2.4:

[VHTR-2.4_1] European safety approach for modular HTR, J.L. Brinkman, J. Carretero, F. Dawson, S. Ehster, F. Parmentier, ReActor for Process heat, Hydrogen And ELectricity generation (RAPHAEL) FP6, 31/5/2009.

2.5 SPECIFICS RELATED TO SHUTDOWN OR REFUELLING STATES

As all plant operation states should be addressed for L2PSA results efficient use, this paragraph is defined to exhibit the peculiarities of the selected reactors concepts as regards to shutdown or refuelling states.

SFR: During the annual campaigns to refuel the core, some of the fuel, radial breeder and absorber sub-assemblies are replaced, and radial shield elements may be rotated or shuffled. Mistakes made during manufacture, on-site inspection, and identification when handling may result in loading errors. These errors may give rise to inadvertent core criticality which should be detectable by the neutronic instrumentations located in the ACS.

GFR: To date, the process retained for fuel handling is the following: after few days of core power decrease (typically from 5 to 10 days after reactor shutdown), the core cooling is ensured at a primary pressure ranging from 5 to 10 bars and approximately 150/350°C for inlet/outlet coolant temperatures. It is worth remarking at this stage that a slightly pressurized situation should be maintained to ensure an adequate core cooling. This point is of major importance compared to refuelling conditions encountered in LWRs. The fuel sub-assemblies removal from the core region is performed thanks to the handling machine. For redundancy and diversifications concerns, a transport machine is required to handle the irradiated fuel SA from the reactor building to the fuel disposal hall. During transportation, the SA should be cooled by providing a fresh helium circulation in a forced convection mode.

LFR: The extension to cover low power and shutdown states leads to other attributes being included in the definition of the PDS, including, the decay heat level and whether the RCS and the containment are open or closed for maintenance and tests operations. For shutdown states the dominant contributors are vessel open/confinement open states. This is attributed to the high core damage frequency for the mid outage states and the disabled barriers during these states. At this pre conceptual design level all the modes of operation of the plant including start-up, operation at power, low power and all the modes that occur during plant shutdown and refuelling have not yet been established. Significant differences, with respect to full power operation, that could have a major impact on plant behaviour in severe accidents require the consideration for new plant states, to be inputted into the event tree. Some examples include operation in which the primary circuit is open (e.g. during head removal or during refuelling) or the containment is not isolated (e.g. during some refuelling operations). New attributes that could be considered in the definition of PDS for low power and shutdown PSA include therefore the status of the containment and the level of the RCS. The possibility of air ingress during severe accidents, presented during some reactor state as reactor shutdown, has to be accounted for together with the consequences evaluation. This air ingress is perceived to be of greater significance to those shutdown sequences in which the RCS is open to the containment.

VHTR: For pebble-bed concepts, the refuelling is normally operated continuously during operation as pebbles are continuously extracted from the core bottom and then either reintroduced at the top of the core or definitely extracted and replaced by a fresh pebble. Probably some specific safety issues are related to this continuous process of extracting and refilling and to the technical devices connected with this function.

2.6 REVIEW OF EXISTING L2PSA APPLIED TO SFR, LFR, HTR OR GFR

SFR: No L2PSA was performed at our knowledge for the EFR concept. However, it should be underlined that several probabilistic studies were performed in the past for the US PRISM concept, for the SNR-300 reactor and recently for the JSFR [SFR-2.6_1].

GFR: Regarding the safety approach, a large body of both analytical and experimental safety R&D under the LMFBR safety program was assumed to be directly applicable or easily adapted to GCFR fuel rod behaviour. However, the absence of two-phase coolant flow effects reduced the complexity of the accident analysis tasks and accident behaviour evaluation. Regarding HCDA, three classes have been identified which have slightly different phenomenology: ULOF (either initiated by a loss of helium circulation capability or by depressurisation), UTOP and LDHR. Given the occurrence of a HCDA, the reactor vessel provides a first line of protection against FPs release to the environment. In these former GFR concepts (see appendix 3), the PCRV assures the containment integrity by preventing the molten core penetration and the formation of missiles due to accident energetic or overheating failures. For containment integrity, one design requirement for the containment is that it could be able of withstanding the overall helium volume and the energy egress associated with the design basis depressurization accident (DBDA).
In conclusion, and according to the open documentation, it is worth noticing that the safety demonstration of former GFR concepts was mainly based on a deterministic approach and <u>that no L2PSA models were built in the past</u>.

LFR: At present, there are no studies available pertaining to level 2 PSA for LFR

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

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

It would be desirable to get those documents in case they are in free access. The PSA for the US MHTGR is publicly available [ref [VHTR-2.6_4]]. Otherwise, only a limited set of articles has been retrieved. Only one, relative to the HTR-1160 PSA, seems really useful as it gives quite a lot of details about the methodology used. The event tree above is extracted from this article. Neither the PSA of the MHTGR nor the PSA of the PBMR make distinction between level 1 and level 2. In fact they do not contain a level 1. In both PSA the event trees sequences end in a kind of plant damage states that already include part of the confinement status. There is no APET or CET. Source term release is based on a barrier analysis.

References of chapter 2.6:

[VHTR-2.6_1] Results of a German probabilistic risk assessment study for the HTR-1160 concept. Fassbender & Kröger. AIEA.

[VHTR-2.6_2] Preliminary risk assessments of the small HTGR. Everline & Bellis. AIEA.

[VHTR-2.6_3] Probabilistic risk assessment of HTGRs. Fleming, Houghton, Hannaman & Joksimovic. AIEA.

[VHTR-2.6_4] Probabilistic Risk Assessment of the Modular HTGR plant (draft), General Atomics, DOE, June 1986

[SFR-2.6_1] ICAPP'09

3 EXISTING TOOLS FOR ACCIDENT ANALYSES

3.1 EXTEND OF THE KNOWLEDGE AND POTENTIAL LIMITATIONS IN THE MODELING OF SA

For LWRs, L2PSA models were built in combination with an important R&D effort, including experiments and code development (mechanistic codes and also integrated tools), validation processes of physical models and finally uncertainties assessment for risk quantification. The situation is slightly different for other reactor concepts. This chapter aims at providing an overview of the pertaining difficulties that could be encountered for L2PSA model building in the frame of the 4th generation reactors.

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

- The leading phenomena are those ensuring an adequate core cooling in protected or unprotected situations (i.e. with or without reactor scram). In particular, reactor cooling to maintain fuel cladding integrity is of major importance. If the reactor shutdown and primary cooling systems operate as designed, cladding integrity is guaranteed by design. However if active shutdown and primary cooling systems fail, SFR should be capable of inherent reactor power shutdown (thanks to the combination of reactivity feedbacks) and of natural circulation decay heat removal (NCDHR). Much attention has been paid in past R&D (thanks to EBR and FFTF reactor experiments) to develop models that accurately predict the transition to natural convection cooling, the temperatures that are encountered for the fuel and cladding during that transition and the reactivity changes that result. The confidence level in modelling could be considered good.
- Investigations of transient-overpower events has also been an important area of investigation. For the less severe overpower transients, significant data from transient tests in EBR-II could provide significant confidence in the ability to model fuel performance and the consequences of failure (e.g. such tests

included transients on fuel with breached cladding to determine the potential for fuel loss to the coolant). Reactivity effects of mechanical changes in core structure, sodium density effects and changes in fuel structure have also been extensively studied starting with the investigation of fuel pin bowing in EBR.

- The severe overpower and under-cooling transients received a great deal of attention in the 70s and 80s, especially given the potential for leading to HCDA (e.g. through the results of Transient Reactor Test facility (TREAT) and other similar facilities). For such transients, analysis uncertainties are elevated because of the complexity of the events, the rapidity with which numerous different phenomena occur, and the difficulties of performing and instrument experiments. Analyses of phenomena evolving during a HCDA are also complex, particularly due to the potential corium criticality (linked to the geometry).
- Another difficulty in modelling severe accident progression is the lack of hazards studies, e.g. earthquakes impact. There is also a lack in experience feedback for developing reliability or performance data with a good confidence level, particularly for level 2 PSA. Globally, in fast breeders, it is easier to improve prevention than mitigation.

For transition phase: A large document has been issued in the mid-90s as a common work between searchers involved in the SFR transition phase studies in Europe. A part of the document consists in a broad survey of the state of the knowledge and evaluation of future needs. References of the document are given below [SFR-2-7-1_1]. A short summary of the main shortcomings identified at the time is provided hereafter, in order to give a clearer view on what has yet to be understood through experiments, modelling and implementation in codes than to provide a final state-of-the-art on the question:

- The neutronic impact of blanket-material melt-in or drop-in into the pool region.
- The characteristics of transient pool behaviour (dispersion and compaction processes, transient changes of flow regimes, collapse of vapour bubbles under pressure, ingress of cold liquids or structures).
- The heat transfer characteristics of pool are a function of pool composition.
- The thermal attack and erosion of axial blanket structures from the pool-side.
- The propagation of molten pools, in function of power, into radial blanket structures, radial steel structures and control rods.
- The two phase material relocation in bundles and blockage formation.
- The intra-subassembly blockage behaviour under nuclear power burst conditions (blockage melting and blockage movement under pressure).
- The fuel relocation through the hexcan gap system.
- The break-up of pin stubs under nuclear power burst conditions.
- The blanket fuel/blockage behaviour under mild transient conditions (the problem of blanket pellet drop-in).
- Re-entry of sodium at various thermal conditions into a pool area.
- Steel/sodium interactions.

- Fuel sodium interactions at elevated sodium temperature.

For source term assessment: A PIRT is currently taking place in the US and should bring an up-to-date overview of the present state of knowledge.

GFR: As for SFR, one should first distinguish mechanisms for accidental insertion of reactivity from those leading to reactivity effects due to the loss of fuel integrity. As explained before, insertion of reactivity could occur owing to Control Rods Assemblies (CRA) withdrawal, large water ingress or core radial compaction. The loss of fuel geometry is mostly related to the decrease of decay heat removal performance, then leading to core materials melting and slumping in the fissile region. As a result, several phenomena should be accounted for in calculational tools for consequence assessment. Compared to LMFBRs or LWRs, a substantial lack of feedback and experimental validation is associated with dedicated tools for core damage progression assessment. On the other hand, the nature of the coolant (i.e. without phase change when core temperature increase) is an advantage for core damage depiction and assessment compared to liquid coolant.

LFR: Main limitations are related to the analysis tools (i.e. integral codes), since the analyst should be aware of the phenomena addressed in the code and their modelling approach and limitations. The user should have a sound knowledge of the strengths and weaknesses of the code, which should not be used out of the range of situations and conditions for which it has been designed. It should be noted that any limitations in the Level 1 PSA will be carried forward into the Level 2 PSA. This will need to be taken into account in the intended uses and applications of the L2PSA. If the Level 2 PSA has been based on a L1PSA that has a lower scope than this, these limitations need to be taken into account in the application of the L2PSA. Main limitations related to PSA concern the uncertainties (i.e. parameter, modelling accuracy and completeness) and the time treatment which considers the chronology of events instead of actual timing: this implies the consideration for dynamic event trees. It's worth noticing that the overall uncertainty relaying to the L2PSA consequence assessment consists of two distinguished contributors:

- The uncertainties related to the PSA model.
- The uncertainties related to the code related to correlations and data used to model the phenomena. This class of uncertainties may have different origins ranging from the approximation of the models characterizing any physical phenomena, to the approximation of the numerical solutions, to the lack of precision of the values adopted for boundary and initial conditions, and to the parameters that are the input to the phenomenological models, in addition to the analyzer effect for the numerical simulation of the plant.

VHTR: A full-scope identification and ranking of the phenomena involved in the VHTR concepts conducted by an expert panel has been conducted recently in the US. Results have been published and are the object of two NUREG volumes available to everybody interested on the NRC website. It's rather a challenge to resume such a work in just some lines. For what is related to TRISO particles behaviour, there remain quite a lot of uncertainties and it would be difficult to summarize the needs in just some lines. For the other subjects, the authors resumed in a conclusive table the phenomena for which knowledge is still low either from the phenomenological point of view or from the modelling point of view. Below is a reproduction of this evaluation.

Issue	Rationale
Gas composition	Oxygen potential and chemical activity are central issues for chemical reaction modelling and FP specification. Volatility of FPs can depend on chemical form and oxidizing conditions will cause matrix and graphite damage leading to the released of contained FPs. Retention of FPs on metal surfaces can depend on surface oxidation state. This is scenario dependent. Can influence the IC due to gas impurities. Most needed for the air ingress accident.
FP plate-out and dust distribution under normal operation	The plate-out and dust distribution form the IC for an accident. Theory and models lack specifics; must be coupled with flow and mechanical models as high deposition areas subjected to large changes in flow, temperature and mechanical shock/vibration are candidates for re-entrainment during an accident
Matrix permeability tortuosity	Needed for first particle transport modelling and functions as retention barrier for less volatile FPs both in fuel form and as dust). Some form of fairly comprehensive model over the conditions of interest is needed. Note that this affects FP dust (pebble bed) modelling as well
FP transport through matrix	Once through the particle, the matrix is the first barrier. It also collects FPs as dust. Effective release rate coefficient (empirical constant) as an alternative to first principles may be more tractable. Matrix retention can be important for the less volatile FPs. Dust in the PBMR may be largely composed of matrix, so this issue will affect dust FP modelling as well.
FP transport through fuel block	Graphite can offer substantial attenuation to the transport of FPs and retention the less volatile ones. Effective release rate coefficient (empirical constant) as alternative to first principles (IC and TRANS) may be more tractable but could be highly dependent on the type of graphite. Important for prismatic core because of the series path. Absorption in graphite blocks is desirable so that in dust because of less mobility.
Air attack on graphite	Graphite erosion/oxidation can release the contained FPs and change the chemical form of the FPs as well as weaken the core and damage the fuel. Issues such as Fe/Cs catalysis can change pore structure leading to greater FP release. Some historical data is available but the very small acceptable release fraction may require more detail. Major need for severe accidents.

Steam attack on graphite	If credible source of water are present, contained FPs can be released with problems similar to above. This is design-dependent and much less a problem for a helium-only system. Major need for severe accidents.
FP specification in carbonatious material	The chemical form of the FPs in graphite and matrix material affects transport and retention under both IC and accidents. Uncertain and/or incomplete information in this area. The higher temperatures in the VHTR may influence this. This is a major need as chemical forms strongly influence transport.
FP specification during mass transfer	Chemical change can alter FP volatility. There is some historical data but specific data may be needed for the VHTR. There appears to be good information for metals and oxides. Uncertain for carbides and carbonyls. Need to determine the importance of the issue.
(De)Absorption on dust	Dust provides copious surface area for FP absorption and the high mobility of dust allows the transport of FPs throughout the reactor system. Can be a mechanism that works in parallel with FP volatility for the distribution of FPs. Limited experience. Lack specific details. Some data from AVR.
Ag-110m generation transport	Both Ag and Cs can drive a significant dose on power conversion and heat exchanger equipment. Limited data. Unknown transport mechanism. May alloy with metal components and make decontamination difficult. Possible large impact on maintenance shielding.
Aerosol growth	Lopw aerosol concentration and dry environment can result in the growth of particles with high shape factors and unusual size distribution. Regime has not been studied previously and results need to be determined to assess impact. Vented confinement makes even modest aerosol concentrations important.
FP diffusivity, sorbitivity in non-graphite surfaces	These factors determine FP location during normal operation and act as traps during transient conditions. Can impact O&M as well as accident doses. Past work has examined some metals, but little information may be available for the materials and temperatures of interest. Could be sensitive to the surface oxidation state. Major need for modelling the reactor circuit.
Aerosol/dust bounce, breakup during deposition	Aerosol behaviour can modify deposition profile and the suspended aerosol distribution theory, data, and models lacking. Because of the small acceptable releases due to the vented confinement option, aerosols and dusts take on an exaggerated transport importance. Mechanical issues such as vibration and mechanical shocks need to be taken into consideration as well.
Re-suspension	Since the actual FP content of the gas is expected to be low, the FPs can be released from the surfaces of components becomes important. Past analysis

	has often focussed on flow-induced lift-off of oxide layers and dusts but mechanical shock and vibration induced lift-off can be major drivers as well. Lack of data and models for anticipated conditions, especially mechanically induced ones.
Combustion of dust in confinement	Source of heat and distribution of FPs within confinement iif condition allow the dust oxidation. Results may depend on composition (graphite or matrix) and the amount of air in the confinement.
NGNP unique leakage path beyond confinement	

Table 11: Identified phenomena with low knowledge

Moreover, some areas where it's thought more knowledge is needed are connected with the design (for instance evaluation of the core by-pass flow). The phenomena knowledge has usually been ranked as Medium on a 3-level scale and only three times scored at Low level for the following phenomena :

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

References of chapter 2-7-1

[VHTR-2-7-1_1] NUREG-6944 – Next generation nuclear plant phenomena identification and ranking tables (PIRTs) consisting in a PIRT about main physical phenomena involved in safety issues with VHTR reactors. Volume 2 is specifically devoted to "accident and thermal fluids analysis" whereas volume 3 deals with "Fission-product transport and dose".

[VHTR-2-7-1_2] NUREG-6844 – TRISO-coated particle fuel phenomenon identification and ranking tables for fission product transport due to manufacturing, operations and accidents.

[SFR-2-7-1_1] Report of the AGT4/SG8 task force on transition phase and recriticality. Compiled by W. Maschek & al. June 1991.

[SFR-2-7-1_2] The result of a wall failure in-pile experiment under the EAGLE project. Konishi & alii. Nuclear engineering and design 237, 2007.

3.2 EXISTING AND AVAILABLE TOOLS

As the Generation IV concepts will be improved, one should remain that the calculation tools should be appreciated with regards to their scope and to their validation extend through specific experiments. This aimed at furnishing the limitations of the codes with regards to their applicability for L2PSA consequence assessment (i.e. potential large CET, discrepancy of scenarios and phenomena, assessment of the containment response in addition to the calculation of FPs release in the environment).

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

The advantage of mechanistic analyses is that they predict the detailed sequence of events, and therefore can identify key scenarios and phenomena. The integrated tools are mostly useful in terms of the time frame evaluation and the calculation of the FPs release amount. Modules included in the integrated code system may rely on simplified analytical models developed to provide results with low CPU cost. These modules are confronted to analytical tests or mechanical codes results for validation and could handle uncertainties (defined thanks to validation experiments or to benchmarks with others codes).

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

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

All these above-mentioned elements could be described for the successive phases:

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

SFR: The codes below have been used in the past but in some cases the source codes are no longer available. However, written documentation is available.

Core damage progression:

The SAS-4A code has been developed by the Argonne National Laboratory (USA). It aims at modelling the initiation phase of an accident occurring on a fast breeder reactor (it can be operated for sure for sodium cooled reactors but probably also for other liquid metal cooled breeders) until loss of subassembly hexcan integrity. Developments are still going on and it's widely used in several institutes all over the world. Two other codes previously developed for similar purposes: FRAX (United Kingdom Atomic Energy Authority) and PHYSURA (CEA) are no longer maintained.

SIMMER is a code, originally developed at Los Alamos and since beginning of the 90's by JAEA (Japan), to model the transition phase of an accident occurring on a SFR. At the moment, two versions of the code are available: SIMMER-III which is a 2D-version and SIMMER-IV for 3D-modelling. It combines models for thermal-hydraulics in a sodium environment, structural degradation and neutronics. A two-volume assessment document is available. The code is still widely used in the SFR community and has been extended to other similar reactors as the lead-bismuth fast reactors.

Computations with SIMMER from initiating event are possible for specific scenarios (like the total instantaneous blockage of a fuel assembly, TIB) but lack of some models for fuel-pin behaviour enhances that it's generally not correct to work this way for situations leading to generalized core overheating combined with the scram failure (ULOF, ULOHS). The proper computation technique is to run a SAS-4A computation to provide initial data for a SIMMER computation. The SAME interface is available to convert SAS-4A results into SIMMER data.

Post-accident heat removal phase (PAHRP):

Two codes are under development at JAEA. SIMMER-LT (where LT stands for Long Term) is intended to analyse the re-criticality phenomena after material relocation and for a span of around 1000 s. MUTRAN is an implicit code devoted to the PAHRP analysis for several hours. At the moment, the codes are intended for internal use only but short information is regularly given to the SIMMER users [SFR-2-7-2_4]. LIDEB is a CEA code for modelling the debris bed behaviour.

Sodium fire:

A major drawback of sodium is its strong affinity with oxygen and the fire risk. Two kinds of fires are usually considered: pool fires and spray fires. PULSAR is a 2D-code enabling to model spray fires. Developed by the CEA/IPSN, the last qualified version has been issued in 1999. FEUMIX is a CEA/IPSN zone code developed until 1997. Both of those codes are still available and just begin to be used again in some institutes. The SOFIREII code and its derivative, the SFIRE1C code are presently used in India which is building a SFR small reactor. Some references may be found about the PYROS code but no information about this tool has been retrieved.

Source term assessment - Aerosol behaviour:

The CONTAIN code was developed at the Sandia National Laboratory up to the mid 90's. CONTAIN is an integrated analysis tool used for estimating the physical, chemical and radiological conditions inside a containment building following the release of radioactive material from the primary system in a severe reactor accident. It can also predict the source term to the environment (a user manual for the MAEROS module dealing with aerosol is available online). The CONTAIN code described above is able to predict source terms. Tools devoted to some specific problems were developed for SPX safety studies: FUITE for studying the leaks through the upper seal, IRIS for determination of what is brought to the reactor sky and MIRRA for modelling the argon system. HAARM-3, developed for the NRC and still in-use, predicts the behaviour of sodium, fuel and fission products aerosols inside the containment and the subsequent transport into the atmosphere.

Other codes are :

- AEROSOLS/B1: a CEA code. Last version seems to be /B1 but formerly versions /A1 and /A2 have also been used,
- AEROSIM (UKAEA),
- PARDISEKO-IIIb (FZK, now KIT),

NATURE is an AREVA tool to evaluate the consequences of radiological releases on the environment.

Sodium/Concrete interaction:

The SORBET code developed by the CEA/IPSN treats the transport of sodium and water within the concrete including the ablation rate. Last development took place at the beginning of the 90's. The NABE code has been developed at CEA/IPSN at the beginning of the 80's to model the effects of a sodium pool fire aggravated through a sodium-concrete interaction. Documentation is still available but the code is no longer operated.

The Sandia institute in the USA had also at least one tool, the SCAM code for similar purposes. Here too, it seems the tool has not been operated for quite a long time.

Failure modes of the containment:

Transient loads resulting from a CDA on structures have been studied in the past through a lot of computational tools (a quite long list can be found in reference [SFR-2-7-2_1]) intended for fast dynamics. Now, work is resumed using existing codes such EUROPLEXUS, not specially dedicated to nuclear environment.

GFR: To date, it is intended to make use of exiting codes (with modifications to account for the gaseous coolant, i.e. without phase change but for which radiation processes could be of major concern) for the consequence assessment following the core damage onset. At this stage, the concomitant uses of integral and mechanistic codes is foreseen (like ASTEC as integral tool for core degradation for slow protected transients (possibly including air ingress and nitrogen ingress) into which the physico-chemistry is important to assess and SIMMER for mechanistic evaluation of core damage progression for unprotected scenarios governed by neutronics). However, at pre-conceptual design phase of the GFR, the use of simplified tools allowing for uncertainties propagation all along the scenario is envisioned to clearly and fully describe the accident and also to define the R&D effort that is required for a better understanding of phenomena. The ideal solution identified would be to couple ASTEC with a neutronic code like ERANOS in order to take into account in core degradation simulation, radiative exchanges, the coupling between thermalhydraulics and physico-chemistry and the coupling between materials relocation and neutronics. This kind of tool requires heavy developments not foreseen up to now considering the status of the GFR at CEA.

These simplified tools should be validated by cross-comparisons with mechanistic codes like SAS4A, SIMMER and CONTAIN.

LFR: The status is the following:

- Integral codes as MELCOR and ASTEC are suitable for severe accident and level 2 PSA analysis. MELCOR has been extensively validated against experimental data and is adopted by a world-wide group of users in regulatory, research and utility organisations. ASTEC is a reference code for several European research organisations. It is modularly constructed and validated against many experiments;
- Mechanistic SIMMER code is adopted to evaluate the interaction between lead and water.

VHTR: For commodity of reading, codes are classified according to the key-phenomenon dealt with.

Fuel particle behaviour codes: Reference below gives a brief survey of the numerical tools (past and present) and their main capabilities for modelling fuel particle behaviour. No less than 15 codes are listed but some of them seem quite old. The article alludes to a comparison between 12 of those codes going on in the IAEA.

The existing fuel performance models can be categorized in models that use closed-form analytical solutions for the stress-strain displacement relationships and in those that use numerical approaches, such as finite element analysis or methods (FEA or FEM). The first category is the dominant approach and is based on the analytical solutions. It allows a fast calculation of the stress solutions as a function of fast neutron fluence. However, multi-dimensional effects can be very difficult to implement in these 1D models. On the other hand, the models that use pre-computed Finite Element calculations allow the study of many phenomena, such as asphericity, faceted particles, amoeba effect, layer debonding, localized cracks in the PyC layers. An additional 1D model is then used to statistically take into account these multi-dimensional effects. However, this kind of analysis requires much more computational resources in case that the Finite Elements cases need to be re-computed.

Some of the codes of the first category (and above which some papers have been issued recently) are:

- The PANAMA code developed in the Jülich research center (FZJ) only accounts for the SiC layer. The stress in the SiC layer is calculated with the thin shell model and is purely elastic (pressure vessel model).
- The model developed by Sawa at Japan Atomic Energy Research Institute (JAERI);
- The Russian GOLT code, developed at the Bochvar All-Russia Research Institute of Standardization in Machine Engineering;
- The TIMCOAT code developed at the Massachusetts Institute of Technology in the U.S.A. provides a particle/element model for pebble bed or prismatic block geometries;
- The STRESS3-STAPLE codes developed at BNFL in the U.K.: STRESS3 is the code that performs the stress analysis in the coating layers, and STAPLE is used to apply statistical variations of some properties in STRESS3;

- The PASTA code developed jointly at Delft University of Technology and Idaho National Laboratory: Also based on Miller's solution of stresses in a multi-layer particle, the PASTA code also takes into account the irradiation effects of the graphite matrix in which the TRISO particles are embedded;
- The CRYSTAL code developed at Delft University of Technology and Nuclear Research and consultancy Group in the Netherlands: similar to the PASTA code, the CRYSTAL code also takes into account the thermal expansion of the coatings, variations in particle geometry and material properties, and possible effect of PyC cracking on the TRISO particle behaviour;
- The COPA code developed at the Korea Atomic Energy Research Institute (KAERI).

In the second category, PARFUME and ATLAS are the most known codes:

- The PARFUME code developed at Idaho National Laboratory in the U.S.A.: PARFUME incorporates multi-dimensional effects into a 1D model by using fully 3D ABAQUS calculations to aid the 1D materials model. A suite of ABAQUS calculations are run to account for 3D effects such as shrinkage cracks in the IPyC or particle asphericity; these effects are fed back into the overall fuel performance predictions by using correlations of how they impact 1D-symmetric particle calculations;
- The ATLAS HTR code developed in France at CEA in partnership with AREVA is similar to the modelling approach in PARFUME.

Reactivity insertion code: The TINTE code developed in Jülich research center is available for pebble bed reactors. We have no information about its applicability to prismatic elements reactor nor about a specific code for those reactors.

Core thermal behaviour codes: Modelling of the reactor thermal behaviour may be accomplished with many codes. The problem rather lies in the proper way to operate those codes. The core material being very complex, some homogenization techniques are required to model its thermal behaviour. On one side, those techniques should take into account the uncertainties on the fuel particle distribution and on the other hand, they should also manage with the highly complex graphite behaviour (operating both contraction and dilatation depending on temperature and irradiation level) so that gaps may form and close depending on the core situation and history.

FPs behaviour in the primary system: The following system codes have been used for (V)-HTR.

- MELCOR is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in light water reactor nuclear power plants. The code is capable of modelling fission product transport. A model of sorption of fission product vapours on various surfaces.
- SPECTRA (Sophisticated Plant Evaluation Code for Thermal-hydraulic Response Assessment) is a fully integrated system code designed for thermal-hydraulic analyses of nuclear or conventional power plants. The following sorption models are included in the SPECTRA code:
    - Sorption Model 1 (SPECTRA model). A simpler model, similar to the one adopted in the MELCOR code 0, developed by Sandia.

- Sorption Model 2 (PATRAS/SPATRA model). A more detailed model adopted for the codes PATRAS, SPATRA 0, 0, developed at Jülich.

FPs behaviour inside the containment: The following system codes are applicable and have been used MELCOR, SPECTRA, ASTEC.

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4 SCREENING OF THE COMPLIANCE WITH L2PSA GUIDELINES OF LWRS

Regarding the design features and the main phenomena involved in the selected Generation IV representative concepts, the above-mentioned elements should be considered as sufficient for a first examination of compliance with methodology guidance for a L2PSA building, as regards to the design phase of these reactors.

4.1 COMPLIANCE WITH PWR PHENOMENA AND SYSTEMS FOR L2PSA BUILDING

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

In-vessel core degradation:

- Core degradation: during the heat-up phase and chemical interactions amongst core materials (eutectics potentially lowering the melting temperature, exothermal chemical interactions like oxidation, nitriding…) and relocation processes (potential blockage formation, core collapse);
- Induced-RCS rupture including induced-SGTR;
- Hydrogen production;
- Restoration of core-cooling;
- Vessel cooling from outside;
- Consequences of in-vessel water (or coolant) injection: Assessment of core coolability and additional phenomena to be accounted for following fluid injection;
- Containment atmosphere composition and pressurization (including recombiners/igniters effects): distribution/combustion of flammable products in the containment according to compartmentalization;
- Containment venting;
- Corium criticality (in the core region, in the reactor vessel);
- In-vessel energetic phenomena (leak in the RCS, vessel rupture, containment rupture);
- Reactor Pressure Vessel failure modes (creep rupture, jet impingement, plugging and failure of lower head penetrations…) and consequences (delay, break size…);

Vessel rupture phase:

- Direct Containment Heating, including H2 combustion and vessel uplift;
- Ex-vessel steam explosion;
- Corium criticality in the lower plenum of the reactor vessel;

Ex-vessel phase (MCCI):

- Corium coolability and potential re-criticality;
- Base mat lateral and axial erosion (impact of core catcher for Generation IV concepts);

- Impact of fluid injection (e.g. water for PWR);
- Production of fluid vapours (e.g. steam for PWR) and non-condensable gases;
- H_2/CO combustion;
- Evolution of containment atmosphere composition and long term pressurization;
- Containment venting;
- Drywell erosion;
- Pool scrubbing;
- Melt propagation into ducts and channels;

Containment performance (tightness):

- Initial containment performance (pre-existing leakage);
- Failure of the isolation system;
- Evaluation of containment performance in severe accident conditions;
- Quasi-static loading / dynamic loading – Structural response, structural analyses, fragility curve (leak or break);
- Specific issues: example the impact of a steam explosion in the vessel pit on the overall structure behaviour);
- Drywell/suppression pool performance;
- Containment penetrations performance (tightness) in severe accident conditions;
- Identification of specific containment bypass ways (existing pipes in the plant foundations, cavity door);
- Functions outside the primary containment;

Systems behaviour in severe accident conditions:

- Sump recirculation and spray system;
- Containment Heat Removal System;
- RCS safety valves;
- Steam Generator or Intermediate HX;
- Instrumentation;
- Pedestal cavity flooding systems ;
- Hydrogen recombiners and/or igniters;
- Core catcher;
- Reliability of passive systems;

In order to perform this task, it is proposed hereafter to built a compliance table with 5 values of “compliance levels” ranging from “1” which means that recommendation are highly compliant with the phenomena involved for Generation IV concepts, to “5” that is signifying that no compliance could be exhibited (see Table 12). To sum up, the phenomena expected in the primary circuit can be completely different from LWR ones, because of the presence of coolants (liquid or gaseous, i.e. with or without potential phase change), systems and phenomena of different nature.

Items that could be considered compliant are those sharing the same issues like for instance the structure of the event trees, the L1 to L2PSA interface or the release category definition. On the other hand, HRA can be a little less compliant because of the exclusion of emergency plan for Gen IV reactors that can involve the intervention of the human action to a certain extent. However, at this stage, it is worth noticing that the primary coolant pressure could be a discriminated parameter for gas cooled reactors at core damage onset (i.e. it will be an interface parameter of primary importance to assess the potential High Pressure Melt Ejection, as for LWRs), as it could be considered as parameter of secondary importance for LMFBRs (i.e. the pressure will only have an impact on coolant properties, and then on the scenario timing).

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

- **for SFR:** According to the present knowledge, it is feared that compliance with LWRs phenomena or events in L2PSAa will be very different for a new generation SFR. As examples, L2PSA for a SFR could take into account the consequences of chemical release (sodium oxides by-product: sodium hydroxide, sodium carbonate...) and mission time and safety state definition will be different of LWR L1 and L2 PSAs;
- **for GFR:** The phenomena and the scenario timing expected to occur will be completely different from LWR ones, because of the gaseous coolant (depressurization kinetic, without phase change), the core materials (carbides in an inert gaseous atmosphere) and neutron spectrum (with compaction risk), the thermodynamic cycle leading to the presence of turbo machineries linked to the gaseous secondary circuit (and therefore located in the containment building);
- **for LFR:** The phenomena expected in the primary circuit can be completely different from LWR ones, because of the presence of liquid metal instead of water/steam. For example the hydrogen production could occur in the MCCI only because of the presence of water in the concrete. Some items are considered not compliant (rank 5), because the systems and provisions they relate are not foreseen in the design of ELSY. This applies, for instance, to the suppression pool, the core catcher, the pedestal cavity cooling system, the sump recirculation;
- **for VHTR:** First, the question of making a distinction between L1PSA and L2PSA studies for VHTR may be of not much interest for VHTR so that the question of L1-L2PSA interface is not relevant. In addition, the assessment in the compliance table is based on the following arguments:
    - Human Factors: in this concept, passivity plays an important role diminishing the human factor weight;
    - Core degradation: VHTR core degradation is not equivalent to core melting;

- Induced-RCS rupture including Induced-SGTR: The steam system is inside the containment so that there is no risk of a containment by-pass. Therefore, the problematics will be completely different;
- Hydrogen production: some hydrogen may be produced in case of steam!water entering the primary system;
- Restoration of core-cooling: core cooling may be restored during the accident but the specifics of a LWR core cooling restoration (steam explosion, hydrogen production etc.) are not relevant;
- Vessel cooling from outside: residual power will be evacuated towards the reactor cavity but it is completely different from the LWR concepts;
- Consequences of in-vessel water injection (coolability, hydrogen production, RCS pressurization): water entrance in the primary circuit is an accidental initiator by itself and not a means to mitigate an accident. Phenomenology is completely different;
- Containment venting: Containment venting is a part of the accident management strategy so that problematic are common;
- Corium criticality: There may be some reactivity problems connected with water ingress. The corium concept is not relevant for VHTR;
- In-vessel steam explosion and consequences (leak in the RCS, vessel rupture, containment rupture): Dust explosion may be a possibility in case of air ingress but phenomenology would be completely different;
- Vessel rupture (delay, break size …): Vessel rupture is normally excluded from the concept;
- Corium coolability: Vessel rupture is excluded then some of the problematic still make sense;
- Systems behaviour in severe accident conditions (Sump recirculation, CHRS, Spray system): such systems do not exist on VHTR concepts;
- Pedestal cavity flooding systems, hydrogen recombiners/igniters, sore catcher: Such systems do not exist on VHTR concepts.

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
	Relaibility 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

Table 12: Compliance table - phenomenology

<u>**Specific issues for Gen IV concepts**</u>

As obvious from chapter 2, quite are large quantity of phenomena are not handled by the LWRs guidelines as for instance:

- **for SFR:**
  - Risks associated with secondary circuit's sodium fire in the secondary containment;
  - hydrogen explosion (linked to washing phase) of sodium equipment;
  - sodium voiding;
  - secondary sodium fire in the secondary containment;
  - bypass presence of a secondary circuit inside the containment (for reactors involving an intermediate circuit);
  - Large Sodium fires;
- **for GFR:**
  - presence of the close-containment for GFR;
  - impact by an energetic missile (α-mode) due to turbine locations inside the main containment building (accounting in addition for the presence of a close containment in GFR);
- **for LFR:**
  - molten Core Concrete Interaction;
  - chemical reactions with lead.
- **for VHTR:**
  - the impact of an energetic missile from the helium gas turbine (the rotor),
  - impact of very hot non condensable gas on SSC.

4.2 L2PSA STRUCTURE

The commonly-adopted approach for a level 2 analysis (performed after L1PSA) is:

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

4.2.1 L1PSA-L2PSA INTERFACE PARAMETERS AND MODELLING STRUCTURE

L1PSA provides the sequences leading to core damages. These sequences can be regrouped in several representative states called plant damage states (PDS) featured by so-called interface parameters that permit to decouple the physical calculations performed in the L1PSA from those performed in order to simulate the L2PSA scenarios. Most of the time, interface parameters are relying on:

- The 2nd barrier integrity (e.g. intact RCS vs. LOCA) which is mainly linked to the primary pressure during core meltdown; this point is a concern for HPME and related systems to avoid vessel rupture concerns (with a potential link with SAMG for manual circuit depressurization);
- 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).

Accident sequences from PSA level 1 are grouped together into PDSs in such a manner that all accidents within a given PDS can be treated in the same way. Each PDS represents a group of level 1 accident sequences that have similar characteristics of importance for the severe accident scenarios, e.g. accident timelines and generation of loads on the containment, thereby resulting in a similar severe event progression and radiological source terms. Attributes of the accident progression that will influence accident chronology, the containment response or the release of radioactive material to the environment should be identified. The attributes of the PDSs provide conditions for the performance of severe accident analysis. In other terms, the PDS provide the connexion between level 1 and level 2 analyses by defining the initial and boundary conditions for the level 2 analysis.

Therefore, interface parameters will mainly rely on physical bifurcations of the scenarios (threshold effects) and time scale of events that would discriminate containment responses during and after the core meltdown (for instance, rapid or slow pressurization of RCS and containment, subcritical situation of critical one that can lead to mechanical energy release, corium coolable or not, etc.).

However for Generation IV reactors the choice between performing a stand alone integrated L1/L2PSA model describing the accidental sequence from the IE to the containment failure vs. 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 these main advantages:

- 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.
- Containment isolation failure and containment bypass could be integrated in L1PSA models (with an extension to confinement status for shutdown & refuelling states) in order to keep a L2PSA structure homogeneous between conditional probabilities (without unit, for L2PSA) and reliability of systems (for L1PSA).

A peculiarity is related to the VHTR concept for which the question of making a distinction between L1PSA and L2PSA studies for VHTR may be of not much interest.

This issue seems important to define the scope and the deepness of the L2PSA model that could be elaborated for innovative reactors. The choice L1 and L2PSA quantification tools and methodology (integrated vs. separate) is of major importance for taking benefit from a L2PSA model building (easiness of results integration, consistency of results gained with PSAs).

4.2.2 APET/CET

Roughly, two main methodologies are employed for the development of the APETs/CETs: the large APET, which contains virtually all top event questions regarding the specifics of severe accident modeling, and the small CET method, which includes top event questions concerning the major severe accident phenomena, which are then supported by fault trees. The information that is available to model and quantify the progression of accidents consists of a variety of research results including numerous calculations with computer programs that model special important aspects of the accident progression, as well as experimental results.

For L2PSA models, the APET are developed in similar steps:

- establishing a set of questions about possible events;
- design of the logic structure that forms the tree;
- decision on events and phenomena to be included;
- selection of quantities influencing branching probabilities;
- analysis of dependencies between questions;
- review of the consistency of paths especially with respect to the physical reality;
- identification of risk-important, but uncertain issues, for expert judgment.

For Generation IV reactors, questions arising could be related to:

- The general structure of APET/CET assuming that time phases (i.e. before reactor vessel failure, at failure, and after) could be consistent or not with LWRs ones. If the vessel rupture “notion” (i.e. bottom head rupture in LWRs) could be 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 (and the reason of choice) for APET/CET building (and corollary, which of them could be neglected…). This point is also related to the choice of integrated vs. successive model building issue (see above);
- Mission time (as regards to coolant's inertia: e.g. Na, Lead…), mission time for containment engineered systems 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…).

APET examples:

- SFR: For a sodium-cooled fast reactor, the Level 2 PSA event tree will 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. The criticality risk of the core of a SFR is completely different than for a PWR.
- LFR: The event tree envisaged for the LFR is the following. The initiating event e.g., 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. The initiating event is the SGTR (Steam Generator Tube Rupture), 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 we can have failure of the containment due to MCCI (ε mode); γ mode failure results as combustion of H_2 and other burnable gases as CO and CO_2 coming from MCCI; finally we can have late containment failure due to over-pressurization.
- VHTR : Figure 18 below provides displays a typical accidental tree for a loss of coolant flow accident occurring on the german HTR-1160 (figure was issued in the frame of some PSA studies on this reactor project). The characteristic times given on this figure are relative to this specific reactor but the tree is general enough to be applied to each reactor for such an initiating event. Obviously, in case some safety system is absent from one concept, some branches should be erased.

In Figure 19 is provided a generic event tree based on containment degradation modes. The next step would be to try to decline this generic ET for each Gen IV reactor concepts according to their respective dreaded phenomena. However, the scenario timing should be defined by appropriate transient assessment involving the major phenomena that were depicted in chapter 2. For the β-mode, an appropriate and sufficient knowledge of the main containment penetrations is requested to perform probabilistic quantifications.

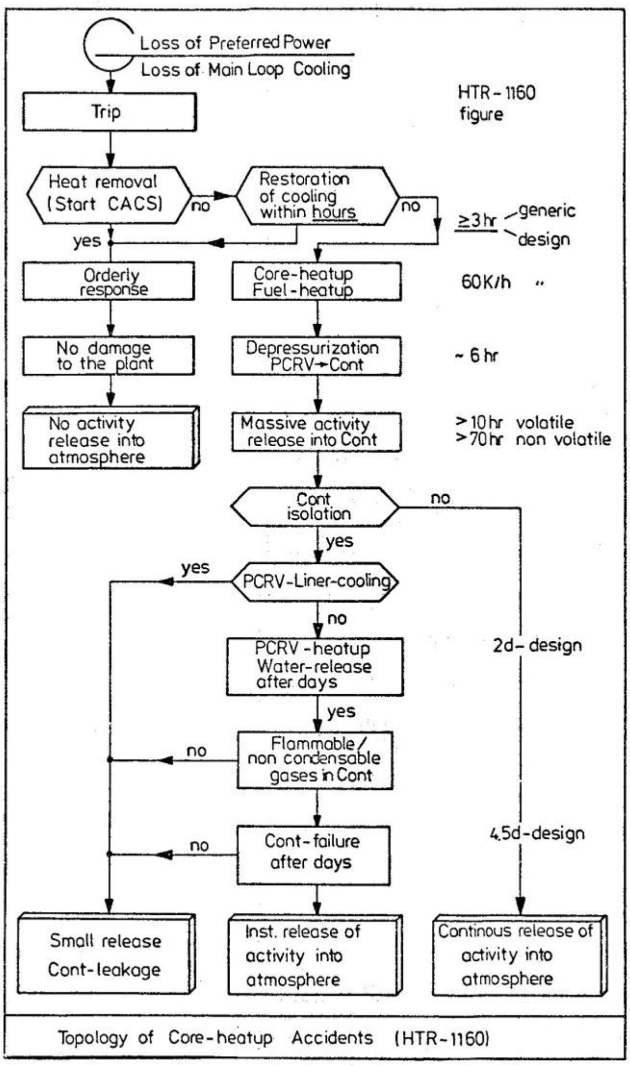


Figure 18: Topology of FP release during core heat-up accidents in VHTRs

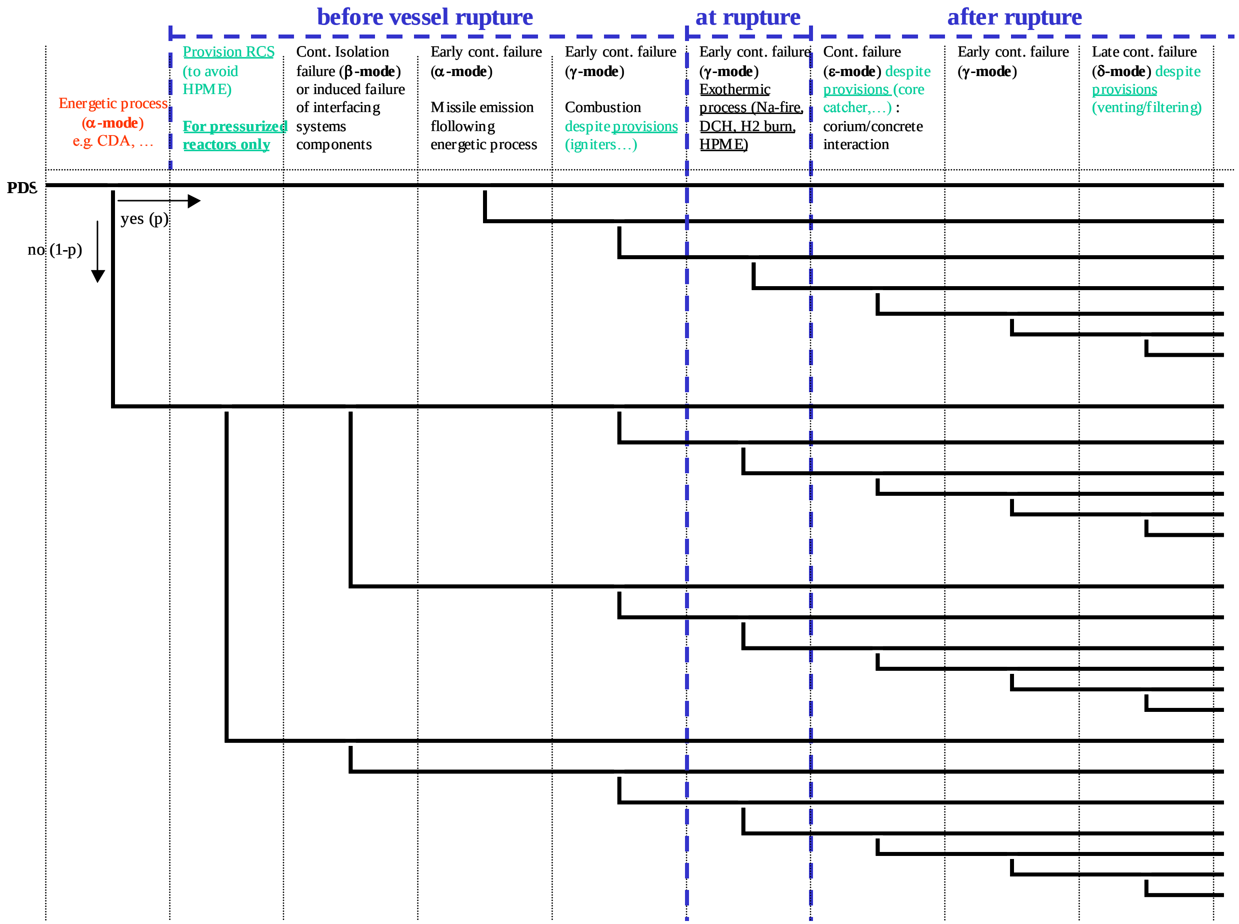


Figure 19: Generic Event Tree related to containment degradation modes

Hereunder are provided some elements regarding specific points that were expressed before:

4.3 HUMAN RELIABILITY ASSESSMENT

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

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

For Generation IV reactors, Emergency Operating Procedures are not defined, nor developed and validated by interviewing and observing control room personnel performance when challenged by events potentially leading to plant damage states.

On the other hand, even if the mitigative strategies can schematically be determined in order to prevent the vessel and the containment failures or to limit the release to the environment can be determined schematically no elements are provided for Generation IV reactors (knowing also that potential actions that can be effective in the mitigation of severe accidents are not always clearly addressed in the Emergency Procedures Guidelines). This constitutes a major difference between L2PSA models that were (or are) built for LWRs compared to those that will be developed for reactors at pre-conceptual design phase (i.e. Generation IV reactors).

However, although mitigation measures and procedures are not defined yet, bounding measures and associated HR data can be established. It is not to be expected that the nature, complexity and timing of mitigating measures will differ to much extend with the present.

As demonstrated by a number of L2PSAs, human actions play a very important role in the safe operation of current Nuclear Power Plants (NPPs). Therefore Human Reliability Analysis (HRA) becomes an extremely important task for the realistic assessment of the plant safety in PSAs (levels 1 and 2).

Also, there is no reason, why the present HRA techniques would (completely) fall short when applied to Generation IV reactors. The specific tasks may change or alter as the sort of work will stay the same.

4.4 QUANTIFICATION OF PHYSICAL PHENOMENA AND UNCERTAINTIES

In principle, a PSA should investigate all possible accident scenarios. A thorough uncertainty analysis can identify areas which need further investigation or special attention (vulnerabilities). Furthermore, if the PSA generates point estimates, an uncertainty analysis may contribute to the credibility of these results.

L2PSA quantification is mainly based on physical calculations of accident progression involving thermal-hydraulics, interaction of physical and chemical processes that can occur in the course of the accident, assessment of barriers integrity. All these elements are transposed as top events in the Accident Progression and Containment Event Trees (respectively APET and CET). Therefore, physical calculations are more a concern than failure of systems. As for LWRs and during the last decades, a large extend of work involving numerous countries was performed to develop of physical models, to imagine experiments for validating these models, to realize transient calculations. This vast experience may be used in Generation IV, but an adaptation of models may be needed, as well as a deep validation. One important issue, maybe the most important, will to manage the lack of data, quantification – codes and validation - with uncertainty ranges for innovative reactors (please also refer to the notes about the uncertainties in 2.7.1).

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

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

The analysis may be first qualitative with the prime objective of identifying and ranking the most important uncertainties. This may include limited sensitivity analyses and can be performed prior to a quantitative uncertainty propagation analysis. Then, a decision has to be made if a formal uncertainty propagation analysis is conducted or just a limited uncertainty analysis i.e. only an identification of major uncertainties through sensitivity studies._The format and the range of the uncertainty parameters of each issue have to be defined e.g. probability distributions or just bounds. After selecting the method of propagating all the important uncertainties through the different steps of the PSA (e.g. Latin Hypercube Sampling), the uncertainties at the different levels have to be combined in any case, to estimate the overall uncertainty of the result. At intermediate levels, appropriate uncertainty measures have to be computed or qualitatively assessed but at last, the uncertainty in the overall results can be only displayed by probability distributions or by mean value (or median) in combination with percentiles.

4.5 PASSIVE SAFETY SYSTEMS

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

The term passive system is generally used for systems that perform a certain safety function using a natural process without the support of other operational systems or human action. In IAEA publication, one defines passive systems as systems that have no need for external input, especially energy, to be able to operate. In this reference passive systems are divided in four categories:

a) physical barriers and static structures (characteristics based on material, condition, design and geometrical placement);
b) movement of fluids/gases (due to phase changes, chemical reactions or neutron flux effects);
c) moving of mechanical parts (for example the opening of a spring loaded check valve as a result of a pressure difference);
d) external signals and potential energy (passive action / active actuation).

Usually passive systems in selected GenIV representative reactors refer to the systems/processes of type b.

Hereunder are recalled the main features of potential Passive Decay Heat Removal Systems:

- Through structures, e.g. decay heat removal through the reactor vessel walls by radiation plus conduction/convection mechanisms with surrounding atmosphere and with specific DHR systems (e.g. RCCS for VHTR, maybe employed for SFR and LFR);
- Moving fluid and engaged forces for Natural Convection Decay Heat Removal (e.g. GFR or SFR), e.g. dedicated DHR loops acting in a natural convection mode;

An important question is how success or failure of passive systems should be incorporated in a L1PSA analysis. In case of active systems there are only two possibilities; the system operates or it fails (binary logic). In passive systems two problems may occur: the passive system does not function at all (the desired operation does not start due to physical processes or conditions) or is being hampered (the function is degraded). In case of degradation the duration of the problem (delayed start) is of importance as well as the magnitude of the degradation. The seriousness of the degradation depends heavily on the operating conditions (temperature, pressure and flow ranges of diverse interacting systems in which the passive system should operate). Therefore there is no simple one to one relation between defined success criteria and the probability of compliance of the passive system.

Within the framework of the 5th FP RMPS project, a methodology has been developed to evaluate the reliability of passive systems characterized by a moving fluid and whose operation is based on physical principles, such as the natural circulation. The reliability evaluation of such systems is based in particular on the results of thermal-hydraulic (T-H) calculations. This methodology can be structured in three parts:

- Identification and Quantification of the sources of uncertainties;
- Reliability evaluations of passive systems with techniques used in Structural Reliability analyses;
- Integration of passive system reliability in the Probabilistic Safety Assessment; The passive safety system reliability can be introduced without any problems in the generally applied static event tree/fault tree technique to model the course of accidents.

To sum up, the reliability of passive systems for decay heat removal relying on physical phenomena as conduction/radiation and natural convection should be clearly addressed, in order to provide valuable reliability figures for the safety studies within a risk informed approach.

4.6 CALCULATION TOOLS AND UNCERTAINTIES

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

4.7 ROLE AND EXTENT OF EXPERT JUDGMENT

Expert judgement plays an important role in assessing the progress and probabilities of events in a L2PSA. This is even more relevant for GEN IV reactors due to the limited experience in comparison with LWRs. To improve the expert judgement process a structured procedure has to be adopted to address the not well known physical phenomena occurring during the accident progression. The issue is therefore not so much the use of expert judgement in itself, but how to distinguish between different experts or to determine who the real expert is. Methods are available and referenced.

Results of those last chapters discussion are summarized in the table below.

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

Table 13: Compliance table - methodology

[4-7_1] A survey of expert opinion and its probabilistic evaluation for specific aspects of the SNR-300 risk study – Hofer et al. – Nuclear Technology, vol. 68, pp 180-225 – February 1985]

[4-7_2] Cooke R., Experts in Uncertainty; Opinion and Subjective Probability in Science, Oxford University Press; New York. Oxford, 321 pages. 1991; ISBN 0-19-506465-8

5 CONCLUSION AND PROSPECTS

As expressed in the Generation 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".

Accordingly to "end-users" requirements regarding the L2PSA guidelines (i.e. tome1), it clearly exhibits that the following issues should be addressed, whatever the reactor concept is:

- the determination of LERF/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).

Therefore, the question arising could be related to the place and extend/use of Level 2 PSA in the frame of Generation IV design improvements and safety demonstration. The main objective assigned to the Work Package 4 (WP4) of the "ASAMPSA2" project (EC 7th FPRD) was first to build the most exhaustive list of mechanisms and provisions involved in the selected Generation IV concepts in order to help verifying the potential compliance of L2PSA guidelines based on LWRs reactors (which are specific tasks of WP2 and WP3) with those of Generation IV representative concepts.

Regarding their "place", L2PSA models and their related containment performance assessments, even performed at an early stage of a reactor design, could furnish valuable insights for:

- 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.

To date and according to the review performed in this document, it seems achievable to perform "simplified" L2PSA for preliminary LERF/LRF assessments. Even if L2PSA models for innovative reactors will not have yet the scope and the deepness of L2PSA models built for operating reactors (i.e. LWRs), it seems that this objective is reachable in order to initiate the unavoidable process (and progress) that is leading to L2PSA. It requires that the main containment features for GenIV reactors could be provided (e.g. main penetrations for β-mode evaluation, systems implemented for FPs retention and associated release strategies, static and dynamic design pressures...).

Then, in a second step, all along the reactor design progress, containment design improvements could be defined owing to insights gained by the use of the L2PSA model results (e.g. additional specific provisions implemented to limit the SA progressions or their related consequences, like core catcher or containment venting/filtering systems). Another step could be related in defining human actions important for safety (i.e. Emergency Operating Procedures) and then Severe Accident Management features, systems and procedures. A support for decision making for design improvements, potentially including cost-benefit considerations, is another step of use of L2PSA results.

Finally, at the licensing phase and in relation with the safety demonstration (i.e. “risk-informed” framework), the L2PSA could help for the practical elimination of sequences or phenomena (e.g. HCDA), that could finally be also based on a precise estimation of the quantities of released radioactive material to the environment (e.g. for LERF/LRF assessment).

Therefore, among all the above-mentioned issues, a hierarchy should be defined for a L2PSA applied to GenIV reactors depending on the reactor design extend and on the expected results. In addition, the scope (e.g. full power only or all reactor states, internal events only vs. treatment of hazards) and the “deepness” (implementation or not of HRA, simplified L2PSA i.e. more like L1+PSA one, integrated L1/L2PSA models, assessment of uncertainties) should be addressed with regards to the high-level objectives and the knowledge extends of the concept when the L2PSA model is constructed.

At this stage, it appear that the major distinction of L2PSA that could be performed for GenIV reactors will start at early stages of design, on the contrary with those performed in LWRs for which probabilistic models were realized while these reactors were still built and operating (i.e. the containment building features were known). As a consequence, in a continuous and progressive way (L2PSA evolving with the design progress, i.e. from pre-conceptual to licensing phases), the successive releases of the L2PSA model should be clearly defined to provide the most useful insights. For instance, the successive modelling could be either attached to the reactor design phases or to the knowledge and the code adequacy progresses.

Compared with LWRs main phenomena, one major feature of fast reactors (i.e. involving a fast neutron spectrum) is related to the phenomenology of the potential core destruction. Indeed, for some scenarios, the core can melt slowly, without mechanical energy release, or it can be destroyed by a rapid nuclear excursion. Then, the core disruptive accidents for Fast Reactors lead to require a refined kinetic assessment for the APET building. Correlatively, mission times (related to scenario kinetic) devoted to systems and “safe” end states are constituting an important issue for new reactor types compared to LWRs one.

Regarding Fission Products, the Plutonium and Minor Actinides inventories in the core (especially for fast reactor cores) would have a large impact on the potential source term outside the confinement. This point seems to be a major concern for FRs, taking also into account the absence of such containment engineered systems like the containment spray system in LWRs (that has a role for FPs deposition in the containment building). Then, it appears that the assessment of the potential source term is an important issue for GenIV reactors.

In addition, risk analyses has to investigate conditional failure probabilities of safety devices in case of accidents as well as the subsequent modes and expected frequencies of release of radioactive material into the environment. Besides the activity inventory inside the core, radioactive material in other locations inside the plant has to be considered, particularly in the spent fuel storage pools. In corollary, for refueling states, specificities of Gen IV reactors should be accounted for in comparison with LWRs. Specific issues should be addressed as, for instance, the fire concern induced by sodium or graphite interaction with air or water (for induced effects on the primary vessel and on the containment (it is worth noticing that it also influences the source term through the chemical form of FP species and the driving force).

Finally, the compliance with L2PSA guidelines derived from LWRs will be effectively addressed only when the basis of probabilistic models for GenIV were exhibited. This could be the further step in continuity of the ASAMPSA2 project to perform a simplified L2PSA model for a Liquid Metal Fast Breeder Reactor (assuming that GFR is not the most promising concept, and knowing the VHTR specificities mentioned in this document).

The main issue for GenIV reactors that are still under design is related to the absence of Emergency Operating Procedures (EOPs) or Severe Accident Management Guidelines (SAMG). However, and even if the preventive strategies related to core damage prevention (i.e. control of reactivity, maintenance of heat removal) can be accomplished with combinations of systems and/or operator interventions that should and will be well defined in EOPs, the mitigative strategies (as to prevent vessel failure, to prevent containment failure or to limit the release to the environment) don’t feature unambiguous, complete and correct directions for implementation. Under these circumstances, it seems precipitate to look at this point for GenIV reactors.

GLOSSARY

AC	Alternating Current
ACS	Above Core Structure
APET	Accident Progression Event Tree
BOC	Begin Of Cycle
BOL	Begin Of Life
BRI	Bulk Rod Insertion
BWR	Boiling Water Reactor
CDA	Core Disruptive Accidents
CET	Containment Event Tree
CREED	Control Rod Enhanced Expansion Device
CSD	Control and ShutDown absorber rods (or drive mechanisms)
DC	Direct Current
DHR	Decay Heat Removal
L2PSA	Level 2 Probabilistic Safety Assessment
DHRTV	Decay Heat Removal Through the Vault
SGOSDHR	Steam Generator Outer Shell Decay Heat Removal
DHX	Direct Heat eXchanger
DRC	Direct Reactor Cooling
DSD	Diverse ShutDown absorber rods (or mechanisms)
EFR	European Fast Reactor
EOC	End Of Cycle
EOL	End Of Life
FCI	Fuel Coolant Interactions
FP	Fission Products
GFR	Gas cooled Fast Reactor
IE	Initiating Event
IHX	Intermediate Heat eXchanger
IS/LOCA	Intermediate size LOCA
L2PSA	Level 2 Probabilistic Safety Assessment
LERF	Large Early Release Frequency
LFR	Lead cooled Fast Reactor
LOCA	LOss of Coolant Accident
MCCI	Molten Core Concrete Interaction
PDS	Plant Damage State
PWR	Pressurized Water Reactor

RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
SADE	DSD rod scram magnet de-energization system
SAM	Severe Accident Management
SAM	Severe Accident Management Guide
SFR	Sodium cooled Fast Reactor
SGB	Steam Generator Building
SGOSDHR	Steam Generator Outer Shell Decay Heat Removal
SGTR	Steam Generator Tube Rupture
SLD	Stroke Limitation Device
SSE	Safe Shutdown EarthquakeL
L2PSA	Level 2 Probabilistic Safety Assessment
VHTR	Very High Temperature Reactor
VS	Vessel System

OTHER REFERENCES

References have been presented at the end of each individual chapter.

VHTR

AIEA means the article is available on the AIEA website.

A general presentation of the VHTR has been the subject of an eurocourse in 2002. The course is available at https://odin.jrc.ec.europa.eu/htr-tn/HTR-Eurocourse-2002.

Articles recently published in the scientific literature and available on the sciendedirect web site (restricted access).

Some other references of interest are listed below :

Experimental and computational study of the pyrocarbon and silicon carbide barriers of HTGR fuel particle. Golubev, Kurbakov, Chernikov. Atomic Energy, vol 105, n°1, 2008.

Statistical approach and benchmarking for modelling of multi-dimensional behaviour in TRISO-coated fuel particles. Miller & alii. Journal of nuclear materials, 317, 2003.

Considerations pertaining to the achievement of high burn-ups in HTR fuel. Martin. Nuclear Engineering and Design 213 (2002)

APPENDIX A: ELEMENTS ON THE PRINCIPLES USED FOR AN EXCLUSION OF SEVERE FUEL CONFINEMENT DAMAGE (CORE MELT) FOR VHTR

To be irrefutable, the justification of "no core melt" should go beyond the approaches required for reactors such as LWR and LMFBR where core melt is considered despite the implementation of a high prevention level.
The justification mainly relies on:

- The development of a coated fuel particle that essentially ensures the confinement function in any situation (i.e., high quality level of design, fabrication and control, adequate qualification programme for irradiation and accident conditions),
- The development of design options to limit the challenges on the fuel particle and aiming at providing slow evolution of accident transients, and then providing a large grace period for implementation of corrective actions for the mitigation of the accident consequences. In particular, the reactor design is optimized to favour mitigation by means of natural behaviour based on the intrinsic and passive characteristics of the plant (e.g., annular core geometry, low core power density, helium as a coolant, coated fuel particles and fuel elements withstanding high temperatures, adequate operational parameters, high negative temperature reactivity feedback, high thermal conduction and inertia of the core graphite, heat transfer by radiation, etc.),
- The high quality level of equipment used for mitigating the consequences of the enveloping situations. Along with the high quality level, the repair capability (requiring grace period and access), redundancy, diversity of systems, and in-service inspection (ISI) might allow to convincingly exclude the equipment from failing completely,
- The prevention of enveloping situations by high performance systems.

The design options adequacy with respect to safety is in particular assessed by means of:

- The establishment of an <u>exhaustive</u> list of events challenging the confinement function,
- The <u>study of enveloping situations</u> mitigating exclusively by inherent behaviour and therefore postulating failure of any active mitigating device (in particular, the unavailability of any power supply is assumed). The enveloping situations are defined with respect to the potential risks and in order to cover a possible lack of exhaustiveness or uncertainties about identified scenarios. Their study aims to prove that the consequences on the fuel particles are limited, and that there is a sufficiently large grace period for implementing corrective actions, such that severe fuel confinement damage can be practically eliminated. The duration of the grace period will be assessed on a case-by-case basis, with consideration of the relevant corrective actions that need to be performed and the phenomena that could occur during the grace period.

Nevertheless, if these objectives are not fully achieved (as is expected for a small number of cases), then the combination of active and passive devices is implemented in such a way that their complete failure can be practically eliminated (i.e., the occurrence frequency of the initiating event combined with the failure of active and passive mitigating systems is very low).

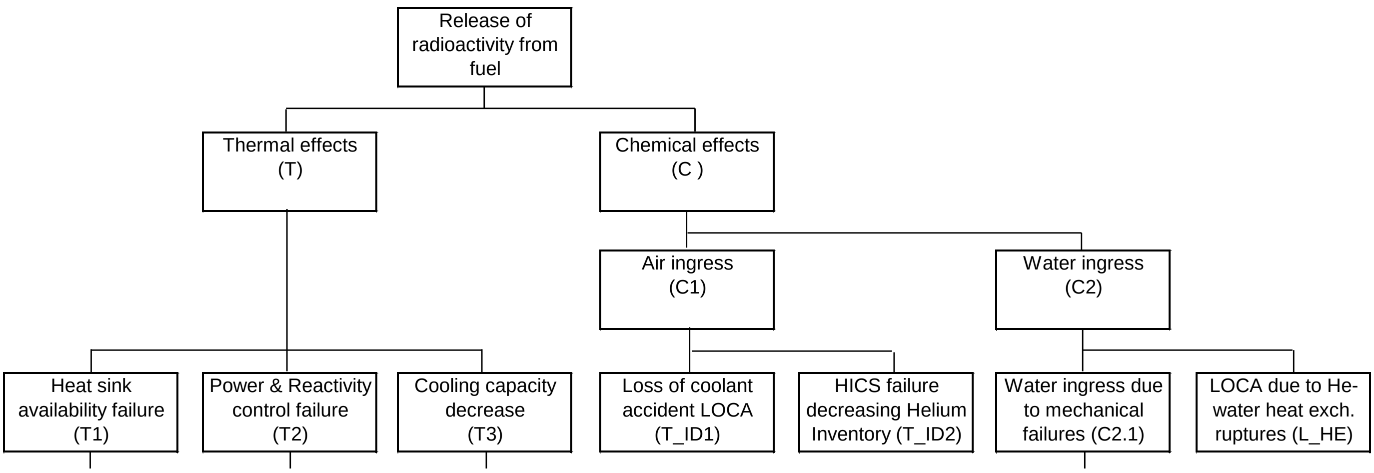
APPENDIX B: TABLE OF INITIATING EVENTS FOR VHTR (PBMR)

SOURCE: Evaluation PSA1 analysis technique for HTR power plants, E. van Wonderen, NRG, 21355/04.63029/C Arnhem, 22 February 2005

A table of initiating events for a PBMR type HTR is presented below. Some elements about the methodology for establishing such a list are given as an introduction. Two classifications of the initiating events are provided.

The presented initiating events list is obtained using the following four methods, which are recommended in general (IAEA. EUR etc) to reach completeness as far as possible:

1) Engineering evaluation. In this technique all possible (partial) failure modes of all systems (operational plant systems, safety systems) and components that can lead directly or indirectly to accident conditions, also in combination with other failures or problems, are assessed. All disturbances that lead to a scram are of course also initiating events.
   Per system, each failure mode that might lead to abnormal or accidental conditions possibly in combination with other disturbances should be identified.
2) Use earlier produced lists. The earlier produced lists may be used as a starting point. The lists should be screened on appropriateness in relation to actual design of the plant or external conditions. In the reports NUREG/CR-2300 and the IAEA safety series a number of generic lists are presented.
3) Master Logic Diagram: A master logic diagram (MLD) is a tool in identifying initiating events and ensures to a high degree the completeness of the analysis. It is a top down approach that starts with the top event "fuel challenge". This event is subsequently subdivided in all possible scenarios that can lead to this event. Successful operation of safety systems and other mitigating action should not be considered. The events at the lowest level are candidates for the list initiating events.



4) Operational experience. In this approach, the operational history of the plant (if any) and of similar plants elsewhere is reviewed for any events, which could be added to the list. This approach is considered only supplemental, as it is not likely that it will reveal low probability events.

Initiating events

Inventory
Loss of Inventory (Large / medium / small LOCA) caused by tube ruptures, RPV failure, PPB failure (ingress of air, water)
Leakage/seal failure/break in HICS/HPS/CCS/FHSS/RPVCS
Leakage from control rods (CRDM seal failure / control rod ejection)
Inadvertent opening of a safety/relief valve (stuck)
He/water heat exchanger leakage/rupture (depressurisation / ingress of water)

HICS Pressure regulation fails: inventory decrease /depressurisation
HICS Pressure regulation fails: inventory increase
HICS Pressure regulation fails closed

Turbine
Electric load rejection (HV breaker opening / generator trip or faults)
Electric load rejection with gas cycle bypass valve failure
Turbine trip (loss of offsite power / cooling water / Resistor Bank)
Loss of turbine (bearing / blades/ shaft / disk etc.)
Turbine trip with gas cycle bypass valve failure

Flow
Control valves malfunction cause increase/decrease in pressure and flow:
Gas cycle Bypass Valve (GBP)
High/low Pressure coolant valve (HCV/LCV)
Recuperator bypass valve (RBP)
Start-up blower system inline valves (SIV)
High/low pressure compressor bypass control valves (HPBC/LPBC)
High/low pressure compressor bypass valves (HPB/LPB)
He-water heat exchanger (Intercooler/Pre-cooler/CCS) He-flow decrease
Flow decrease (blockage) recuperator
Inadvertent operation SBS at power
Trip/loss of High Pressure compressor/turbine
Trip/loss of Low pressure compressor/turbine
Trip of both turbo compressors (HPC and LPC)
Low helium flow during startup or shutdown (SBS failure)
High helium flow during startup or shutdown (SBS failure)

Heat removal
He/water heat exchanger(s) water flow decrease (blockage): He-temp increase
He/water heat exchanger(s) water flow increase: He-temp decrease
He/water heat exchanger(s) water flow temp increase/decrease: He-temp change
Loss of generator cooling (leakage / loss of water flow)
Loss of active cooling system ACS (heat sink)
Loss of Core Conditioning System (CCS) (at shutdown)
Inadvertent operation of CCS at power
Loss of Reactor Cavity Cooling System (RCCS)
Loss of Reactor Pressure Vessel Conditioning System (RPVCS)
HE channel blockage

Reactivity control
Uncontrolled Rod (group) withdrawal/insertion at power
Uncontrolled Rod (group) withdrawal in accident event (re-criticality)
Uncontrolled SAS removal/insertion from reactivity control system
Uncontrolled SAS removal/insertion from cold shutdown system
Inadvertent reactivity control unit actuation
Uncontrolled fuel loading (mix up of spheres, detection errors)
Blockage of fuelling pipe
High flux due to rod/SAS withdrawal at startup
Pressure, temperature, power imbalance--rod/SAS-position error
Reflector geometry modification (top reflector drop)
Scram due to plant occurrences
Spurious trip via instrumentation, RPS /EPS /OCS fault
Detected fault in reactor protection system
Loss of computer control (RPS)
Manual scram--no out-of-tolerance condition

Internal/External events
Loss of offsite power
Loss of auxiliary power (loss of auxiliary transformer)
Loss of DC power bus(es)
Heavy Load Drop
External explosion
Seismic event
Core structural support failure
Aircraft crash
(Turbine) Missiles / Projectiles
Floods

Fire within plant

Alternative arrangement

Large LOCA, non-isolatable

Large Loss of Inventory caused by tube ruptures, RPV failure, PPB failure (ingress of air, water)
Heavy Load Drop
External explosion
Seismic event
Core structural support failure
Aircraft crash
Control rod ejection
Floods

Large LOCA, isolatable

Large Loss of Inventory caused by tube rupture /vessel breach HICS

Intermediate LOCA, non-isolatable
Intermediate Loss of Inventory caused by tube ruptures, RPV failure, PPB failure (ingress of air, water)
Break in CCS/RPVCS
(Turbine) Missiles / Projectiles
He/water heat exchanger tube rupture

Intermediate LOCA, isolatable
Break in HICS/FHSS
Inadvertent opening of a safety/relief valve (stuck)

Small LOCA, non-isolatable
Leakage/seal failure in CCS/RPVCS
Leakage from control rods (CRDM seal failure
He/water heat exchanger leakage

Small LOCA, isolatable

Leakage/seal failure in HICS/FHSS

No heat removal from core (Loss PCU & CCS)
Loss of offsite power
Loss of auxiliary power (loss of auxiliary transformer)
Loss of DC power bus(es)
Fire within plant
Loss of computer control (RPS)
Loss of active cooling system ACS (heat sink)

Decreased heat removal from core / RPV
He/water heat exchanger(s) water flow decrease (blockage): He-temp increase

Electric load rejection (HV breaker opening / generator trip or faults)
Electric load rejection with gas cycle bypass valve failure
Turbine trip (loss of offsite power / cooling water / Resistor Bank)
Loss of turbine (bearing /blades/ shaft /disk etc.)
Turbine trip with gas cycle bypass valve failure
Loss of Reactor Pressure Vessel Conditioning System (RPVCS)
Loss of Reactor Cavity Cooling System (RCCS)
Loss of Core Conditioning System (CCS) (at shutdown)
Loss of generator cooling (leakage / loss of water flow)
He/water heat exchanger(s) water flow temperature increase
He-water heat exchanger (Intercooler/Pre-cooler /CCS) He-flow decrease
Flow decrease (blockage) recuperator
Trip/loss of High Pressure compressor/turbine
Trip/loss of Low pressure compressor/turbine
Trip of both turbo compressors (HPC and LPC)
Low helium flow during shutdown (SBS failure)

Control valves malfunction causing change in pressure and flow:
- Gas cycle Bypass Valve (GBP)
- High/low Pressure coolant valve (HCV/LCV)
- Recuperator bypass valve (RBP)
- Start-up blower system inline valves (SIV)
- High/low pressure compressor bypass control valves (HPBC/LPBC)
- High/low pressure compressor bypass valves (HPB/LPB)

Power/reactivity increase
HICS Pressure regulation fails: inventory increase
Uncontrolled fuel loading (mix up of spheres, detection errors)
Blockage of fuelling pipe
Uncontrolled Rod (group) withdrawal at power
Uncontrolled SAS removal from reactivity control system
Uncontrolled SAS removal from cold shutdown system
High flux due to rod/SAS withdrawal at startup
Pressure, temperature, power imbalance--rod/SAS-position error
Reflector geometry modification (top reflector drop)

Miscellaneous
HICS Pressure regulation fails: inventory decrease /depressurisation
HICS Pressure regulation fails closed (no regulation)
Uncontrolled Rod (group) insertion at power
Uncontrolled Rod (group) withdrawal in accident event (re-criticality)
Manual scram--no out-of-tolerance condition
Scram due to plant occurrences
Detected fault in reactor protection system
Spurious trip via instrumentation, RPS /EPS /OCS fault

He/water heat exchanger(s) water flow increase: He-temp decrease

Uncontrolled SAS insertion from reactivity control system
Uncontrolled SAS insertion from cold shutdown system
Inadvertent reactivity control unit actuation

Inadvertent operation of CCS at power
He/water heat exchanger(s) water flow temperature decrease

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Inadvertent operation SBS at power
High helium flow during startup or shutdown (SBS failure)
Low helium flow during startup (SBS failure)
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APPENDIX C: REVIEW OF FORMER GFR CONCEPTS

- A review of former gas-cooled reactor concepts is first presented.
  - The concept of the ETGBR was studied in the UK during the late 1970s. The reactor design was based on the AGR but embodied a fast neutron core. The ETGBR study was therefore performed when the AGR reactors were well established, operating as the mainstay of the UK nuclear programme and after the European GBR Studies. The point of the study was to establish an economic design that maximized the use of verified technology, avoiding the parameter extrapolation of the GBR design. The ETGBR core design was based on fuel element technology closely related to that developed for sodium cooled fast reactors (i.e. conventional MOX or UOX steel clad pellets), but also took account of the experience gained from AGR and PWR development. For reactivity control, three separate and diverse rod systems for control and shutdown satisfying requirements for independence, diversity and redundancy. A vented containment building is proposed to limit the potential release under severe accidents. A cylindrical building can more readily be designed for higher pressures should this be shown to be necessary by analysis.
  - The European Gas Breeder Reactor Association investigated four design concepts in the late 1960s and early 1970s basing their designs on the then current technology for gas cooled thermal reactors and the LMFR fuel and core. Initially a range of schemes was examined: alternative coolants (He and CO2) and alternative fuel concepts (pins and coated particles). In 1972 the Association decided that the most reliable and attractive prospects for the short and medium term would be the steam-generating system with fuel pins and cooled by helium i.e. GBR4. Their work was then devoted to this concept addressing all feasibility, performance, safety, economic and the R&D questions related to the design. For reactivity control, two independent shutdown systems were employed. There were also good self-shutdown characteristics due to negative expansion coefficients that reinforced the Doppler coefficient. In addition, for an untripped loss of flow (ULOF), there are fuses on the absorber rods that melt and release the rods. A containment building comprising an inner steel liner and outer concrete shell provides a leak tight barrier for activity release and maintenance of a 2.5 bar equilibrium pressure to reduce circulator power for decay heat removal following a depressurisation fault.
  - The GCFR program has been supported in the United States since the 1960’s but more intensively since 1978 in order to develop in parallel alternative breeder technologies to LMFR. The gas cooled fast reactor technology has been considered because of its promise of higher breeding ratio, simplicity in operation compared to sodium cooling and potential lower capital costs taking advantage of the work done on HTR development (Peach Bottom and Fort St. Vrain reactors). The design is LMFR based with Niobium stabilized 316 stainless steel pins and wrappers and pressure equalisation for the pins (i.e. vented pins) allowing a higher clad temperature without

rupture and preventing fission gas release in case of clad rupture but with questionable acceptability today. For reactivity control, two independent and diverse shutdown systems were employed. The containment was made of two barriers (the pre-stressed concrete reactor vessel and the reactor containment building / reactor confinement building), as the fuel clad was questioned as the first barrier because of the pressure equalisation system.