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A comparative study on severe accident phenomena related to melt progression in SFR and PWR

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Abstract

The nuclear safety approach has to cover accident sequences involving core degradation in order to develop reliable mitigation strategies for both existing and future reactors. In particular, the long-term stabilization of the degraded core materials and their coolability has to be ensured after a severe accident. This paper focuses on severe accident phenomena in Pressurized Water Reactors (PWR) compared to those potentially occurring in future GenIV-type Sodium Fast Reactors (SFR). Firstly, the two considered reactor concepts are introduced by focusing on safety aspects. The severe accident scenarios leading to core melting are presented and the initiating events are highlighted. The paper focuses on in-vessel severe accident phenomena, including the chronology of core damage, major changes in the core configuration and molten core progression. Regarding the mitigation means, the in-vessel retention phenomena and the core catcher characteristics are reviewed for these different nuclear generation concepts (II, III and IV). A comparison between the PWR and SFR severe accident evolution is provided as well as the relation between governing physical parameters and the adopted mitigation provisions for each reactor concept. Finally, it is highlighted how the robustness of the safety demonstration is established by means of a combined probabilistic and deterministic approach.

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1. Introduction

General safety objectives are adopted for Generation II, III and Generation IV of nuclear reactors. These objectives corresponding to the radionuclide maximum release target are proven to be reached thanks to the implementation of a safety approach relying on Defence-in-Depth (DiD)[1]. Among the various reactor operating conditions and reactor situations addressed with DiD, this paper points on the accidents with core melting on both, PWR and SFR reactors, as illustrations of Gen-II/III and Gen-IV concepts.

In this paper, the term ‘severe accident’ refers to an event causing significant damage to reactor structures and resulting from more-or-less complete core meltdown. These accidents are on the focus of considerable research work because the release of radioactive products into the environment would have serious consequences. Some accidents can be classified as practically eliminated, and these are not detailed further in this paper. In order to “practically eliminate” a situation, the designer shall first examine the possibility for making it physically impossible. Where physical impossibility cannot be achieved, provisions shall be implemented to justify with a high degree of confidence that the situation is extremely unlikely. For accidents with core melting that cannot be practically eliminated by prevention measures, design provisions have to be taken so that only limited protective measures in area and time are needed for the public and so that sufficient time is available to implement these measures [2].

This paper addresses reactors currently in operation (Gen-II/III) as well as future reactors (Gen-IV). Some comments will be addressed to advanced PWR concepts, referred as Gen III+ that have superior characteristics to the older designs with respect to gaining grace time and reliability for severe accident management.

The previous work of different authors concerned the comparison of PWR and SFR in view of core design and characteristics [3], or the comparison of different advanced reactor technologies [4]. The objective of this paper is to focus on comparison of PWR and SFR in terms of safety, providing comparison of severe accident phenomena and mitigation strategy.

The concept of a severe accident mitigation within the operating PWR and the nuclear power plants under construction is being studied in the frame of different European programs [5][6]. The concept of a severe accident, which is associated with core melt for second and third generation reactors, has yet to be defined for some fourth generation reactors [7][8]. The objective of this paper is thus to provide a brief safety study framework that is consistently applied to each generation’s concepts [9]. A comparison between the PWR and SFR severe accident evolution is provided as well as the relation between governing physical parameters and the adopted mitigation provisions for each reactor concept. The research in this area thus aims to strengthen further understanding of the physical phenomena and to reduce the uncertainties affecting their quantification, with the ultimate goal of developing models that can be applied to reactor cases.

2. Overview of PWR and SFR reactor concepts

In this paper, the study will be focused on design and operation characteristics related to safety affecting accident transients. In PWRs the moderator is borated water while the SFR operates without a moderator and has a core cooled by molten sodium metal. It is generally planned that mixed plutonium and uranium oxide fuel will be used, at least in European reactor concepts. As compared with thermal-spectrum neutrons, fast spectrum neutrons more efficiently convert natural uranium, a fertile material, into plutonium, a fissile material. Existing and planned SFRs are of two types: “pool-type” reactors (such as in French experience) [9][10][11] where the primary system is totally contained in a vessel where the reactor coolant pumps and heat exchangers are immersed in sodium, or “loop-type” reactors (as in Japan experience) [11][12] where the primary sodium flows in loops connecting a main vessel with other vessels where the large components are located. Compared to PWR (for instance 1300 MW_{el}) which are commercial plants operated mostly in “loop-type”, the coolant inventory is about seven times higher in a SFR (For instance the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) demonstrator 600 MW_{el} i.e. ~2000 tons of sodium versus ~280 tons of water).

The point is that a large coolant inventory, like in “pool-type” reactors, has a significant positive impact on the reactor accidental behavior. Together with a high capacity of natural circulation flow establishment in SFR, the boiling of the total sodium mass under residual power deposition (after loss of forced circulation) takes several hours. This duration is about 15 hours to get the total coolant boiling, without taking into account natural circulation flow. This is larger time compared to about half- hour timescale for the same phenomena in PWRs. In this evaluation, it is considered that the steam generator inertia can slow down the heating because the primary and secondary circuit are strongly coupled in such black-out like transient and the PWR concept can take benefit of a large secondary side inventory assumed here to be equal to the primary one.

Moreover we note that there have been various designs of advanced PWR named GenIII/III+, which is equipped with passive cooling system, which does not rely on active systems (e.g. Advanced Power Reactor AP1000 [13]). These designs have large coolant inventory and / or water condensation system to sustain core coolant inventory, typically for three days or more.

The time delay to coolant boiling is strongly related to the fact that under nominal conditions, the margin to coolant boiling in SFR is higher compared to the operating PWR as is indicated in Table 1. Moreover, in SFR ‘pool-type’ reactors, the loss of coolant due to loss of pressurization e.g. pipe rupture is practically eliminated. Consequently, the severe accident scenarios with fast total loss of cooling are practically eliminated.

Table 1 Thermohydraulic characteristics of PWR and SFR

	SFR (EFR)	PWR
Power density in the core	~300 MW/m ³	~100 MW/m ³
Coolant outlet temperature	~550°C	~300°C
Operating pressure	Very Low pressurization, ~0.1-0.5 MPa	~15 MPa
Margin to boiling	~350°C	none
Fuel assembly design	Triangular array with hexagonal tubes	Rectangular array
Coolant inventory in vessel	Larger sodium inventory for 'pool-type' reactors as ASTRID like, ~2000 tons	~280 tons
Coolant boiling appearance under residual power and no flow	~15 h	~30 min

One can deduce from these design and operation characteristics, that to consider significant fuel core degradation and melting in SFR (in time and in whole core scale), more failures of safety systems compared to PWR should be considered. In addition to accidents with loss of cooling considered in a PWR, unprotected accident transients are always added as hypothesis to SFR severe accident scenario for evaluation. This is related to the fact that in water reactors the loss of coolant will induce negative void worth that is not always the case of sodium reactors. Thus, in a PWR the reactivity insertion accidents (e.g. control rod ejection and steam pipe rupture) is part of design-bases accidents and measures are taken to practically eliminate the whole core melting. In sodium reactors, the core configuration is not the most reactive during the nominal reactor operation. Therefore, its geometry change or coolant voiding can induce reactivity insertion. Thus, in order to be conservative in severe accident transient evaluation, no reactor scram is considered. This will be detailed in the next paragraphs.

3. Severe accident scenarios

As introduced in the previous section, the loss of cooling in PWR after reactor shutdown results in a favourable time delay to first coolant boiling. Thus, more time could be available for PWRs to apply the severe accident management procedures compared with sodium fast reactors. In order to be conservative and to cover the most severe scenario hypotheses on core degradation and its consequences, the failure of reactor shutdown is assumed in sodium reactor accident analyses. Such unprotected transients takes into account that, in SFR (contrary to PWR) the core is not in the most reactive configuration during the normal operation, thus there is a possibility that the core melting transient ends up with positive reactivity changes due to coolant boiling, clad melting and relocation and fuel compaction. Here we talk about the core disruptive accident of a SFR compared to core meltdown governed by residual power and gravity collapse in PWR core (see Fig. 1). In Fig. 1 it is illustrated how the core degradation in SFRs may lead to energetic core expansion due

to interaction between molten core and surrounding sodium. Contrary, in PWRs the coolant vaporization within the core region result in gravity driven downward molten core propagation.

In PWR and SFR, the severe accident transient is related to the loss of the first safety barrier i.e. fuel cladding. The melting of the fuel and subsequent cladding failure in severe accident transient lead to the relocation and release of radioactive materials . The safety measures are taken to preserve the integrity of second and/or third safety barrier i.e. reactor vessel and containment / basemat respectively. One of the main objectives of severe accident scenarios is thus to evaluate the risk of possible pressurization of containment and thus the risk of its failure.

Then, the failure of safety injection systems in PWR is assumed, leading to loss of coolant and core degradation and melting.

To summarize, the accident events leading to core degradation for PWR are:

- Loss of cooling and loss of heat sink combined with the loss of safety injection system (e.g. loss of coolant accident (small break/large break) or blackout without action of safety injection systems).

Accident situations, which can eventually result in core degradation in SFRs, are grouped into two main categories, referred as severe accident families, reactivity insertion (unprotected transient overpower) and core cooling failure accidents (unprotected loss of flow). Reactivity insertion can occur from the inadvertent withdrawal of one or several control rods, from sodium drainage (gas bubble moving through the core) or from any compaction movement. Core cooling failure accidents can arise from the global decrease of sodium flow by the loss of electrical power that supply the primary pumps or local blockage that prevents subassembly flow (local default accident spreading the whole core due to scram failure) These families of sequences refer to the initiating event combined always with reactor scram failure.

To summarize, the accident sequences leading to core degradation for SFR are:

- unprotected transient overpower (UTOP);
- unprotected loss of flow (ULOF);
- local cooling default accidents where the whole core propagation occurs due to scram failure (unprotected sub-assembly fault) (USAF).

For both, SFR and PWR severe accident transients, we distinguish different phases. These are illustrated in simplified format in Fig. 2 for both, sodium reactor and water reactor. Generally, in PWR, the core uncovering can be reached in hours, possibly some days, depending on the core initial state and the accident scenario. This ‘early phase’ of core degradation, in water reactors is a succession of possible physical events: heat up of uncovered core due to residual decay heat, clad deformation and failure, oxidation of metals (mainly zirconium) by steam, chemical interaction among all the materials. The late phase of core degradation in PWR is characterized by an accumulation of molten materials within the core region, forming corium pool. Also there may be a collapse of structures (fuel rods, control rods, grids...). Consequently, corium relocation into the vessel lower head may occur, with vaporization of water present in the lower head. We note here that according to severe accidents expert’s community, the term corium (fuel containing material) is only used in term of PWR and will not be used in SFR terminology since in this latter concept, contrary to the PWR, the liquid materials are not the consequences of various chemical interactions but consist of fuel oxide and steel.

In SFR safety evaluation, each family of sequences can be decomposed into four accidental phases: primary, transition, secondary and relocation phase [14][15]. This decomposition permits to better follow transient evolution and to print out driving phenomena. The primary phase is characterized by sodium boiling, fuel pin degradation, fuel and/or clad melting. During the primary phase each subassembly (SA) remain separated from others since the SAs shrouds (hexcans) keep their integrity. In the transition phase, hexcans undergo a loss of their integrity, beginning of radial material motion and beginning of molten pool formation. Subsequent hexagonal tube openings, axial and radial melt propagation is characterized for the transition phase. In late accidental phase, the core relocated outside from the core region is the scope of interest.

4. Uncertain in-vessel severe accident phenomena and R&D needs

One of the main objectives of severe accident R&D is to further understand the physical phenomena occurring during each phase and to reduce the uncertainties surrounding their quantification, with the ultimate goal of developing models that can be applied to reactor cases. Thus, we highlight here the main physical phenomena occurring during the severe accident transient in PWR and SFR reactors. The ranking of these phenomena according to their uncertainty and importance is not the scope of this paper and can be found in [16] for PWRs, for SFR the work is ongoing.

4.1 Early phase/ Primary phase phenomena

The Table 2 compares the main physical phenomena occurring during the first phases of severe accident in PWRs and SFRs. The fuel cladding in PWR reactors is made of Zircaloy (alloy of zirconium). In SFR, steel have demonstrated success or improved performance as structural components of fast reactor fuel assemblies, with particular emphasis on fuel cladding.

In the early phase of severe accident transient in PWR, cladding oxidation and hydrogen formation occur. The hydrogen produced may escape from the reactor coolant system (via a breach) and mix with containment air, which may lead to a fast deflagration or even a detonation and resulting pressure effects that can lead to a direct threat to containment integrity [7]. Consequently, the capacity to evaluate hydrogen production (instantaneous and cumulative) is a key issue in safety studies. Moreover, in PWR reactor concept, during the early phase of core degradation, some core structures as spacer grids may prematurely interact with cladding. These reactions form eutectics with a melting point below that of Zircaloy. The last phenomenon related to cladding in PWR is its rupture, due to rising temperature and fission gas formation within the pellets. This overpressure within the fuel causes cladding swelling as a result of creep. This phenomenon, called ballooning may lead to cladding rupture and fission product release [2].

In SFR severe accidents, the steel cladding melting and relocation impacts mostly core reactivity. Firstly, in case of UTOP, the location and instant of cladding rupture will impact molten fuel and fission gas release. The ejection of molten fuel through this clad opening will induce fuel coolant interaction within fuel assembly channel. Both phenomena result in voiding of fuel assembly channel leading to a fast positive reactivity insertion. In case of ULOF that is governed at the beginning by thermal effects, we may attend firstly the clad melting and relocation. Here again,

the molten steel relocation (upward or downward) along the fuel pins has an impact on neutronics, and in some case can lead to reactivity increase. Thus, these phenomena are important for further severe accident transient evaluation.

For both, PWR and SFR severe accident transients, the cladding failure/melting followed by molten fuel and cladding relocation are the processes which have the main impact on core geometry. Several experimental studies have been conducted in this area and while the level of knowledge is considerable, modelling is not yet satisfactory [2].

Table 2 Early phase /primary phase phenomena during severe accident

PWR	SFR
Early phase/ Primary phase phenomena	
Cladding oxidation and hydrogen formation Cladding rupture Molten metals and interaction with intact rods Release of fission products Zircaloy melting and fuel dissolution	Cladding melting and relocation/ neutronics Fuel Coolant Interaction (FCI) in the core zone/neutronics Fission product release/neutronics Impact of fission gas/ irradiated fuel on pin degradation and molten fuel movement
Corium progression FCI Oxidation of molten metal Molten pool formation and relocation to the lower head Hydrogen combustion Containment pressurization	Molten pool formation, in-pool stratification of steel / neutronics Impact of B4C chemistry within molten pool Heat transfer between molten pool and assembly hexagonal tubes Vapor bubble expansion

In early phase in PWR, and during the primary phase in SFR, the molten/degraded fuel and cladding relocation may start. The understanding of molten/degraded core materials progression is important for both, PWR and SFR severe accident transients. In PWR, this is related to the possibility of reflooding of degraded core within core region as severe accident management procedure [17][18]. However, here the pressure increase due to steam formation and hydrogen production are key point of interest. In SFR concepts, the molten/degraded core progression impacts, as said before, the neutronics. Thus, the important phenomena are related to molten pool formation (mixing/segregation) within hexagonal tubes, impact of B4C chemistry within this molten pool. Moreover, the heat transfer between molten pool and assembly hexagonal tubes to predict the instant of hexcan integrity loss and the first fuel discharge is important phenomena. If the fuel discharge does not occurs, the reactor power does not decrease due to ongoing neutron reaction. Thus, the power deposition into fuel can reach very high values, leading even to fuel and steel vaporization. Thus, more stress is put to research and analyses of mechanical energy release due to this vapor expansion [19] in the vessel.

4.2 Late phase/ Relocation phase phenomena

In this paragraph, for PWR severe accident transients we focus on physical phenomena related to transient phase where the degraded core relocates into the vessel lower head Fig. 3-left. In the SFR, the concept purpose is that in case of severe accident, the major degraded core mass is relocated downward into lower plenum, situated below the core Fig. 3-right. In this paragraph, the main physical phenomena in PWR and SFR are highlighted.

Table 3 Late phase /Relocation phase phenomena during severe accident

PWR	SFR
Late phase/ Relocation phase phenomena in the vessel lower head	
Corium jet fragmentation, debris formation Vapor explosion Debris bed dryout, reflooding possibilities Molten pool formation Natural convection in molten pool Focusing effect Oxidation, hydrogen formation Vessel loadings and lower vessel failure	Molten materials jet fragmentation, debris formation Vessel and DHX thermomechanical loading due to FCI in sodium plenum and bubble expansion Long term molten material coolability (mainly debris bed) Risk of recriticalities on core catcher if designed Thermal erosion of core catcher if designed Sodium fire (above the slab) Fission product transfer and recombination
Ex-vessel phenomena	
Molten corium-concrete interaction Direct containment heating Vapor explosion Hydrogen combustion Containment pressurization Fission products transfer and recombination	

In the PWR late phase of the accident, the uncertain physical phenomena are related to estimation of two main risks when corium reaches the lower head. One is that the steam produced when hot corium comes into contact with residual water will cause a pressure spike, or even an in-vessel vapor explosion. The other risk is that upon contact with corium, the vessel will undergo a heat flux, which may be locally of considerable magnitude, potentially resulting in vessel rupture [20]. In the frame of in-vessel retention principle for some reactor concepts, there is thus a critical need to predict the changes that corium will undergo, from its relocation towards the lower plenum until its cooling or its transfer out of the reactor vessel [21][22]. The main phenomena governing these changes are listed in Table 3.

It can be seen in Table 3 that some phenomena in late phase/relocation phase are common in PWR and SFR severe accident evaluation. These are related to molten material jet fragmentation, when

the discharged material gets into contact with coolant. As a consequence, the vapor formation due to FCI is an important phenomenon to be evaluated related to the risk of thermomechanical loadings on surrounding structures. Moreover, in SFR concepts the impact of jet on core catcher, in case of non-complete fragmentation is also being evaluated. In PWR as well as in SFR transients, the common key point to be evaluated is the debris bed coolability [23][26].

In PWR concepts, additional phenomena being evaluated related to molten pool arrival into the lower head ([21]): jet fragmentation and related debris size distribution, debris agglomeration, formation of lava tunnels, coolability limits of debris, 3D corium relocation, formation of molten corium nests within debris, migration of molten metallic phases, inverse stratification (controlled by zirconium oxidation and effect of B4C). The chemistry, oxidation and hydrogen production are important phenomena. The stratification of phases within the formed molten pool is studied in detail, in terms of heat flux and temperature evaluation. More precisely, so-called focusing effect where the metal layer is stratified above the molten pool, may endanger the vessel integrity due to important heat transfer towards the vessel. The main phenomenon that controls coolability is related to critical heat flux both on the surface of the corium pool, in debris beds, and in the gap which is supposed to have formed between the vessel and the crust around the corium pool. Moreover, the efficiency of debris coolability, corium pool coolability and gap cooling is questionable at low pressure [21]. The above-mentioned phenomena related to molten pool formation and coolability in PWR lower plenum are related mainly to the low water inventory within lower head and its quick vaporization.

We note here, that similar phenomena related to molten pool formation, steel stratification and heat transfer towards surrounding structure are considered in SFR, but mainly during the primary phase, regarding the heat transfer from molten pins towards hexagonal tubes structures (as fuel pins in SFR are present within FAs with hexagonal steel hexagonal tubes) as it is introduced in section 4.1.

In SFR concepts, mainly due to large inventory of sodium and its thermal properties, debris bed is considered, in most of the scenario to represent the final state (i.e. no re-melting occurs). On the other hand, more attention is paid in SFR concepts to evaluate the possible re-criticalities. So some sacrificial material is considered to be designed within this lower plenum region. The internal core catcher is a solution to mitigate almost all the SA phenomena occurring in the vessel lower head i.e. jet impingement, criticality, coolability of the debris, thermal loading.

Finally, in the case of the vessel failure, the ex-vessel phenomena are considered for both concepts. For PWRs, these are related to long term cooling issue by spreading the corium out of the vessel region, if external core catcher is considered (e.g. in Gen-III European Pressurized Reactor (EPR)). In Gen-II concepts, the phenomena to be considered are related to molten corium concrete interaction (MCCI) and direct containment heating (DCH). Both phenomena may lead to hydrogen formation, combustion, overheating and pressurization of the containment atmosphere, and may result in damage to the containment building and loss of containment integrity. Moreover, in case of MCCI, when the ablation of concrete is not considered to be stopped, there can be a risk of ground water contamination. Basically, the out-vessel issues related to PWRs are aimed at investigating the prevention and mitigation actions to foresee in order to prevent containment loss-of-integrity due to overpressure and failure of the basemat due to MCCI. Taking into account the SFR pool concept, the only ex-vessel phenomena able to build-up some pressurization deals with sodium fire in case of energetic accident. Actually, in such a situation, a sodium slug can impact

the reactor slab and burn in the room above it. This fire consists in a pressure driver able to transfer fission product in the reactor building.

Further, a common issue for both PWR and SFR is the transfer of radionuclides in the reactor building and at the end its transfer to the environment. The transfer from the molten core to the reactor building is done through the primary opening in PWRs and through the leak across the primary vessel and across the slab for SFRs. Finally, the source term that can be transferred through reactor building depends on complex physical processes that are worth investigating, like transport, deposition, wall condensation, etc.

5. Severe accident management strategies

The common interest of PWR and SFR concepts is to implement, in case of severe accident, severe accident management strategies to insure long-term stabilization of degraded core materials, that is sub-criticality and coolability. The aim of the severe accident management is to prevent the development of the accident scenario towards more severe conditions.

A comparison between the PWR and SFR severe accident evolution in Section 4 provides the relation between SA event sequence, and here the adopted mitigation provisions for each reactor concept will be introduced.

The combination of engineering judgment and probabilistic methods is used to determine the prevention and mitigation measures. Some examples of mitigation strategies for Gen-III concepts of reactors may be found for in-vessel [27][30] or ex-vessel retention [31]. For Gen-IV concepts, the in-vessel retention with internal core catcher is considered as mitigation strategy, together with devices aiming at insuring fuel discharge from the core zone. All these strategies lead to long-term stabilization and coolability and sub-criticality of the degraded core during severe accidents.

In PWR, the main idea of in-vessel retention consists in flooding the reactor pit to fully flood the vessel or at least to flood the lower head. The PWR lower head containing the molten pool is cooled from outside, which keeps the outer surface of the vessel wall cool enough to prevent vessel failure. The decay heat is transferred through the reactor pressure vessel wall to the surrounding water and then to the atmosphere of the containment of the nuclear power plant. The aim of this strategy is to localise and to stabilise the corium inside reactor vessel. The applicability of the in-vessel retention concept is only for low power density, medium size reactors such as Water cooled water moderated reactor (VVER)-440 and the AP1000 design [13]. For very large, high power density reactors (e.g. EPR), ex-vessel strategy for corium localisation and stabilisation is considered [28]. The ex-vessel corium retention is an alternative approach to localise and stabilize the corium. This approach is based either on corium retention in core catcher located in reactor pit below the reactor vessel (e.g. VVER-1000 design)[29] or corium retention in special spreading compartment located under the reactor vessel (e.g. EPR). However, preventing the spreading/dispersion of such large amounts of radioactive material appears to be more attractive in order to avoid very long and very expensive post-accident decontamination tasks. The in-vessel or ex-vessel retention strategies will prevent further progress of a severe accident, passively, with

the corium (or degraded fuel) in a stable and coolable configuration, thus avoiding the largely uncertain accident evolution with the molten debris on the containment floor. Most of the severe accident mitigation scenarios are thus based on passive safety technology. The safety systems include passive safety injection, passive residual heat removal and passive containment cooling. These systems provide long-term degraded core cooling and decay heat removal without the need for operator actions and without reliance on active safety systems.

However, the implementation of one of those passive technologies within the design of an operating nuclear power plants (Gen II) is difficult from the financial or technical points of view. Thus, the reflooding (injection of water) is possible if one or several water sources (active systems) become available during the accident. The injection of water into the core (even if degraded) could, in certain conditions, stop the progression of severe accident [32]. This may significantly contribute to the improvement of mitigation efficiency on PWR. On the other hand, it must also be recognized that at elevated core temperatures, the reflood is likely to lead to an enhanced hydrogen formation and risk of containment damage (if hydrogen/air fast deflagration occurs) [18]. However, from a safety point of view, it is important to evaluate chances of coolability of the reactor core during a severe accident. This is in line with the safety philosophy of DiD which requires to foresee and to analyse all options to stop an accident at any stage. The hydrogen risk issue may be mitigated by installation of devices for hydrogen recombination or with an inert atmosphere.

The advantage of Gen-IV reactor concepts is that during their pre-conceptual phase, the severe accident mitigation can be taken into account. Here as well, the severe accident management is based on considerations of passive systems. In order to satisfy this general objective of limiting the radionuclide releases, the aim of safety provisions is to maintain the integrity of the second barrier (at least to the progress of the corium, main primary vessel) and the leak-tightness of the third barrier (safety vessel: additional shell around the main primary vessel), and thus to reduce the risk of occurrence of severe energetic accidents that may affect these barriers. In practice, two temporal phases of the accident scenario can be distinguished, during which the confinement must be preserved:

- A short-term phase in which it is necessary to control the generation of mechanical energy which could result from the accident and could lead to a failure of the second barrier.
- A longer term phase during which the relocation and the cooling of the degraded core must be managed in order to decrease the risks of recriticality and of failure of the vessel lower head. Regarding this long term phase, keeping of the confinement implies, in particular, that enough decay heat removal capabilities are maintained.

Even if in most of the cases, the value of mechanical energy should be low versus primary vessel capabilities, it has been investigated to foresee provisions aiming at reducing it. One of the goals of the SFR safety approach is to avoid situations leading to significant mechanical energy release. For that, it is necessary to take into account for scenario variability and uncertainties on knowledge of physical phenomena. Furthermore, another design goal is to control the course of the accident scenario as much as possible and thus to manage the molten material relocation. For all these reasons, the French and Japan R&D actions aim to implement mitigation devices in the SFR reactor :

- Mitigation tubes in French SFR concepts in order to limit the amount of fissile material in the core during the secondary phase and therefore to limit the likelihood and the amplitude of a power excursion [15];
- FAIDUS concept (Fuel subAssembly with Inner DUct Structure) in Japanese SFR concepts in order to evacuate early the liquid fuel fraction from fuel assembly after the initial power excursion. The pressure is the main driver for upward liquid fuel discharge. The quick molten fuel discharge from the core region should guarantee the mitigation of energetic primary phase [33].
- A core catcher in the vessel lower head aiming at collecting, spreading and cooling molten materials in order to prevent primary circuit failure and to ensure long term cooling of the core materials [26].

6. Demonstration tools for safety studies

The numerical simulation on severe accidents generally adopts this approach:

- Integral codes or code systems that simulate the whole accident, from the initiating event to the possible release of radionuclides taking into account the main safety systems;
- Detailed or mechanistic codes that provide a more refined simulation of the one phenomenon or group of phenomena.

For reactor simulations, R&D focused on the reactor scale by seeking to develop macroscopic codes and by carrying out the most demonstrative experiments. The severe accident phenomena are very complex, related to multi-physics, multi-material, multi-scale issues and some of them are represented within the experimental database e.g. [25][20][30][24]. Gradually and following the general tendency to have increasingly accurate calculation tools, the positioning is shifting towards the approach of development of more predictive tools. In order to compensate the small number of large-scale experiments and / or prototypical materials, it is essential to follow an approach where the tools developed are based on analytical experiences, supplemented by simulations carried out at local scale. This is to develop models based on physical bases, and to feed codes to a larger scale. This approach should reduce dependency on parametric laws or correlations with a limited domain of validity. These codes must be verified, validated by parties and generally qualified on the basis of experiments to justify the extrapolation to the reactor case.

Among the integral codes used for severe accident integral analyses in PWR we can list: ASTEC, MAAP, MELCOR, SAMPSON [34][2] ...Indeed, different projects focusing on these code's benchmarks (e.g. on Fukushima accident transient [35]) have shown in some cases the differences, in prediction of key-phenomena of severe accident transient sequences. These parametric severe accident codes have been developed mainly to be used for Level-2 Probabilistic Risk Assessment where large uncertainties are taken into account in probability distributions for the uncertain parameters.

For the SFR, the value in R&D of calculation tools is put on combination of probabilistic and deterministic approaches for demonstration of robustness of severe accident mitigation strategies [15]. Among the integral codes for severe accident transients, there is a plan to built SEASON platform that includes detailed mechanistic code SIMMER (Sn Implicit Multiphase

Multicomponent Eulerian Recriticality) [14][21]. In term of verification studies on severe accident mitigation strategy, the difference between the safety criterion of the considered component and calculation results on phenomena impacting this criterion is checked (see Fig. 4).

This latter is assessed through a mechanistic calculation (SIMMER, eventually coupled with other codes within SEASON platform). Additionally, the spreading of this result is assessed through physico-statistical calculations (planned so called PROCOR-Na tool [22]) that have to demonstrate that the risk to exceed the criterion is low enough in term of probability. Those physico-statistical calculations will encompass the various sources of uncertainty on physical models and on state of the reactor when the accident occurs. So they will enable to investigate the integrality of the possible scenario evolutions by propagating uncertainties in a fast running tool. Based on a large set of calculations (typically several thousands), a statistical treatment of the calculation results will permit to estimate the probability to exceed a limit on a safety barrier. Finally, the combination of the difference between the best-estimate calculation and the criterion and the risk that the calculated parameter exceeds the criterion gives the design margin [36][37].

7. Conclusions

The prevention and mitigation of severe accidents is being studied for existing and future nuclear power plants. In this paper, the key phenomena concerning severe accident phenomena on Sodium Fast Reactors and Pressurized Water Reactors are compared. Firstly, the design and operation characteristics of both reactor types are summarized. It can be seen that the sodium inventory and its thermal properties provide high safety margins, especially once the reactor shutdown is delayed. Moreover, contrary to PWR, the SFR core is not in the most reactive configuration during the normal operation. Thus, there is a possibility that the transient ends up with positive reactivity changes due to coolant boiling, clad melting and relocation and fuel space concentration. In SFR we talk about the core disruptive accident compared to core meltdown under residual power and gravity collapse typical for severe accident transient in PWR core. In time scale, it is important to note that the severe accident transient, from the instant of coolant boiling up to the instant of whole core melting can last a time period of the order of seconds/minutes in SFR compared to hours/days in PWR.

The uncertain in-vessel phenomena related to severe accident progression are identified and R&D needs are highlighted. It can be seen that some of the phenomena are common for both, SFR and PWR severe accident studies. These are related e.g. to identification of heat flux between molten pool and surrounding structures, molten steel/zircaloy relocations, fuel coolant interactions, liquid fuel jet and fragmentation in coolant, debris bed coolability, etc. On the other hand, the safety analyses of PWR should deal with hydrogen risk (due to clad oxidation) that is not the case of SFR. In sodium fast reactors, the reactivity effects related to changing of core geometry are the key point to be treated in severe accident transient analyses. Actually, the power released into fuel is not fixed and can reach very high values, leading even to fuel and steel vaporization. Thus, more stress is put to research and analyses of mechanical energy release due to vapor expansion.

In both reactor concepts, a severe accident mitigation strategy is proposed to assure the long-term stabilization, sub-criticality and the coolability of degraded core materials. This is based on assumption of core-catcher and/or in-vessel retention principle. For this purpose, the withstanding of safety barriers to thermal and mechanic loading is a key point for safety analyses. Consequently, the safety design measures are being proposed, supported by reactor calculations. The safety analyses and safety demonstration and identification of risk and margins are based on combination

of probabilistic and deterministic approaches. The computer codes used for calculations are still being improved and the objective is to improve knowledge about uncertain parameters. For this purpose, the large R&D programs all over the world are being realized to improve the knowledge and to share the understanding of severe accident phenomena in order to propose mitigation measures for any type of reactor.

Nomenclature

AP	Advanced Power
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
ASTEC	Accident Source Term Evaluation Code
B4C	Boron Carbide
CEA	French Alternative Energies and Atomic Energy Commission
DCH	Direct Containment Heating
DER	Department of Reactor studies
DHX	Decay Heat Exchanger
DiD	Defence in Depth
EPR	European Pressurized Reactor
FAIDUS	Fuel subassembly with Inner DUct Structure
FCI	Fuel Coolant Interaction
Gen	Generation of nuclear power plants
IRENE	Research Institute for Nuclear Systems for the Production of Low Carbon Energy
MCCI	Molten Corium Concrete Interaction
PWR	Pressurized Water Reactor
R&D	Research and Development
SA	Subassembly
SFR	Sodium Fast Reactor
SIMMER	Sn Sn Implicit Multiphase Multicomponent Eulerian Recriticality
ULOF	Unprotected Loss Of Flow
UTOP	Unprotected transient overpower
USAF	Unprotected Sub-Assembly Fault
VVER	Water cooled water moderated reactor

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Figure Captions List

Fig. 1 Illustration of core degradation process under severe accident in SFR and PWR

Fig. 2 Different phases of severe accident transient

Fig. 3 Late phase/ relocation phase of severe accident into lower plenum, comparison of PWR-left and SFR-right [26]

Fig. 4 Combination of mechanistic and physico-statistical calculations for margin assessment

Table Caption List

Table 1 Thermohydraulic characteristics of PWR and SFR

Table 2 Early phase /primary phase phenomena during severe accident

Table 3 Late phase /Relocation phase phenomena during severe accident

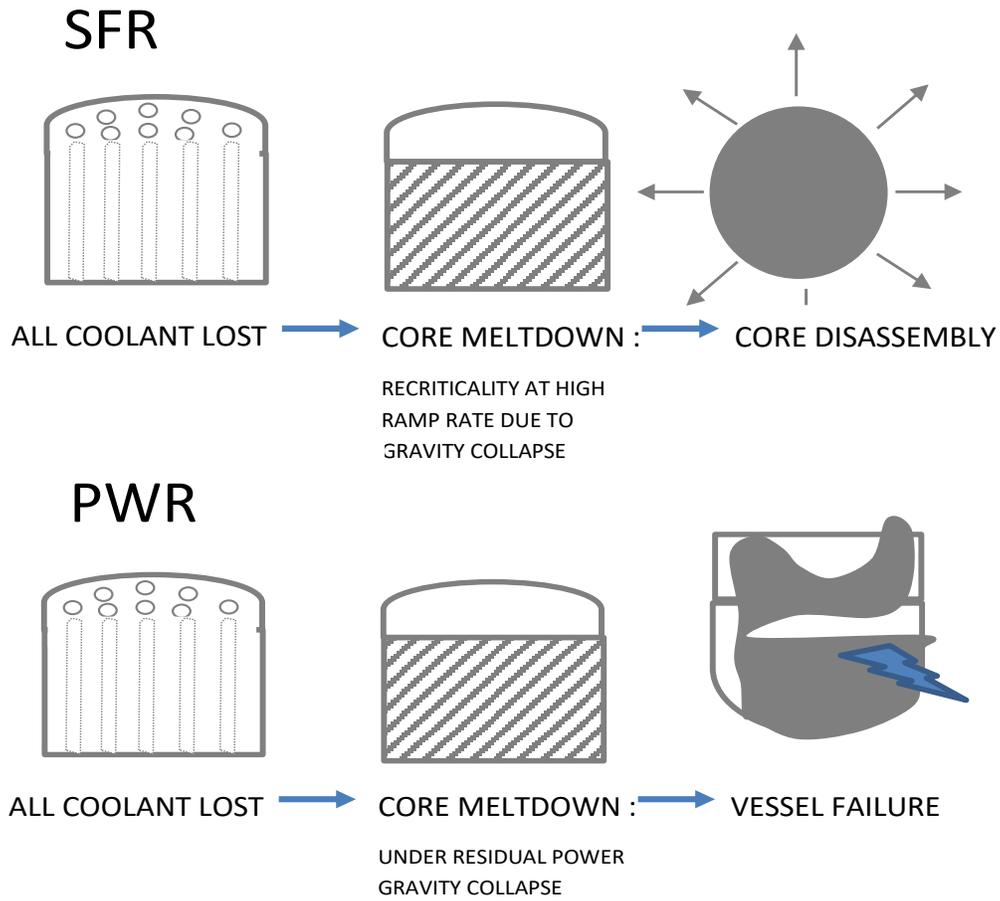


Fig. 1 Illustration of core degradation process under severe accident in SFR and PWR (pressure driven versus gravity driven)

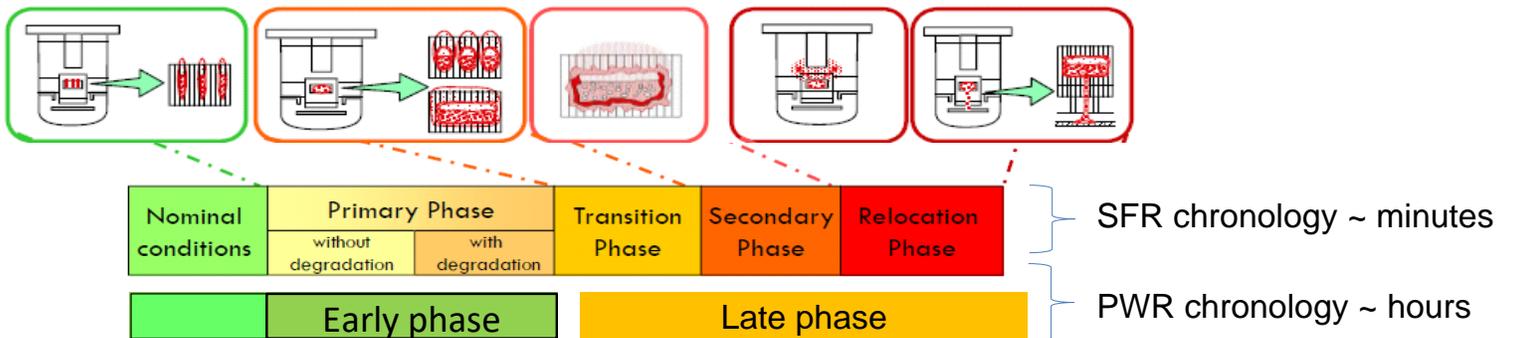


Fig. 2: Simplified schematic view of different phases of severe accident transient

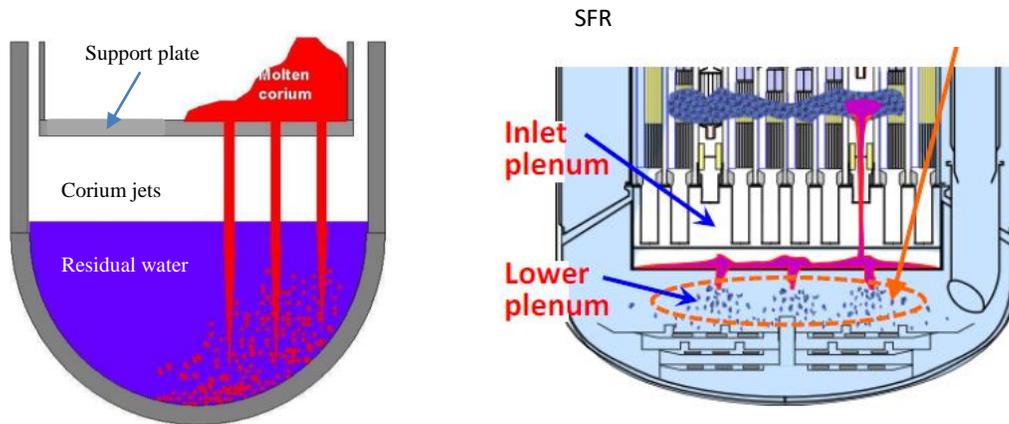


Fig. 3: Late phase/ relocation phase of severe accident into lower plenum, comparison of PWR-left and SFR-right with internal core catcher [26]

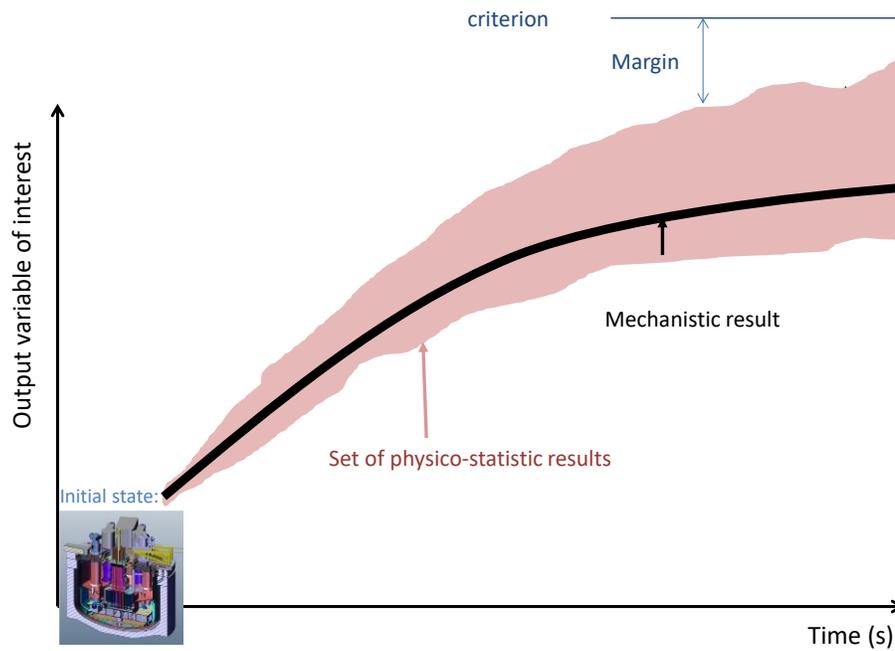


Fig. 4: Combination of mechanistic and physico-statistical calculations for margin assessment