Status Report on Ex-Vessel Steam Explosion

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NEA
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THE COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) shall be responsible for Nuclear Energy Agency (NEA) programmes and activities that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations, with the aim of implementing the NEA Strategic Plan for 2017-2022 in its field of competence.

The Committee shall constitute a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It shall have regard to the exchange of information between member countries and safety research and development (R&D) programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee shall review the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensure that operating experience is appropriately accounted for in its activities. It shall initiate and conduct programmes identified by these reviews and assessments in order to confirm safety, overcome discrepancies develop improvements and reach consensus on technical issues of common interest. It shall promote the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings (e.g. joint research and data projects), and shall assist in the feedback of the results to participating organisations. The Committee shall ensure that valuable end-products of the technical reviews and analyses are provided to members in a timely manner, and made publically available when appropriate, to support broader nuclear safety.

The Committee shall focus primarily on the safety aspects of existing power reactors, other nuclear installations and new power reactors; it shall also consider the safety implications of scientific and technical developments of future reactor technologies and designs. Further, the scope for the Committee shall include human and organisational research activities and technical developments that affect nuclear safety.

The Committee shall organise its own activities. Furthermore, it shall examine any other matters referred to it by the Steering Committee. It may sponsor specialist meetings and technical working groups to further its objectives. In implementing its programme the Committee shall establish co-operative mechanisms with the Committee on Nuclear Regulatory Activities in order to work with that Committee on matters of common interest, avoiding unnecessary duplications.

The Committee shall also co-operate with the Committee on Radiation Protection and Public Health, the Radioactive Waste Management Committee, the Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle, the Nuclear Science Committee, and other NEA committees and activities on matters of common interest.

LIST OF ABBREVIATIONS AND ACRONYMS

ALISA	Access to Large Infrastructures for Severe Accidents
BWR	Boiling water reactor
CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives
CSNI	Committee on the Safety of Nuclear Installations
DEFOR	Debris bed formation
DNS	Direct Numerical Simulation
DROPS	FCI- Droplet Experimental facility
EC	European Commission
EDF	Électricité de France
EPR	A PWR design by Areva NP and EDF
EVSE	Ex-Vessel Steam Explosion
FARO	FCI-Experimental facility-JRC-Ispra-Italy
FCI	Fuel-coolant interaction
FP-6, FP-7	Framework Program (European Commission research programmes)
ISP	International Standard Problem
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (France)
KAERI	Korean Atomic Energy Research Institute
KROTOS	FCI-Experimental facility CEA-Cadarache-France
LWR	Light water reactor
NEA	Nuclear Energy Agency
O/M	Oxide-to-metal ratio
OECD	Organisation for Economic Co-Operation and Development
PLINIUS-2	CEA future large-mass experimental platform
PWR	Pressurised water reactor
RCS	Reactor coolant system
RPV	Reactor pressure vessel
R&D	research and development
SAFEST	Severe Accident Facilities for Europe Safety Targets
SARNET	Severe Accident Research Network of Excellence
SE	Steam explosion
SIGMA	FCI- Droplet Experimental facility
TOP	Technical opinion paper
TROI	Experimental facility KAERI, Dejon, Korea
WGAMA	Working Group on the Analysis and Management of Accidents

TABLE OF CONTENTS

EXECUTIVE SUMMARY

The Status Report on Ex-Vessel Steam Explosion (EVSE) summarises the significant progress made in the field of Fuel Coolant Interaction/Steam Explosion (FCI/SE) knowledge during these last ten years, especially in the framework of the OECD/SERENA-Phase 2 program (2007-2012).

Key phenomena for steam explosions have been ranked for their knowledge base and possible consequences for the containment building. From this ranking, it can be stated that jet orientation and fragmentation, triggering, solidification and oxidation phenomena are considered to have limited knowledge, but also high importance on SE energetics.

The consequences on the integrity of the containment building due to an ex-vessel SE have been also reviewed. The consequences of ex-vessel steam explosion can be divided in two categories: direct consequences (mechanical constraints on the concrete structures, typically the reactor pit and on the reactor vessel) and indirect consequences (motions of heavy reactor circuit components such as reactor vessel or steam generators that can affect leak tightness of the containment). It is also important to consider impact on medium and long term severe accident management e.g. with regard to possible damage of equipment responsible for long term cooling of the debris. The methodologies used to estimate the consequences of ex-vessel steam explosions in pressurised water reactors (PWRs) and boiling water reactors (BWRs) and the maximum acceptable impulse load on the containment are at least slightly different among countries, which emphasises the difficulty in reaching a common international approach on this point.

After Fukushima-Daiichi severe accidents, the regulatory bodies in the participating countries were surveyed on the existing regulatory requirements for steam explosion. Main observations from this survey responses were: (1) the regulatory requirements varied between countries; (2) in most countries, if not all, the consideration of ex-vessel steam explosion remains an open issue, mainly due to unresolved uncertainties; (3) the methodologies available for steam explosion risk assessment have the same basic elements but the complexity and sophistication varied among countries: the existing methodologies used a deterministic assessment of steam explosion loads (impulse and/or pressure) with a structural mechanics assessment of containment fragility and a probabilistic framework to evaluate the risk of containment failure associated with a steam explosion; and finally, (4) the need for and scope of continued research on the subject varied between the countries.

In spite of the important progress made in the frame of the OECD/SERENA-Phase 2 program, the knowledge is still limited on some important aspects of EVSE and therefore the impact of this issue on containment building integrity.

There are ongoing new FCI/SE programs, even after the OECD/SERENA-Phase 2 program: in 2017 at national level in France ICE-Post-Fukushima program, and in Sweden; and at international level in Europe, the SARNET network in the frame of NUGENIA program, providing exchanges among the experts and promoting access to some of the facilities in Europe in the frame of the European SAFEST (2014-2018) and ALISA (2015-2018) programs. Korea finished their national FCI program in early in 2017.

Small scale experiments are suitable to improve models currently used in FCI codes, and these improvements can then be validated using large scale experiments. Among the experiments that are needed to improve analytical models, experiments dedicated to study the oxidation of single drops, the

solidification of single drops, the fragmentation of prototypical drops in reactor conditions, and film boiling at very high temperature are more useful.

In the short term (2017-2021), it seems useful to build an international program in order to address some of the required improvements and the needs identified for reactor applications: additional tests could be performed in the TROI and KROTOS facilities with a focus on the impact of more realistic corium composition on SE energetics and debris bed cooling (debris coming from either premixing or explosion). Among the potential prototypical corium tests, the useful tests are with corium compositions with large solidification intervals and sub-stoichiometric composition. These tests could be performed taking into account conditions more representative of ex-vessel scenarios (for example higher initial velocity of the melt, "real" water with boric acid and painting aerosols, and submerged vessel). Experimental tests in "stratified" configuration could also be considered.

In the middle term (2022-2026), main FCI/SE phenomena at large scale (80-120 kg) with different corium compositions -including metallic corium- would allow reliable qualification and validation of the EVSE modelling for the extrapolation to reactor case. Configurations which have scarcely been evaluated previously (i.e. pressurised and inclined jets, partial melt stratification, lateral jets impacting on walls) could be studied. These experiments will also be representative of intermediate scale between KROTOS/TROI facilities and reactor scales, and thus would allow more reliable qualification and validation of the EVSE modelling for the extrapolation to reactor case. The future CEA/PLINIUS-2 experimental platform should be able to answer this final and indispensable step of validation for premixing and steam explosion for reactor case code application. Additionally, international collaboration on such large scale programs would be desirable in order to rationalise research efforts.

ACKNOWLEDGEMENTS

I. INTRODUCTION

The interaction of the reactor core melt with water for light water reactors (LWRs) (Generation 2&3), known as molten fuel-coolant interaction (FCI), is one of the most complex technical phenomena, involving a coupling between thermal-hydraulic and physico-chemical phenomena. FCI may occur for Ex-Vessel configuration, when the reactor vessel lower head fails and a molten corium jet is ejected into a flooded cavity in the reactor pit. Each of the severe accident scenarios with vessel failure and water in the cavity may lead to an energetic FCI, commonly known as "steam explosion" (SE), which represents potentially serious challenge to the reactor containment integrity and severe accident management. Both energetic and non-explosive FCIs result in melt fragmentation, quenching and formation of a debris bed in the reactor pit.

This Status Report is devoted to ex-vessel steam explosion (EVSE) status of knowledge for LWRs reactor applications. It has been prepared by the main experts involved in SERENA-2 program (*Steam explosion* resolution for nuclear applications), belonging to Committee on the Safety of Nuclear Installations/Working Group on the Analysis and Management of Accidents of OECD /NEA/CSNI/WGAMA. The expressed experts' opinions are summarised here in support of regulatory decision making process for the nuclear power regulators in addressing the FCI issues consistent with their own regulatory philosophy.

The EVSE Status Report begins with a brief historical perspective of the progress made on FCI/SE, in particular, the knowledge gained on the subject through decades of research. Then, it summarises the findings obtained through the framework of the OECD/SERENA-Phase 2 program. Expert's ranking of the EVSE key phenomena and evaluation of their status of knowledge are then summarised. In another section, consequences of EVSE on containment and reactor building integrity are presented, in particular to evaluate plant specific potential margin in case of ex-vessel steam explosion and the degree of accuracy necessary for the evaluation of Steam Explosion energetics. Approaches to addressing the EVSE issues by various national regulatory bodies after Fukushima Daiichi severe accident are then summarised. Finally, main improvement needs for reactor applications are identified.

II. STATUS OF THE UNDERSTANDING OF THE EX-VESSEL
STEAM EXPLOSION (EVSE)

II.1   Short historical perspectives

Historically, operating light water reactors (LWRs) in many OECD countries were commissioned at a time when steam explosion (SE) did not surface as an issue. Consequently, there were no regulatory requirements set forth for these reactors at the time.

However, since the identification of the issue in WASH-1400 (USNRC, 1975), the safety analysis practice in most of all OECD countries included an assessment of SE risk. Specifically, containment failure from an impact generated by the failed reactor upper head as a result of an in-vessel steam explosion event and subsequent missile generation was designated as the alpha-mode failure.

In 1985, the first Steam Explosion Review Group (SERG-1) (USNRC, 1985) was convened to deliberate on the probability of alpha-mode failure, based on the then available information.

The second Steam Explosion Review Group (SERG-2) (NUREG, 1996) was convened in 1995 to revisit the deliberations from SERG-1 in light of the technical progress made and new findings therein. An important conclusion from the SERG-2 (NUREG, 1996) meeting was that the alpha-mode failure issue was considered to be resolved from a risk point of view, meaning that this mode of failure is of very low probability. The alpha-mode failure is considered to be of little or no significance to the overall risk from a nuclear power plant, and that any further reduction in residual uncertainties is not likely to change the probability in an appreciable manner.

The CSNI ISP-39 International Standard Problem (NEA, 1998) exercise has been completed on the test FARO L-14 to benchmark the predictive capabilities of computer codes used in the evaluation of FCI and quenching phenomenologies of relevance in water cooled reactors accident safety analyses. The exercise demonstrated a large spread between FCI code predictions for the first time.

A previous Technical opinion paper (TOP) devoted to FCI status was published by OECD/NEA (NEA, 1999) in March 2000. Different topics relevant to nuclear safety were analysed in this paper including quenching and coolability of corium, and energetic FCI. While it was recognised that the experimental data base with prototypic reactor core material was insufficient, concerning energetic FCI the general opinion was that the consequences of steam explosion under reactor conditions may not be as severe as previously envisioned in the 1990's. The March 2000 TOP recommended performing integral small-scale and large-scale experiments with prototypical materials to verify and to assess the models and codes performances, and also to apply the FCI codes at the reactor scale. It was believed that the material characteristics play a fundamental role in SE explosivity and energetics. Furthermore, it was advised to focus on improving both probabilistic and deterministic methodologies, and on better understanding of the physics of corium jet break-up and fine fragmentation. Finally, there was a general consensus that fully validated codes, while desirable, were not an absolute requirement for reactor applications and that a fit for purpose validation approach would be sufficient.

In the framework of the OECD/SERENA analytical program (NEA, 2007) (SERENA-Phase 1, 2002-2005), FCI integral codes have been applied to experimental tests and reactor cases. Initially, selected pre-mixing and explosion tests were calculated and the results compared, for verification of the tools and for

identification of the common areas where uncertainties impacted the explosion results. Then, two reactor cases for "PWR-like" configurations were chosen: in-vessel and ex-vessel steam explosions (EVSE). For the in-vessel configuration, whatever the modelling and numerical approaches may be, all codes were able to calculate the reactor events of concern. Despite the different choices for setting the code parameters for the reactor applications and respective spread of the predicted pressures and impulses, all the calculated loads were below the mechanical failure of a nuclear reactor vessel.

These results showed that the safety margin for in-vessel FCI might be sufficient for the configuration and scenario selected in the frame of SERENA-Phase 1. For the ex-vessel configuration, the situation was different: the calculated loads were in some cases above the load bearing capacity of typical cavity walls for the reactor containment. The wide scatter (1-2 order of magnitudes) of the predicted loads raised the problem of the quantification of the safety margin for EVSE. The major sources of uncertainties concern the characteristics of the flow regime in the pre-mixing phase, especially void behaviour, and of the fragmentation of corium melts in the explosion phase (due to material effect). In particular, it was planned that confirmation for an extended range of corium melts that the void fractions in the mixing area and fragmentation characteristics would confirm the "mild" characteristic of explosion with reactor melts. At the end, the SERENA-Phase 1 report further confirmed the conclusion of SERG-2 but recommended prototypic material experiments to provide data base for improvement of analytical tools.

II.2 Status of the EVSE knowledge gained in the frame of SERENA-Phase 2

The OECD/SERENA-Phase 2 (NEA, 2015) was carried out in 3 tasks (experimental, analytical, and reactor application) and focused on EVSE studies, especially on the two following phenomena, partially understood and identified as having impact on SE energetics:

- formation of the void (steam content and spatial/time distribution) on pre-mixture/fragmentation stage and its impact on steam explosion;

- physical relationships between corium melt materials compositions and-pre-mixture/ fragmentation and steam explosion.

II.2.1 Experimental task

Complementary experiments were performed in the two FCI facilities and using prototypical corium[1]: KROTOS, at the *Commissariat à l'Energie Atomique et aux Energies Alternatives* (CEA) in Cadarache (France), and TROI at Korea Atomic Energy Research Institute (KAERI) in Daejeon (Korea). The KROTOS facility features one-dimensional behavior of mixing and explosion propagation, and allows the characterisation of premixing (melt and void distribution), escalation, and propagation behavior. The TROI facility is more suited for testing the FCI behaviour in multi-dimensional geometry.

The experimental test matrix (see Table 1) was composed of two series of 6 tests: one in the KROTOS configuration with about 3-5 kg of prototypical corium melt, and the other in TROI configuration with about 15-20 kg of prototypical corium melt. To guarantee the occurrence of a steam explosion, all tests in both facilities were artificially triggered just before melt-bottom contact of the test section.

Four corium material compositions representative of reactor case conditions were chosen (see Table 1). The first composition was the 70% UO_2-30% ZrO_2 so-called *"eutectic"* composition, a reference case identified as having spontaneous and more energetic explosions in TROI conditions (Song et al., 2003) than the second composition 80% UO_2-20% ZrO_2 so-called *"non-eutectic"* composition. Two new prototypical corium compositions, more representative of reactor conditions, were chosen: one sub-oxidized composition (so-called *"corium C-50":* 50% of the zirconium assumed to be oxidized in the

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[1]. Prototypical: corium with nuclear fuel UO_2.

vessel, final oxide-to-metal ratio O/M =O/(U+Zr)<2.00), and the last one including *low volatile oxide fission products* (source term representative of PWR spent fuel) and *oxidized iron* (internal components, vessel).

The void fractions were measured in both facilities using different measurement techniques: in the KROTOS facility, an X-ray source[2] able to distinguish water, void (mainly steam) and corium, and in the TROI facility a differential pressure transducer system. For the first time, it was possible to have access to the local void and corium volume fractions (spatially and in time) in the KROTOS facility during the premixing/fragmentation stage.

Table 1: OECD/SERENA-Phase 2. KROTOS and TROI experimental test matrix

Series		KROTOS (KS)	TROI (TS)
1	Challenging conditions	Standard geometrical conditions High system pressure (0.4 MPa)	High system pressure (0.4 MPa) Reduced free fall (melt jet velocity) and thick melt jet
		Material 1: 70% w/o UO$_2$-30% w/o ZrO$_2$	
2	Geometry effect Effect of geometry by comparison between KROTOS and TROI	Standard conditions: jet of diameter 3 cm	Large jet at penetration (5 cm)
		Material 1: 70 %w/o UO$_2$-30% w/o ZrO$_2$	
3	Reproducibility tests	Idem Test 2	Idem Test 2
4	Material effect Oxidic composition	Standard conditions	Large jet at penetration (5 cm)
		Material 2: 80% w/o UO$_2$-20% w/o ZrO$_2$	
5	Material effect Oxidation/composition	Standard conditions	Large jet at penetration (5 cm)
		Material 3: 80% w/o UO$_2$-11.5% w/o ZrO$_2$+8.5%Zr	
6	Material effect Large solidus/liquidus ΔT	Standard conditions. Effect of fission product: higher melt superheat	Large jet at penetration (5 cm). Effect of fission product: higher melt superheat
		Material 4: 73% w/o UO$_2$-20% w/o ZrO$_2$ +4% FeO+ 3%low volatile fission products	

The SE efficiency (expressed in terms of *conversion ratio η*, representative of a given geometry)[3] spanned a range between 0.1% and 0.6%. Compared to previous tests performed in KROTOS and TROI facilities using prototypical corium, the maximum value of the conversion ratio was higher. This was attributed to the melt pouring conditions in both facilities in the newer experiments, allowing larger

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[2]. Linatron X-Ray device-high energy: E = 6-11MeV - 0.33Gy/s.

[3]. Steam explosion efficiency ratio (%), a criterion for SE energetics qualification for a given geometry:

$$\eta = \frac{E_{kinetic}}{E_{thermal}}$$

overheated melt masses to be delivered in the test section and participate in a steam explosion, all the other parameters being equal. It can be noted that these values are lower than for those obtained in previous KROTOS experiments with simulant material (e.g. alumina, maximum efficiency: η ~2%), and considered as conservative reference case in the absence of oxidation effects and neglecting the so-called *"material effect"*. Experts have concluded: prototypical corium compositions used in previous tests did not encompass sufficient variations to support conclusively the "mild" explosive behavior of prototypic corium observed up to now for the same geometry.

TROI and KROTOS tests series 1, 2, 3, 4 have shown same behavior for "eutectic" (Material 1) and "non-eutectic" (Material 2) corium compositions for the steam explosion. TROI and KROTOS tests series 5 (Material 3) have shown very low energetics release, stressing the importance of the oxidation mechanisms (heat release and hydrogen production) on SE and its potential to limit the mechanical loads. TROI and KROTOS tests series 6 (Material 4) have shown different behaviour, stressing the importance of solidification phenomena on SE energetics: for TROI experiment, the highest energetic SE of the SERENA-Phase 2 series program has been observed (η~ 0.6%), whereas it was not the case for KROTOS test.

These two last series of tests, argued to be more prototypic of reactor situations, have shown the relevance of the oxidation and solidification processes as well as the importance of the corium composition on SE energetics.

The OECD/SERENA-Phase 2 program generated important new data and understanding of ESVE, showing the impact of more realistic corium compositions on SE energetics. At the end of the OECD/SERENA-Phase 2 program, it was not possible to propose a definitive modelling of the oxidation and solidification phenomena in SE due to the limited number of experimental tests (only one integral test in each facility). Complementary experimental data using prototypical corium (current TROI and KROTOS facilities) would be desirable as they would likely improve our understanding of oxidation and solidification phenomena, and help develop appropriate models for implementation in FCI codes, and in the future, at larger scale (CEA-PLINIUS-2 project) development and validation of respective models in FCI integral codes.

II.2.2 Analytical task

The main results obtained can be summarised in the following points:

- *Jet fragmentation and void behavior in subcooled pool conditions*, is confirmed once again as a key phenomenon that determines the amount of melt mass participating in explosion. Experts noted that improvements in modeling jet fragmentation are highly desirable if further reduction of uncertainties in energetics is warranted for specific safety applications. At the same time, some experts also opined that a simplified approach as is currently adopted, based on global fragmentation correlations, might be adequate for some reactor applications.

- *Melt solidification* is a major contributor to the limitation of energetics for oxidic and sub-stoichiometric corium melts. Attempts to incorporate the solidification effect in FCI codes led to important model improvements, e.g., modeling of crust formation and its impact on fine fragmentation, and modeling of drop size distributions to capture effect of different solidification times. It is also necessary to confirm the limiting effect of solidification for a wider range of materials, especially prototypical corium material with large temperature interval of solidification such as the material 4 of the SERENA-Phase 2 experimental test matrix, more representative of reactor case.

- *Melt density* as a material property is considered by some experts to be a key parameter in explaining the material effect on explosion energetics in previous KROTOS experiments (prototypical corium, alumina). The high melt density of oxidic prototypic core melts yields smaller droplets, thus faster solidification and higher voiding. This may explain the lower energetics of the explosion with prototypic melts. Another explanation has been proposed (Piluso et al., 2005): the difference of behavior between simulant materials and oxidic prototypical materials could be linked to their ability to be *supercooled* (i.e. to remain liquid below the melting point). Alumina is known as being a material with a high supercooling (more than 600 K) whereas few data exist on ability to be supercooled for oxidic corium compositions.

- *Oxidation effects* in codes needs development of modeling and complementary experimental data for validation, especially for sub-stoichiometric corium melt (O/M<2.00).

II.2.3 Reactor application task

With regard to reactor applications, in the framework of SERENA-Phase 2, the scattering between FCI codes was reduced as a result of better understanding and modelling of the premixing/fragmentation phase. Nevertheless, the range of pressures and impulses due to steam explosion are still of the same order of magnitude which may induce some damage to the reactor cavity pit and could have impact on containment integrity. While the voiding phenomena did not play an important role at the experimental scale, its importance at the reactor scale cannot be ruled out. There is also a need to confirm the importance of solidification. If the effect is moderate at experimental scale, the calculations indicate a strong mitigating effect at reactor scale (the proportion of solidified drops increases with the scale). Reactor case calculation with FCI codes, performed for PWRs and BWRs ex-vessel generic configuration, have shown impulses of 150 kPa.s at bottom and 125 kPa.s at wall for PWRs and 110 kPa.s at bottom and 90 kPa.s at wall for BWRs. For impulses to remain on the safe (not too conservative) side, the level of dynamic load to the cavity wall needs be lower than 100 kPa.s.

It is finally recognized that consistent improvements can be obtained by discretising the drop population in a multi-size approach (codes MC3D, JEMI, JASMINE). The large size spectrum allows in itself a reduction of the uncertainty.

Finally, there is still an outstanding issue concerning FCI code predictions at the reactor scale, stemming primarily from the basis of extrapolating the experimental results to reactor scale. Expert judgments considering also experimental results are to be part of any safety demonstration for SE.

II.3 Ranking and knowledge of EVSE key phenomena

Based on the findings from the OECD-SERENA program (NEA, 2015) and also from the European SARNET programs and other national programs (ongoing as well as recently completed), the experts have ranked, for this status document, the importance of key EVSE phenomena as well as assessed the status of knowledge. Especially over the past few years (2012-2016), important collaborative efforts have been achieved to improve a common understanding and modeling of SE (Meignen et al., 2014) in the frame of EC-SARNET 1 (FP-6) and 2 (FP-7). For this section, it is important to note that all relevant phenomena are identified in the list regardless of their importance ranking. However, it should be noted that the ranking of importance was not necessarily a consensus one for some phenomena, neither was the assessment of knowledge for the experts involved in the EVSE status document.

Table 2 describes the status of the knowledge and the ranking on the main properties having a role during FCI/SE phenomena. Table 3 describes the main reactor case scenario parameters having an impact on ex-vessel steam explosion energetics and the status of knowledge. Table 4 is focused on the main

phenomena occurring during the premixing/fragmentation and steam explosion steps and the status of knowledge.

Table 2: *Main properties having an impact on ex-vessel steam explosion phenomena*

Properties	Ranking[a]	Status of knowledge[b]
Liquidus temperature	H	M
Solidus temperature	H	M
Other thermodynamic data (cp, thermal diffusivity…)	H	I
Density	H	M
Melt undercooling ability	M	I
Surface tension	H	M
Viscosity	L	M
Emissivity	L	I

Three levels have been established to quantify the degree of relevance:

a. <u>Ranking</u>:   H: High   M: Medium   L: Low

b. <u>Status of knowledge:</u> S: Satisfactory M: Medium   I: Insufficient

For the properties involved in SE, the following can be added:

- *The liquidus and solidus temperatures, and thermodynamic parameters* are of first importance for SE energetics evaluation because these data are used for solidification points or cooling modelling. The level of knowledge is not the same according to the nature of corium, and, from a general point of view some essential thermodynamic data of the corium liquids are still missing.

- *The ability of supercooling/formation of metastable solids* according to the corium compositions and cooling conditions will play a role on SE. Taking into account the nature of corium (metallic, sub-stoichiometric or stoichiometric), different behaviors may exist; small analytical experiments are useful to establish criterions of supercooling according to the nature of corium.

- *The thermophysical properties* of the corium are partially known from general point of view. As for thermodynamic properties, it depends on the nature of the corium. Experts note that surface tension is the key parameter for fragmentation process. In contrast, viscosity (dynamic viscosity-liquid state), and emissivity, are expected to be of limited importance.

Table 3: Main scenario conditions having an impact on ex-vessel steam explosion energetics and status of knowledge

Reactor scenario conditions	Ranking[a]	Status of knowledge[b]
Jet velocity	H	M
Jet diameter	H	I
Break location (orientation of corium jet)	H	I
Melt (corium properties)	M-H[4]	M
Melt superheat	H	S
Coolant (water) temperature	H	S
Coolant (water) impurities	M	M
Water level	H	M
Containment pressure	M	M

For the reactor scenario conditions, the following comments can be added:

- ***The melt jet fragmentation*** occurs mainly once it penetrates the water. Jet velocity at the entrance into the water impacts strongly the corium fragmentation process and so the SE energetics. Experimental data on corium jets for ex-vessel configuration have been essentially obtained in vertical configurations with gravity driven velocity assuming scenarios for which the reactor vessel will fail through a central hole in the lower plenum. An important experimental data base now exists for a given range of prototypical corium, especially thanks to SERENA/Phase 2 program. Nevertheless, other conditions of failure of the reactor vessel have to be considered for PWRs and BWRs modifying the corium fragmentation phenomena, as for example:

  - ***"Focusing effect" and lateral break***: non-vertical jet, non-round jet, presence of vessel thermal insulation and of concrete vertical wall for the first impact of the corium jet. This mode of failure is mostly relevant to the scenarios where external vessel cooling is provided

  - ***Instrumentation tube guide failure****:* turbulent conditions, formation of spray, non-vertical jet.

  - ***Pressurized jets***: velocity of the jet is driven by residual pressure difference between inside and outside the vessel. The effect is expected to be as non-negligible on premixing-fragmentation phenomena, based on expert judgement. Unfortunately the knowledge is currently very limited on this topic due to the lack of experimental data and parametric laws in this configuration.

- ***The coolant temperature and the water sub-cooling*** play an important role on steam explosion energetic. Some already existing experimental data have shown the importance of this parameter on void formation.

- ***The nature of the water*** can also have an impact. Up to now, all the FCI/Steam explosion experiments performed all over the world since 25 years used pure distilled water whereas in the real reactor severe accident conditions, water will be a more complex liquid with paint

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[4]. *Corium properties: ranking Medium to High, depending on specific properties considered (see Table 4).*

constituents, fission products, boric acid, sea water, like in Fukushima Daiichi. The main properties of the water will be the same order of magnitude (e.g. with H_3BO_4 or NaCl), but the *formed steam*, and especially the interface properties between the steam and the water or the corium droplet, could be different, having possible impacts on SE phenomena. Further knowledge regarding the impact of realistic water composition on steam explosion energetic might be obtained with small scale experiments. Previous work with non-nuclear materials and specific conditions for the water (adding additives) has shown the suppression of steam explosion triggering (Furuya and Kinoshita, 2002).

- *The level of water in the reactor pit* plays an important role on steam explosion energetics for PWRs and BWRs. Two families of scenarios can be considered for LWRs reactor pit: partially water filled pit *("shallow"* water) or water full pit *("submerged"* vessel). When the pit is partially filled, the energetics are a function of the water level, depending on the initial corium jet diameter. For the configuration with a low level of water (a few tenth centimeters- "shallow" water), recent tests performed with simulant materials have shown possibly spontaneous and energetics steam explosion (Grishchenko et al. 2013) with conversion ratios approaching that of other simulant materials. These recent tests have been considered in the literature as *"stratified"* SE, and generated an interest in the subject. For the submerged vessel configuration, there are few experimental data (Na et al. 2016). Given that experimental results are very limited for both of these reactor cases ("shallow" water or "submerged" vessel), new experimental data would be necessary to establish a criterion regarding SE energetics or triggering for these configurations.

- *The containment pressure* (range from 1 to 4 bars): the impact of low ambient pressure is expected to be moderate on the energetics.

Table 4: Main phenomena parameters having an impact on ex-vessel steam explosion energetics and status of knowledge

Phenomenon	Ranking[a]	Status of knowledge[b]
Jet fragmentation in gas space	L	S
Jet fragmentation in water	H	M
Melt droplet characteristics in premixing (mean size, size distribution, nature of the corium: metallic, sub-stoechiometric, oxidic)	H	M
Heat transfers	H	S
Melt solidification	H	M
Melt oxidation	H	I
Voiding in premixing	H	M
Triggering	H	M
Fine fragmentation in explosion	H	M
Stratified steam explosion	H	I
Mechanisms of pressurization	H	M

For the phenomena involved in SE, some comments can be added:

- ***The coarse fragmentation and the resulting drop size*** is of high importance, since it affects heat transfers and fine fragmentation rate during explosion. The knowledge on oxidic corium with small temperature interval of solidification is relatively well known, but much less is known for metallic, sub-stoichiometric and large temperature interval of solidification of corium mixtures. Concerning the jet fragmentation, there is a limited knowledge for the lateral vessel failure, for example, in case of "focusing effect" and also for the pressurised jet.

- ***The heat transfer*** characteristics are obviously of utmost importance in steam explosion for steam production and melt solidification. Knowledge is estimated as relatively satisfactory, although some aspects related to heat transfer around drops and fragments still need some improvements.

- ***The solidification and oxidation phenomena*** are both expected to be important and there is knowledge gaps remain. Both are strongly dependent on the corium composition and conditions of cooling and steam removal at the surface of the droplet. Considering metallic, sub-stoichiometric (O/M<2.00) or stoichiometric corium with a large solidification temperature interval, up to now, very few experimental data exist with no parametric laws.

- ***The void in premixing by boiling and hydrogen production*** was considered to be of high importance for mitigation of steam explosion. For hydrogen production, too few experimental data exist to propose a model and evaluate the impact on SE energetics.

- ***The triggering from thermal fragmentation***. The triggering mechanisms for melt-bottom contact and melt spreading need clarification and characterization. Predicting triggering in a deterministic way seems out of reach. A statistical approach seems more adequate for the inherently chaotic phenomena.

- ***The fine fragmentation*** in the process of the explosion is of utmost importance. Knowledge is globally limited, particularly considering the effect of partial solidification and possible supercooling for the corium.

- ***The mechanisms of pressurisation*** (micro-interaction versus direct boiling), despite recent clarifications, further improvements are needed in understanding and convergence of views.

II.4   EVSE containment building consequences

The methodology to estimate the consequences of ex-vessel steam explosions for LWRs and BWRs and the maximum acceptable values on the containment is different according to country specific regulatory requirements. It can be based on probabilistic or deterministic approaches, which are not mutually exclusive.

Today, both one-dimensional and multidimensional FCI codes can be used in the deterministic approach. For probabilistic studies, fast running simplified tools and surrogate models have been developed based on the results of the full models of the FCI codes (Grishchenko et al., 2016). The surrogate models allow for extensive sampling to address aleatory uncertainty from a large number of possible severe accident scenarios and epistemic uncertainty in the modeling due to various closures. One important point concerning the nature of triggering in the reactor pit is that it can be assumed to be stochastic under severe accident conditions. Several methods have been proposed. No physical criterion or empirical law has been justified to determine the spatial and time localization of the triggering for ex-vessel configurations. Some criteria have been used (e.g. melt bottom-contact), but they are based on a

heuristic approach and expert judgment. It is consequently difficult to establish general bounding case for ex-vessel steam explosion. Alternatively a statistical treatment has been proposed for the effect of the triggering time (Grishchenko et al., 2015), although it requires significant computational efforts for the estimation of explosion loads with different triggering time for the same melt release conditions. The method provides a comprehensive treatment and can be used for both the bounding loads and more detailed data on distribution of the impulses.

The mechanical consequences depend on the reactor pit design (PWRs, BWRs). Even among the LWR reactor family, important differences on the design of the reactor pit and primary circuit can significantly affect the mechanical response to a given load. The main risks, which are primarily plant specific are: damage of internal concrete structure and potential reactor vessel detachment under the effect of high pressure in the cavity, with a risk of missile formation which can strike the containment wall. In addition, with the displacement of internal structures, the floors can strike the containment wall, and the heavy materials (such as steam generators) can be destabilized, leading to failure of some containment penetrations with a risk of containment radiological by-pass. It is also strongly dependent on the scenario leading to ex-vessel steam explosion, especially the break location of the reactor vessel, the melt jet velocity, the water level in the pit and the corium melt composition. The current knowledge is limited for some important phenomena (e.g. oxidation) and the capabilities of the tools to handle realistic 3D situations.

It is now possible to propose two classes for consequences of ex-vessel SE:

- *Direct consequences:* mechanical constrain on the containment concrete, on the access doors and on the reactor vessel.

- *Indirect consequences:* motions of elements of the reactor circuit, that can affect leak tightness of the containment (e.g. containment penetrations for the main steam line pipes) or equipment responsible for long term cooling of the debris.

  - *For the direct consequences*, the main risk is the failure of concrete containment or access doors though imposed mechanical loads, possibly by the displacement of walls due to the impacts. The question is then the assessment of the fragility of concrete structures against the steam explosion loads, noting that steel rebars are present in the concrete structures. For preliminary evaluations, it is common to make reference to the steam explosion impulse of the load, although some studies have shown that it cannot be the only factor since the maximum pressure also has an impact. The loss of integrity of the reactor pit for PWR's should occur after *several tens to several ten of kPa.s of impulse*. For large pit BWR's, the loss of integrity of the reactor pit might be obtained with smaller loads due to thinner walls. Nevertheless, the loads themselves might be smaller in larger pits due to the venting effect. The pressure wave following the inverse square law of the pit's radius, the distance at which the explosion occurs from the wall will be one of the most important parameter regarding the reactor pit integrity. For some BWR designs, access hatches have doors that can withstand about 6-10 kPa.s and can be reinforced up to ~20-50 kPa.s.

  - *For the indirect consequences*, the main risk concerns the displacement of heavy elements present in the containment as: Reactor Pressure Vessel (RPV), concrete floors, cross pipes, Steam Generator. Deformation/rupture and entrainment of Reactor Coolant System (RCS) piping enters in this category. In the range of 100-200 kPa.s, it can be considered that motion start to be significant, especially for the vessel, and can cause, for certain designs, failure of the containment integrity in the penetrations.

In this approach, it is also important to take into account the mid-term management of a severe accident of the nuclear reactor: if the containment failure does not come right after the explosion, steam explosion can cause irreversible damages on the operability of equipment needed for accident management (pipes, pumps... needed for continuous water supply, containment cooling and/or venting) that can render the situation unrecoverable for Severe Accident long term management.

In the next paragraphs, some national examples of EVSE consequences evaluation are presented.

In the United States, an early study (Theofanous et al., 1994) provided an assessment of ex-vessel SE energetics for operating PWRs and BWRs. The reference plants in this study included all BWR and PWR containment designs. The study concluded the steam explosion induced containment failure is not a concern with the exception of BWR Mark III containment, where structural failure could not be excluded. An assessment of ex-vessel steam explosions in the AP600 advanced PWR was performed (Esmaili et al. 1996) in mid-1990s. The study looked at three different scenarios for the potential of steam explosion (unsubmerged RPV, partially submerged RPV, and fully submerged RPV) and a large number of sensitivity parameters to calculate maximum impulse loading on the cavity wall. There was a large spread in the calculated impulse load ranging from as low as 15 kPa.s to as high as 600 kPa.s. An assessment of ex-vessel steam explosions in the AP1000 reactor was performed (Esmaili and Khatib-Rahbar, 2004) in 2004. This study was similar to the one for AP600 and resulted in impulse loads in the range between 10 kPa.s to 300 kPa.s.

In Switzerland, an assessment has been completed for BWR (Leibstadt) (IAEA, 1997) and it has shown the possibility of destroying the pedestal by an impulse of 150 kPa.s, with consequences on the reactor containment building.

In France, evaluations where made separately by IRSN (Cenerino et al., 2013) and EDF (Markovic and Idoux, 2011), with different conclusions. IRSN produced a PSA-2 study on PWR's reactors with linked calculations of loads and mechanical impact with simplified 2D calculations. The considered risk was the displacement of floors and impact on the containment. It was concluded that the risk of failure was conditionally important and needed to be better assessed with, in particular, more accurate mechanistic evaluations. This was done latter on by IRSN using 3-dimensional complex mechanical models. Although some conservatisms of the previous IRSN study could be pointed out, the risk of containment failure could not be ruled out, particularly considering the uncertainty of the load evaluations and their lack of identified conservatism. For the EPR reactor, design provisions have been implemented in the reactor pit before the vessel failure to prevent any energetic corium/water interaction.

In Japan, researches on the containment failure probability due to ex-vessel steam explosions were performed for Japanese BWR and PWR model plants based on the model validated by FARO and KROTOS 22. In these researches, a steam explosion was evaluated assuming the external trigger at the time of maximum premixed mass in the pedestal area (or in the suppression pool of) a BWR model and in the reactor cavity of a PWR model plant. After SERENA-2, the evaluation model was validated based on KROTOS and TROI and major uncertainty parameters were identified in pre-mixing and fragmentation phases (Hotta et al. forthcoming). Output of this study has been applied in a sample evaluation of actual plants assuming cavity injection.

These mechanical evaluations show clearly that the values of impulses and pressure of steam explosion are too close in some cases to mechanical or motion limits for PWRs and BWRs (Gen 2 and 3), having consequences on reactor building containment integrity. Thus, it is necessary to have a better knowledge and modeling of ex-vessel steam explosion. An improved understanding will decrease the uncertainties on consequences of EVSE on reactor containment and enable development of mitigation measures for EVSE. Another point concerns the medium and long term management of the severe accident in case of SE even if the containment is still safe; this point has never been evaluated.

II.5   EVSE national approach after Fukushima

Over the last decade, both experimental and analytical research on the topic of SE in nuclear reactors has greatly improved our understanding of the subject. Notwithstanding that, there is a strong desire in nuclear safety organisations of some OECD countries to achieve a better precision in their analytical tools for an assessment of SE risk. This desire is predicated, in part, by an anticipated need to make a more robust safety case for the operating reactors, prompted by the accidents at Fukushima.

The safety analysis practice in most OECD countries included an assessment of SE risk, albeit, with varying levels of complexities and sophistication in assessment methodologies. With the conclusion of the OECD SERENA program (Phases 1 and 2), the nuclear safety researchers in the OECD countries felt that improvement in predictive capabilities of various analytical tools is needed to gain more confidence in calculated steam explosion risks and safety margins.

The members of the writing group discussed the subject at length during the development of the EVSE status paper, and determined that the regulatory bodies of the participating countries be polled for regulatory requirements concerning steam explosion.

The main observations from the survey responses are:

the regulatory requirements vary between countries;

most countries, if not all, consider EVSE still an open issue, due to remaining uncertainties in large part;

the detailed methodologies for SE risk assessment vary between countries in complexities and sophistication though most of them have same basic elements – a deterministic assessment of steam explosion loads (impulse and/or pressure) with a structural mechanics assessment of containment fragility and a probabilistic framework to evaluate the risk of containment failure associated with a steam explosion; and finally;

the needs for and scope of continued research on the subject vary between the countries.

The writing group of the EVSE status paper note that the events at Fukushima, with the current knowledge and interpretation of the accidents, have no particular bearings on the likelihood of SE or its severity. At the same time, the writing group recognizes that changes in national regulatory requirements post-Fukushima are dictated by considerations other than the above. Further, the experts group recognizes that in light of new regulatory requirements in some countries and for specific reactor designs, there is a need to have an improved precision in steam explosion risk assessment, despite SE code improvements made during SERENA-2.

Concerning the main experimental SE programs developed after Fukushima severe Accidents, we can quote the following national programs:

In France, in the frame of the French national program RSNR-Post-Fukushima supported by the French Research Ministry, a new program called ICE (Piluso, 2015) (Water Corium Interaction, RSNR-Post-Fukushima action), led by IRSN has started in 2014 and will be finished in 2019: this program is devoted to the studies of new FCI/SE configurations (main scientific topics: oxidation, solidification, jet fragmentation), closer to real configuration with an experimental part in CEA/KROTOS facility with prototypical corium. MC3D FCI code (IRSN) is being developed to complement the experiments and will capitalize the knowledge gained on FCI/SE.

In Korea, The Korea Atomic Energy Research Institute (KAERI) has conducted the TROI) program for a study on fuel-coolant interaction (FCI) since 2001: more than 70 experiments using several prototypic materials have been performed. KAERI, as a part of national program finished in February 2017, has carried out tests to simulate a fuel coolant interaction using ZrO_2 and corium under the conditions of the reactor vessel submerged in water (Na, et al., 2016)[Error! Bookmark not defined.].

In Sweden, the recent experiments undertaken at KTH using simulant materials, called PULiMS (Pouring and Under-water Liquid Melt Spreading) and SES (Steam Explosion in Stratified configuration) investigating the effects of melt spreading on the pit floor lead to regular spontaneous explosions at high melt superheat and moderate water sub-cooling (5-25 K) (Grishchenko, 2013). This configuration is an important issue because in some accident scenarios (large jet, small water height), the melt should effectively spread as a liquid on the pit floor and this melt might participate in the explosion, or trigger it. In the frame of European program "Severe Accident Facilities for European Safety Targets" (SAFEST) (Euratom, 2014-2018), complementary experiments are planned to have a better knowledge of this energetic steam explosion configuration.

Concerning FCI/Steam explosion activities in 2016, it is to be noted that in Europe, the SARNET network is still continuing in the frame of NUGENIA program, providing in particular exchanges among the experts and promoting access to some of the facilities in Europe in the frame of the European SAFEST program (2014-2018).

II.6   EVSE improvements and needs for reactor application

In the frame of OECD/SERENA-Phase 2 program, valuable improvements to the understanding and modeling for Ex-Vessel SE have been obtained on the important stages of the premixing/fragmentation and explosion phenomena. Simplified generic configurations for BWRs and PWRs reactor case (composition of corium, vertical jet, 2D configurations) have been studied. Consistent progress on the FCI/SE phenomena knowledge and robustness in modelling, have been obtained as described in the previous sections of the status paper.

Nevertheless, these EVSE generic configurations are representative of a limited number of scenarios whereas there are other configurations like e.g. for the vessel failures (central hole or lateral failure). The exact status on EVSE has been recently provided in a common paper (Meignen et al., 2014), stressing the important needs and phenomena to be studied in priority for FCI/SE for application to the reactor case. Together with the above established tables, it is possible to highlight the major requirements to fulfill the needs for more realistic reactor applications.

Some important topics are now summarized here:

__Jet orientation and fragmentation__: up to now, only strictly vertical, gravity driven jets have been investigated in experiments and for FCI/SE reactor case studies (PWR or BWR, Gen2). Nevertheless, for reactor applications, other cases have also to be considered like lateral jet fragmentation. Indeed, in case of ex-vessel cooling strategy by natural convection (for example, some Gen 3 reactors), a probable way of vessel failure is a lateral breakthrough of the thin upper metallic layer caused by the *"focusing effect"*. Another configuration to be also considered concern the difference of pressure between the reactor pit and the vessel: even if it is planned to have depressurization procedure in case of severe accident, it is necessary to evaluate conditions with low remaining pressure difference between the vessel and the reactor pit (a few bars, ejection speeds of around 10 m/s). So complementary experiments with prototypical corium have to be performed respecting these two configurations to validate and improve the modeling currently used in FCI codes.

__Melt solidification__: this topic was a major issue of the OECD/SERENA-Phase 2 program, through the so-called "material effect". It has been found difficult to provide precise models combining solidification and fragmentation during the premixing. Concerning the impact of solidification on fine fragmentation, no models exist and the impact is based on criteria discriminant drops that should fragment and those which are solid. The models and criterions are however facing a difficulty with a lack of experimental data, with variations of corium compositions. Three corium composition families can be identified representative of reactor cases: metallic, sub-stoichiometric (O/M<2.00) and oxidic with large interval of solidification. Based on explosion simulation results one can estimate the combined effect of the melt droplets solidification and the void in mixture on the reduction of the explosion strength in the experiments. To

facilitate the extraction of the influence of melt solidification on the explosion strength from the combined influence of solidification and void, it would be advantageous to perform experiments with different solidification levels respecting similar premixing conditions.

__Oxidation__: past experimental programs ZREX/ZRSS (NEA, 1994) have shown the importance of oxidation phenomena in the case of a SE with a pure metallic melt, a case representative of some particular situations for the vessel failure due to the focusing effect. The OECD/SERENA-Phase 2 program has shown the importance of oxidation process for sub-stoichiometric or stoichiometric melts with other consequences on SE (mitigation by hydrogen release). One of the major issues regarding the modeling of oxidation in the current FCI codes is that it is still largely parametric due to an important lack of analytical experimental data and models related to the oxidation of a high temperature single drop or fragment. The ICE French program (Piluso et al., 2015) was built to answer partly to this point, but this program is covering a limited range of corium composition for reactor case application. Furthermore, integral tests with large masses of prototypical sub-stoichiometric and metallic alloys of corium (Zr, Fe, U) will be necessary to evaluate the impact of hydrogen and chemical energy releases at large scale, and to validate oxidation models in FCI codes.

__Melt fine fragmentation__: the modelling of fine fragmentation in the FCI codes is currently based either on isothermal single drop experiments, either on experiments using simulant drops at relatively low temperatures (1 000-1 500 °C) in a water flow (DROPS, SIGMA, DROPSG). The hydrodynamic fragmentation of a prototypical drop in reactor conditions was never studied experimentally. Thus, possible effects which are specific of reactor conditions, such as very high temperatures, partial solidification, and very high pressures are currently not well known. These effects are therefore either ignored by the codes, either taken into account but with no possibility of validation. Therefore, in order to improve the current modelling of fine fragmentation in FCI codes, it would be necessary to study experimentally the hydrodynamic fragmentation of a single prototypical drop in reactor conditions.

__Voiding in premixing__: voiding in premixing is the result of two physical phenomena: vapor (and possibly hydrogen) production due to heat and mass transfer around the melt drops, and the two-phase flow description of the coolant (heat and mass transfer around bubbles, bubble dynamics, etc.). Heat and mass transfer around melt drops is relatively well known, but some points require improvements, such as for example the heat fluxes distribution between vapor production and liquid heating. Improvements may be obtained by developing more detailed descriptions (modeling or direct numerical simulation, DNS), and by additional experimental data (for single spheres or cylinders with diameters comparable to the drops in premixing, there is no experimental data on convective film boiling for temperatures above 1 000°C). Regarding the two-phase flow description of the coolant, the physics are well known, but the main question is which level of detail is actually necessary in FCI codes in order to be able to correctly predict voiding in large scale experiments and reactor cases. At the end, in order to validate improvements in FCI codes related to voiding, it will be very important to have large scale experiments with instrumentation allowing measurement of spatial distribution of void with enough precision for FCI code validation.

__Mechanism of pressurisation__: some important experimental data related to this topic has been obtained recently in the TREPAM (Berthoud, 2009) experiment, which simulated heat transfers between fragments and coolant in conditions close to an explosion in reactor conditions. Some further improvements are however needed in order to better evaluate the void produced by a fragment, and thus the resulting pressurization. Additional analytical experiments at small scale (one droplet) could be performed and also some DNS calculation, which could be a promising tool for this purpose.

__Triggering__: it is considered at the moment as a stochastic event. This conservative assumption is not satisfactory because it has a huge impact on SE energetics evaluation. Some experts estimated that self-triggering through thermal fragmentation with small perturbations has a low probability of occurrence for corium. Melt-bottom contact is for some other experts a preferential criterion. Recently, at KTH (Kudinov, 2014), systematic spontaneous and energetic steam explosions have been observed during the spreading phases of simulant materials in a limited height of water, while no explosion ever occurred in the past

DEFOR tests done with exactly the same conditions except a much larger water height (complete jet breakup before reaching the bottom), confirming thus the tendency for self-triggering during contact with the bottom. This tendency needs however to be confirmed for prototypical corium. This configuration for SE has to be taken into account to understand the phenomenology and improve, at least, criteria. Nevertheless it is necessary first to understand clearly the triggering phenomena in KTH experiments before any extrapolation to real materials and reactor case conditions.

Melt "stratification": it has long been considered that the stratified melt could not participate actively in a steam explosion. However, this assertion was based on idealized situations with simulant materials. Considering the potential of the stratified melt to increase the load and recent tests performed at KTH (Kudinov, 2014) with simulant materials (2014-2016), confirmations of a weak impact for more realistic conditions is necessary. Experimental corium spreading on concrete under water should be considered as a possible configuration for EVSE. Nevertheless it is necessary first to understand and model the SE phenomena involved in KTH experiments before any extrapolation to real materials and reactor case conditions.

These highlighted topics show that there are still needs in the future for both small-scale analytical experiments in order to enhance the models currently used in FCI codes, and large scale experiments for a more reliable qualification and validation of the codes and also to investigate new EVSE configurations (lateral or metallic jets, stratified configurations for example).

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