Past and Future R&D at IRSN on Corium Progression and Related Mitigation Strategies in a Severe Accident

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ABSTRACT

Reactor core degradation and in-vessel and ex-vessel corium behavior have been major research topics for the last three decades to which IRSN strongly contributed by the coordination of, or by the contribution to, large R&D programs and through the development and validation of the severe accident ASTEC code. In the last years, the balance of research efforts tipped on analyses of pro and cons and assessments of mitigation measures. The outcomes of risk significance analysis (including fuel-coolant interaction (FCI), hydrogen combustion and molten core concrete interaction (MCCI) risks) performed in France as well as the retained orientations for corium behavior research are described. The focus lies nowadays in (1) in-vessel melt retention (IVMR) strategies for future reactor concepts but also related to the need to establish the reliability of such strategies when implemented in existing reactors (2) in-containment corium cooling for existing reactors.

The paper summarizes main achievements and remaining issues related to understanding and modelling of:

- reflooding of a degraded core where, despite substantial knowledge gained through R&D programs (e.g. DEBRIS, PRELUDE, PEARL) additional efforts are required to establish the efficiency of such a measure and the associated risks notably for largely degraded cores;
- corium behavior in the Reactor Pressure Vessel (RPV) lower head where, despite the MASCA program results, efforts remain necessary to predict RPV thermal loadings resulting from corium layers evolution and RPV resilience with and without IVMR measures (internal and/or external cooling);
- fuel-coolant interaction for which, despite OECD SERENA program results, the knowledge is not sufficient to assess with confidence the induced risk of containment failure;
- MCCI where the knowledge on corium cooling in the containment by top and/or bottom water flooding is insufficient to conclude on the efficiency of such measures. Of particular interest for top flooding are the water ingress and corium eruption processes. Specifically for top flooding respective impacts of water ingress and corium eruption processes remain to be quantified in reactor conditions.

In support to these activities, substantial efforts are also being conducted at IRSN to constantly improve and validate nuclear material property databases which are key tools for corium behavior analysis.

The paper describes on-going and future research programs performed at IRSN or internationally with IRSN coordination or participation to tackle the identified remaining issues (e.g. PEARL and follow-up, CORDEB, H2020 IVMR, ICE, CCI, etc.) and summarizes foreseen progress in modeling for SA codes, in risk analysis and in severe accident management.

KEYWORDS
In-vessel melt retention, fuel-coolant interaction, molten core concrete interaction, corium cooling

1. INTRODUCTION

Following the Fukushima accident, most nuclear countries have launched reassessment of the safety margins of their NPPs towards extreme events and severe situations. This has generally led them to implement additional safety measures to improve the robustness of the defense in depth approach. Nevertheless in the field of Severe Accidents (SA), in spite of large R&D efforts for more than three decades, some major issues remain incompletely resolved due to phenomenological uncertainties. Some of them may still lead, for existing plants, to modifications and/or SA Management Guidelines (SAMG) revisions, notably to reduce the risks of containment failure and of uncontrolled radioactivity release at short and long terms, should a significant core melt occur. Ensuring the containment leak-tightness and integrity against SA phenomena at short and long terms may require implementing robust mitigation features. For the future reactors, the safety enhancements can more easily be applied at their design stage.

We discuss hereafter the assessment of SA mitigation strategies for French NPPs and R&D open issues related to corium progression during a SA as it may challenge the reactor pressure vessel (RPV) integrity and then the containment of radionuclides. Despite remaining uncertainties on knowledge of corium progression and on assessment of the risk it represents for the containment, more robust mitigation strategies are currently investigated for operating plants in France to comply with safety requirements associated to the containment (e.g., elimination of the risk of containment failure at short term and of uncontrolled radioactive releases at long term). Further, safety related R&D on corium is pursued to reach a consensus regarding remaining uncertainties and to support the demonstration of the efficiency of mitigation strategies implemented at the operating plants and designed for the future plants.

2. ASSESSMENT OF SEVERE ACCIDENT MITIGATION STRATEGIES IN GENERATION II FRENCH NPPS

2.1. Introduction

The French electrical utility Electricité de France (EDF) is operating a fleet of 58 standardized Pressurized Water Reactors (PWRs), composed of 3 series of 900, 1 300 and 1 450 MWe NPPs. In France, reactor safety upgrades follow assessments performed in Periodic Safety Reviews (PSRs) conducted every ten years[1] for each reactor series [1]. These PSRs have been completed by Safety Reviews for the NPPs Long Term Operation (LTO) plan [2] proposed by EDF where French regulatory framework does not fix the duration of operation of plants and complementary safety evaluations (CSE) conducted after the Fukushima's accidents [3].

[1] Upgrades can be implemented on reactors during their ten-yearly outage. The 3rd PSR for the 1300 MWe PWRs (20 reactors) is currently under progress and their 3rd ten-yearly outage is planed from 2015 to 2021. The 3rd ten-yearly outage for the 900 MWe PWRs (34 reactors) is on-going and their 4th PSR starts in 2014 as their 4th ten-yearly outage is planed from 2019 to 2029.

The identification of necessary reactor safety upgrades and the analysis of the related expected benefice are based on both deterministic and probabilistic approaches, using notably for the later Level-2 Probabilistic Safety Analysis (L2 PSA) [4].

Severe accidents were not considered at the design stage of the generation II French PWRs. Nevertheless, all operating plants already include equipments and measures for the SA management (e.g., Passive Autocatalytic Hydrogen Recombiners (PARs) for hydrogen combustion risk reduction, reinforced Reactor Coolant System (RCS) depressurisation components and procedures notably for Direct Containment Heating (DCH) risk reduction, Emergency Filtered Containment Venting System (EFCVS) for reduction of containment failure risk by slow pressurization, severe accident instrumentation to guide the conduct during the accident (e.g., corium arrival detection in the reactor pit)) as a result of previous PSRs [5].

2.2. Risk Analyses Results

L2 PSA allows identifying the containment failure modes that contribute the most to the global risk, notably by examining the basic risk metric based on the product of the estimated frequency of a given containment failure mode by the corresponding amplitude of radiological consequences[2] [1,6]. Using such an approach, it was determined, notably in recent analyses conducted by EDF and independently by IRSN for 1 300 MWe French reactors [1] that the global risk for these reactors can be reduced significantly through material or procedures improvements, notably that arising from some importantly contributing containment failure modes (specific attention is paid to accidents with the highest frequencies and the most severe impact; these are provided in Figure 1), such as Induced Steam Generator Tube Rupture (I-SGTR), containment isolation failure in case of SBO, containment bypasses[3], containment failure after Direct Containment Heating (DCH). These improvements concern mostly the implementation of additional electric supplies to tackle with SBO situations and increasing the robustness of equipment and procedures to timely depressurize the RCS.

If such improvements are set for (and they are in the process of being implemented in all French PWRs), results of the study also indicate that the residual risk is then essentially due to the following containment failure modes:

- hydrogen combustion (in the containment during the in-vessel core degradation phase or during the MCCI phase and in the annulus[4]);
- heterogeneous dilution[5];
- ex-vessel steam explosion;
- concrete containment basemat melt-through by MCCI;
- internal and external hazards.

A similar study conducted earlier for 900 MWe PWRs [6], led to similar conclusions than with 1 300 MWe PWRs with less importance given to hydrogen combustion scenarios due to the difference in containment type and robustness to high pressure and temperature loadings. It should be emphasized that both studies do not consider the radiological consequences due to ground contamination in the case of

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[2] Radiological consequences can be expressed using various criteria, e.g. amplitude of releases, effective dose at a given distance from the plant and for a given time lapse after the accident.
[3] By-pass due an initial SGTR, a LOCA on a pipe outside the containment connected to the RCS, a failure of the equipment hatch during shut-down states.
[4] 1 300 MWe reactors have double-walled containment but no steel liner. The annulus is the space between the two walls. A hydrogen combustion can take place in the annulus during MCCI.
[5] Heterogeneous dilutions prediction and consequences are subject to high uncertainties, it is conservatively assumed in the performed study that they lead to an early containment failure and large radioactive releases.

concrete containment basemat melt-through by MCCI. Doing so would attribute even more importance to these scenarios in the risk analysis.

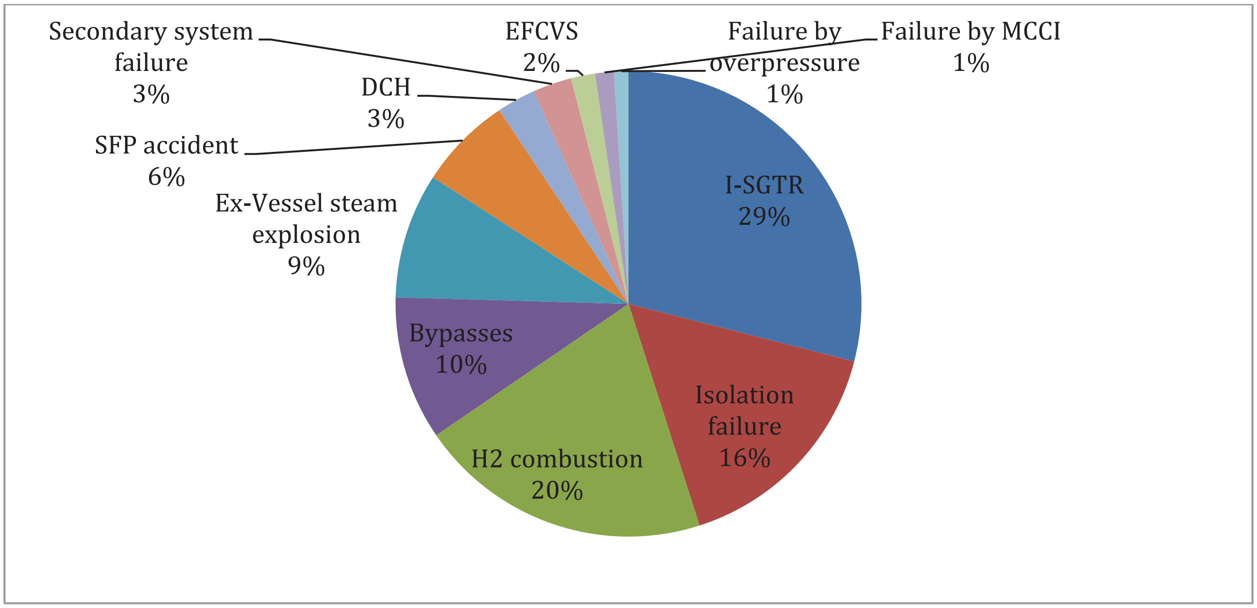

Figure 1. Estimated Contributions to the Global Risk of Containment Failure Modes (Ranking Associated to Atmospheric Releases) for a French 1 300 MWe PWR [1]

The residual risk may be more accurately assessed and eventually reduced by improving the knowledge on these processes, notably on the containment failure risk they induce, and by implementing mitigation strategies. IRSN is conducting studies and research to progress in both directions. Concerning internal and external hazards, L2 PSA development must progress to characterize the associated risks.

Studies and research associated to hydrogen combustion, heterogeneous dilution or hazards impact on SA risks will not be discussed here. In the following, the focus is on risk reduction of ex-vessel steam explosion and concrete basemat erosion by MCCI and the design of corresponding mitigation strategies. Such strategies are obviously strongly dependent on water supplies management during the accident.

2.3. Management of the Ex-vessel Steam Explosion Risk

For French generation II PWRs, the design of an optimal management of water supplies during a SA was extensively discussed considering that:

- the risk of RPV failure may be lowered by water injection inside. However, the probability to recover some previously lost water supply sufficiently soon after the start of fuel degradation, is low. Also, achieving melt retention and cooling in the RPV by water injection alone after the onset of melt formation is not guaranteed. Further, water injection will yield oxidation of remaining non-oxidized material and hydrogen production thus increasing the risk of hydrogen combustion in the containment;
- the risk of RPV failure may also be lowered by cooling the RPV externally by reactor pit flooding (in French PWRs, the reactor pit can be partly or totally filled with water after activation of the inner containment spray system). However, achieving melt retention and cooling in the RPV by external cooling is also not guaranteed. Indeed, transient 3D corium configurations in the RPV lower head can induce high thermal loads on the RPV wall, by transient focusing effects, leading to a fast failure [7]. In case of failure with external cooling, the breach for French PWRs is

considered to be most plausibly local and located on the lateral wall of the RPV [7]. Melt-coolant interaction at the breach then increases the risk of ex-vessel steam explosion.

Even if the risk of RPV failure could be reduced significantly by implementing both in-vessel (if available) and ex-vessel flooding, RPV failure for transient 3D corium configurations cannot be totally excluded. Then, implementing a strategy with external cooling would result in a steam explosion risk with possibly high consequences and affect also the corium spreading over the whole reactor pit basemat surface with possibly higher local thermal loads.

With the present knowledge on steam explosion due to melt-coolant interaction and the uncertainties on the assessment of the resulting consequences on important safety systems and components for the plant and on containment leak-tightness, steam explosion yielding significant damages to systems, components and the containment and eventually large early radioactive releases cannot be excluded. Thus, for French generation II PWRs, a strategy with a dry reactor pit is presently the preferred option. Such a strategy would eliminate the risk of large early releases due to steam explosion and, as shall be seen in paragraph 2.4, may also ease the management of ex-vessel corium cooling.

Concerning in-vessel water injection, if water supplies can be recovered during the accident, such a measure is considered to be beneficial for melt cooling inside the RPV. To be efficient and innocuous, it should however be operated in a timely and controlled manner and adequate means should be implemented to mitigate the resulting H_2 risk, such as PARs. Note that water supplies management has a strong impact on the hydrogen combustion risk (e.g., hydrogen production due to in and ex-vessel corium cooling, reduction in containment steam content due to spray activation).

To summarize, IRSN considers that an in-vessel retention (IVR) strategy for French generation II PWRs based both on in-vessel cooling by water injection and ex-vessel cooling by reactor pit flooding (flooding can be made voluntarily or not and may take few hours) is too uncertain and presents, with the actual knowledge of fuel-coolant interaction, unacceptable risks. It is too uncertain since (1) the availability of both a water injection into the vessel and a fully flooded reactor pit are not guaranteed (2) even if both cooling modes are made available, the RPV lower head failure may be inevitable with transient focusing effects and high relocalised corium mass. It presents “unacceptable” risks as the melt-coolant interaction at RPV failure may result in steam explosion and large early radioactive releases[6].

Concerning the steam explosion risks, recent experimental observations made in KTH experiments (PULIMS ans SES) [8] suggest that configurations with a shallow layer of water in the reactor pit may be conducive of energetic steam explosion if the melt jet at RPV failure results in a large layer of melt below the water layer (these configurations are called “stratified melt-water configurations”) with a large thermal energy potentially available for an explosion. Even if at this stage, the mechanism explaining the self-triggering of the energetic explosions in these experiments is not understood and if such explosions occurrence for actual reactor melts and configurations is uncertain, the risk of energetic explosions with partially flooded reactor pit cannot be excluded. Again, a dry reactor pit would eliminate such a risk, at least for the first pour if an early ex-vessel flooding strategy is retained as accident management strategy.

For the French generation II PWRs, the reactor pit which is a relatively “closed pit” may be kept dry up to the RPV failure by implementing rather simple structure modifications. A strategy with a dry reactor pit should be complemented by a strategy to cool the corium once it has spread on the reactor pit concrete basemat as discussed in the next paragraph.

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[6] As mentioned earlier, the French Safety Authority has prescribed for generation II PWRs the practical "elimination" of accidents leading to large early radioactive releases.

2.4. Management of the concrete basemat melt-through risk

From 2009, the French Safety Authority (ASN) with the support of IRSN started the evaluation of the strategy proposed by EDF for the NPPs Lifetime Extension (PLE) from 40 to 60 years in the specific context of the construction on the Flamanville site of a first EPR of Generation III (or III+), close to two Gen II 900 MWe PWRS operating since the mid-eighties. In line with the WENRA position, ASN required EDF to refer to the safety objectives of the Generation III reactors for all the safety studies made for the Generation II PLE. Concerning SA, these safety objectives for Generation III reactors including the EPR should lead to “only very limited protection measures in area and time for the public”. The proposition made by EDF does not only focus on the ageing management of Safety Systems and Components (SSC) but also includes a safety enhancement program in which EDF intends to examine the possibility to implement a so-called “hardened safety core” of measures that includes the improvement of the EFCVS efficiency, the improvement of the containment decay heat removal without opening the EFCVS and measures to avoid corium basemat melt-through by MCCI in case of RPV failure. Post-Fukushima complementary safety evaluations (CSE) in 2012 led to the ASN resolution 2012-DC-0283 in which ASN required EDF to investigate for SA the possibility to include the two last above mentioned improvements into the so-called hardened safety core (*cf*. the ECS-ND1 and ECS ND16 of resolution 2014-DC-0403). Solutions proposed by EDF will be evaluated in the CSE context, plant by plant, to account for specific external hazards. French regulatory framework does not fix any duration of operation for nuclear facilities and propositions made by EDF for PLE will also be evaluated plant by plant.

Containment designs not only differ in between the different concepts of reactors but in the different implementations of a same concept of reactor. In countries where unique safety and regulatory requirements exist, these differences arose from specific seismic risks or specific geologic configuration that led to constraints on the foundations, from flooding risk prevention, from the type of concrete components (mortar, cement, aggregates) available close to the plant location. These differences must be accounted when evaluating basemat melt-through risks. As already mentioned NPP operated by EDF in France belong to three different series (900 MWe, 1 300 MWe, 1 450 MWe). Nevertheless significant differences exist in their implementation. Basemat thicknesses and basemat concretes depend on the plants site. Most of the 900 MWe PWRs have a basemat around 4.5 meters thick whereas the basemat thicknesses of 1 300 MWe and 1 450 MWe range in between 3.1 meters and 3.5 meters. 900 MWe CP0 plants located near Fessenheim and Bugey were designed with a thinner basemat of 1,5 meters and 2.2 meters respectively. Most of the plant basemat are made using siliceous-rich concrete whereas the plants in 3 locations have been built using LCS concrete. Additionally to specific external hazard, these differences make MCCI basemat penetration risk necessarily evaluated plant by plant, site by site.

EDF has been requested by the ASN to significantly reduce the risk associated to the basemat melt-through before being granted a 10 years lifetime extension of the Fessenheim plant. EDF proposed very significant modifications that have been evaluated by the ASN with the support of IRSN and considered to be satisfactory. Modifications implemented are a thickening of the basemat of both the reactor cavity and of an adjacent room with a 0,5 meter thick layer of self-levelling LCS concrete. The reactor cavity has been connected to this adjacent room using a transfer channel including a fusible plug. In this adjacent room, concrete vertical walls have been built to prevent water to fill the surface area devoted to corium spreading. These modifications allow increasing significantly the basemat melt-through delay. Contrary to the initially designed configuration and with the current state of knowledge on MCCI, no accident scenario leading to a basemat melt-through in less than 24 hours after the accident start – duration needed in France to implement first population protection measures – has been found. For these modifications to be fully efficient, the corium should spread on the whole surface areas before cooling water is injected on top of the spread corium which illustrates the tight link between MCCI mitigation and water management strategies.

In addition to the elimination of the steam explosion risk at RPV failure (cf. paragraphe 2.3) and related increased probabilities to keep functional the structure, systems and components (SSCs) needed for severe accident management after RPV failure (e.g. water injection systems), a reactor pit maintained dry till the failure would allow corium melt spreading on a larger area and then its cooling by water injection on top as soon as the water injection systems are recovered. This strategy is being currently assessed within the NPPs long term operation safety evaluation process. Related EDF proposals are expected for the 900 MWe series in the framework of their 4th PSR. First related technical discussions should take place in 2015.

Concerning containment basemat melt-through, some other impacts have to be considered, including especially the short to long term ground contamination risks as illustrated by the Fukushima situation.

3. ASSESSMENT OF SEVERE ACCIDENT MITIGATION STRATEGIES FOR FUTURE REACTORS

Severe accidents are considered at the design stage of generation III reactors. In particular, measures should be implemented to avoid containment basemat melt-through for these reactors. In this objective, two different approaches are generally implemented, one considering the ex-vessel corium cooling after vessel rupture (notably the case for the EPR reactor using corium cooling by spreading in a dedicated collecting system and by top flooding and bottom heat extraction with water) and the other considering the in-vessel melt retention (IVMR) (notably the case for the Westinghouse AP1000 and some VVER reactors using vessel cooling by external water circulation). IVMR strategies, if their robustness is demonstrated, i.e. if it is shown that the RPV integrity can be maintained for all conceivable SA scenarios, would best fulfill safety objectives by avoiding any corium transfer outside the vessel and reducing significantly the risk of containment failure and of radioactive releases to the environment.

Analyses conducted by IRSN using present knowledge on corium behavior in the vessel lower head during a SA, have shown that, if the RPV and the external cooling system designs are such that the heat extraction is optimized, IVMR strategies would be applicable for reactors with a power of up to 600 MWe. These analyses do not consider any recovery of water injection inside the RPV during the postulated accident. For reactors with a power above 600 MWe, with the actual level of knowledge, the robustness of IVMR strategies cannot be established for all conceivable situations and RPV failure risks are increasing with increasing reactor power. The increase in RPV failure risks increases the steam explosion risk due to corium interaction with the water from the external cooling system.

Thus, IVMR strategies for reactors with a power above 600 MWe, need to be consolidated through R&D efforts and/or technological evolutions, with a challenging safety demonstration due to the high number of conceivable in-vessel degradation scenarios and the complexity of related physical phenomena.

IRSN recently engaged in R&D programs to strengthen the knowledge for the assessment of the robustness of IVMR strategies which may be proposed in some existing reactors or for new concepts of reactors. These programs are dealing with the study of reflooding of degraded reactor cores (in particular for debris bed configurations to reduce the corium mass relocated in the lower head), of corium behavior in the vessel lower head (to determine the bounding conditions which may challenge the RPV integrity) and of external vessel cooling to determine maximal extractable heat fluxes.

Whatever the retained option – IVMR or Ex-Vessel Melt Retention (EVMR) –, stable coolant injection in the RPV and heat evacuation from the containment have to be set for. The associated risks in terms of containment failure, by pressurization or by dynamic solicitations have to be assessed.

4. EXPECTATIONS FROM RESEARCH

IRSN objective is to get the scientific bases to assess severe accident management, particularly the water supplies management, and mitigation. Fulfilling this objective requires the development and validation of models reducing existing uncertainties and improving the predictability of simulations. These models are implemented in the SA accident code ASTEC [9,10] with the exception of fuel-coolant interaction modelling which is implemented in the MC3D code [11,12].

IRSN has thus engaged a research program on four axes:
- degraded core reflooding and the capacity to cool down a debris bed in the RPV;
- configurations of corium pouring to the vessel bottom head and melt pool formation and evolution;
- mechanisms of the fuel-coolant interaction;
- mechanisms of corium cooling during MCCI.

All these research axes comprise experimental, analyses and modeling activities.

4.1. Degraded Core Reflooding

Concerning the first research axis, work mainly concerns:
- the modeling of conceivable debris bed configurations (porosity, exchange surfaces, permeability, …) considering knowledge of debris geometry characteristics [13];
- the modeling of thermal-hydraulics for these different configurations;
- the modeling of the oxidation of oxide-metals corium mixtures and of the non-condensable gas production;
- the performance of studies coupling all phenomena.

The models development and their qualification are notably based on tests performed in a small scale facilities at IRSN (PRELUDE) and elsewhere [14,15] which will be completed by larger scale tests which are currently being performed at IRSN in the PEARL facility [16].

Core reflooding is an important SAM measure in order to stop or at least delay the progression of a SA in a NPP. Depending on the on-going accident scenario and on the time at which water sources can be made available, two quite different geometrical core configurations have to be considered for the reflooding process (1) a rod-like geometry representative of a quasi-intact core (with only ballooned claddings) or of the early degradation phase (possibly exhibiting some candling of molten materials but somewhat limited, i.e. without any local flow blockages) (2) a debris bed geometry representative of the late degradation phase as seen in the Three Mile Island degraded core. To predict accurately coolant flows in such configurations, adequate models were developed at IRSN and have been implemented in the SA ASTEC code.

Concerning the improved reflooding model addressing rod-like geometries, details about the adopted physical and numerical modelling can be found in [17,18]. For debris bed reflooding, the particularity is the higher temperatures, the smaller hydraulic diameter and tortuosity so that the coolability is more difficult to achieve. The recently developed model takes into account the two-phase flow through the heated porous bed [18, 19]. It is based on the modified momentum balance equation and on the specific heat transfer laws between the debris and the fluid.

The new model for debris bed reflooding was shown to reproduce satisfactorily temperatures evolutions and quench front progression in the spherical steel particles debris beds studied in the PRELUDE experiments for initial debris bed temperatures of 400°C and 700°C and for different water injection velocities [19].

Next steps will be to improve this reflooding model of severely damaged cores using in particular the data that will be produced in 2015 and 2016 by the larger scale PEARL experimental programme where 2D effects will be studied (e.g., by simulating flow bypasses around a compact debris bed, a configuration that probably existed during the Three Mile Island accident).

Starting in 2016, IRSN will continue its research work on the debris bed reflooding issue investigating the cooling of more complex debris bed configurations which are more prototypic (e.g., heterogeneous debris bed, debris beds with included compact zones) and the hydrogen production resulting from the reflooding. The work will encompass experimental work in the PEARL facility and models development for ASTEC SA code to be able to assess the efficiency of core reflooding for different conceivable configurations of degraded reactor core.

4.2. Corium Transient Behavior in the RPV Lower Head

In a core melt-down accident in a PWR, the degradation of fuel rods and melting of materials lead to the accumulation of core materials in the RPV lower plenum. The heat flux across the vessel wall becomes a key issue if a strategy of IVMR is implemented to keep the corium inside the RPV and stop the accident progression. Many experimental studies, such as COPO, ACOPO, BALI, RASPLAV-SALT and SIMECO, have dealt with the issue of heat flux distribution around a volumetrically heated melt pool. All those experiments were conducted with simulant materials with the aim of reaching a high Rayleigh number prototypic of a large melt pool in the RPV lower head. The objective of these investigations has been to determine whether the heat flux imposed by the melt on the RPV wall would not exceed the heat removal capability (critical heat flux) on the external surface of the RPV [20]. The heat focusing effect of a thin molten metal layer is considered as a potentially important cause for RPV failure because the RPV wall in contact with the metal layer would be submitted to increased heat fluxes.

IVMR by external cooling has been investigated by several countries for different reactor designs (existing or future concepts), assuming that the metal layer would be on top of the corium oxidic pool and would contain only steel. This corresponds to what one may call the ``standard analysis" [21]. Calculations have confirmed that thin metal layers on top of the oxide pool may produce heat fluxes on the RPV wall that could exceed the critical heat flux under certain conditions.

However, the existence of a miscibility gap between two liquids in the U-Zr-Fe-O diagram makes the situation more complex than assumed in earlier studies. The metal phase may actually be heavier than the oxide pool [22]. The first consequence is that it is necessary to derive models that are able to deal with conceivable configurations where the metal layer lies on top or below the oxide pool and with transitions between these configurations. The second consequence is that it is not possible to consider the different phases present in the lower plenum (solid debris or crusts, oxide pool, metal pool) as independent. Any relevant modelling of the stratification and solidification processes must include possible exchanges between phases and involve thermo-chemical considerations (as discussed in paragraph 4.6).

The position of the metal layer is likely to be initially at the bottom (Figure 2). Because of the small amount of molten steel and the incomplete oxidation of the corium pool, it is then mostly composed of heavy metals. Later, when the corium pool is almost fully oxidized, the position of the metal layer is likely to be the “classical” one, *i.e*. on top of the oxide pool, containing low U and Zr amounts dissolved in a light metal phase.

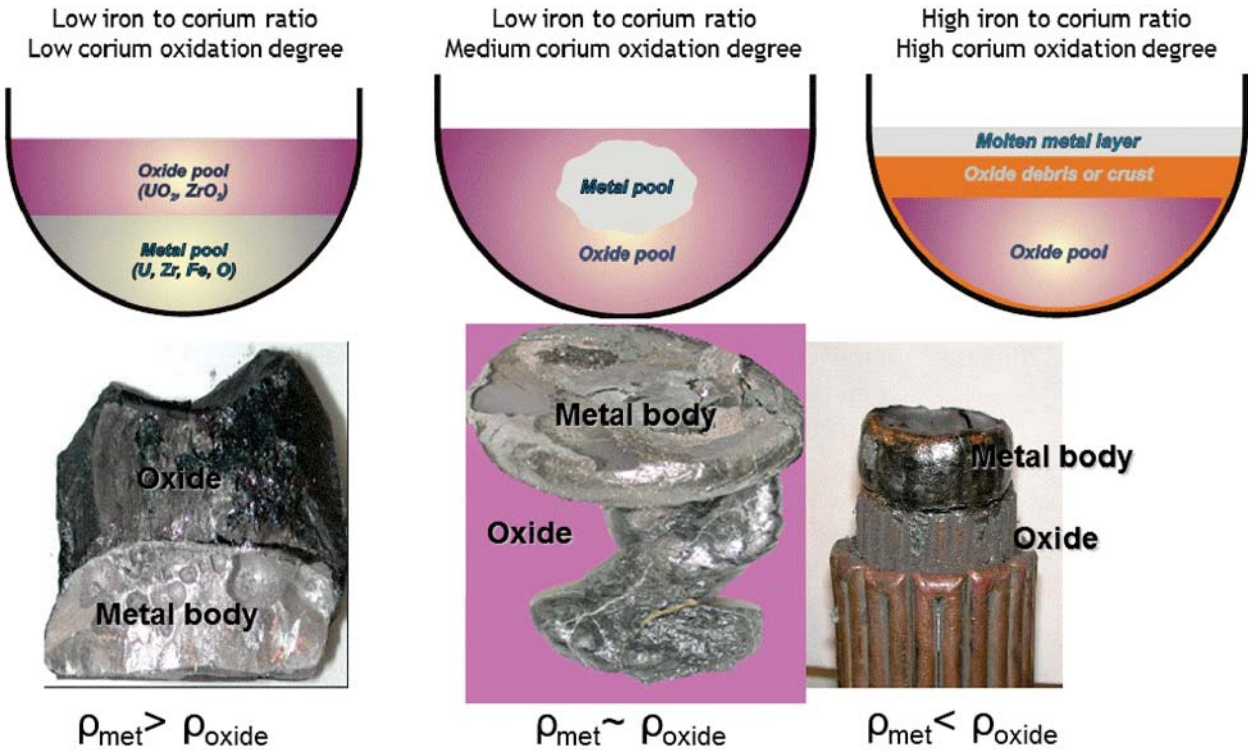

Figure 2. Examples of Conceivable Melts Layers Configurations in the RPV Lower Head during a SA Based on Experimental Observations (CORDEB Tests Performed by the NITI, Russia)

Between these two equilibrium configurations, three layers configurations may exist, in particular when the bottom metal layer becomes progressively lighter than the oxide pool. The evolution mainly depends on the amount of molten steel added to the melt in the RPV lower plenum. This depends on the reactor steel structures and the accident development in the RPV. Models have already been developed and implemented in SA codes like ASTEC to simulate transient evolutions [23]. However, such models generally consider a "global thermo-chemical equilibrium" in order to predict the compositions of layers and their densities but they do not explicitly look at the processes of mass transfers at the interfaces between metal and oxide, except for a tentative that was presented in [24].

On this second research axis, IRSN is therefore currently doing research to develop models able to evaluate the heat flux profile along the RPV wall in contact with molten material during a SA in view of assessing safety margins for IVMR. These models would describe the transient mass transfers between layers due to the changes of density of metal and oxide phases taking into account local conditions at the interface and not just a global equilibrium approach. Such developments are based on cold crucible experiments performed in the CORDEB program conducted by the Alexandrov Research Institute of Technology (NITI) in Russia to which CEA, AREVA, EDF and IRSN are collaborating. An initial program of about 20 tests was conducted between 2012 and 2015 using simulant corium materials in various configurations to quantify main involved processes: physico-chemical phases separation, layers inversion by turbulent instability, interactions between debris and layers. Further investigations will be conducted in the future in a larger collaborative initiative (*cf.* next paragraph).

4.3. Research on IVMR strategies

More globally on the IVMR issue, IRSN is coordinating a large international project involving 20 partners which is funded within the European Union H2020 framework. The project was launched recently (mid-2015) and its main objective is to develop knowledge and tools to appreciate on deterministic grounds if IVMR strategies are indeed safe for all conceivable melt-down accidents configurations in LWR concepts relying on such strategies for powers of 1 000 MWe or above.

The project will encompass:

- reviewing analytically the capacity to retain efficiently the corium inside the vessel thanks to external cooling for several kinds of reactors in Europe (existing or under project), following the standard safety assessment used for some existing VVER-400 and to new concepts such as AP-600, AP-1000 and APR-1400;
- investigating several options to improve the IVMR safety assessment for high power reactors by reducing the presently used conservatisms, notably for the evaluation of the focusing effect, and by proposing design evolutions to avoid inherent risks associated to vessel melt-through in a flooded cavity;
- providing new experimental results to assess the models used in the safety assessments, in particular to cover all possible configurations of corium in the lower plenum and all RPV lower head designs (e.g., for VVER-1000 and BWR geometries which have been less studied up to now);
- elaborating an updated an harmonized safety assessment approach for the analysis of IVMR that will be used for various types of reactors and grounded on best estimate validated models implemented in SA codes used in Europe.

In this project, IRSN will continue its research work on debris bed reflooding looking more particularly on the cooling of compact zones in debris bed and on corium behavior in vessel lower head, completing the on-going PEARL and CORDEB programs. Works performed by other partners will notably deal with external cooling (including technological development) and the mechanical resilience of a partially eroded vessel. An important part of the project will involve benchmarks for different reactor concepts in view of defining a shared methodology to assess the robustness of IVMR solutions.

4.4. Mechanisms of Fuel Coolant Interaction

For the third research axis, the activity is encompassed in the ANR-RSNR[7] "ICE" program dealing with the fuel-coolant interaction and is based on the conclusions of the OECD SERENA2 program [25]. Even though the SERENA 2 project generated valuable experimental data for prototypic melts in the Korean TROI (KAERI) and the French KROTOS (CEA) facilities for development or improvements of FCI models, there is still after the project a too large scatter in calculated dynamic loads. As such, the project results did not provide a definite resolution of the ex-vessel steam explosion issue. It however helped focusing the research towards FCI key mechanisms that need deeper understanding:

- the intrinsic mechanisms of jet fragmentation driving the interaction;
- the effect of material oxidation and solidification;
- pressurization processes during the explosion;
- 3-D effects (e.g., non-vertical corium jets, un-centered corium flows).

It was also concluded that the validity of FCI models developed from small-scale analytical experiments to assess FCI effects at the reactor scale needs to be further assessed. Syntheses of the current understanding [26] and modeling [11,12] of FCI were recently released.

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[7] ANR = French National Research Agency ; RSNR = Nuclear Safety and Radioprotection Research Program

It also appeared recently, in the light of the recent KTH observations [8], that the potential triggering mechanism in reactor situations needed to be clarified. Following the OECD SERENA program, it is still not possible to predict whether a steam explosion will occur for a given situation although conditions favoring spontaneous steam explosion are known. It was notably observed that spontaneous explosions often trigger in performed tests when the melt comes into contact with the bottom of the test section. In a reactor, three modes of interactions between the melt and the coolant are of particular interest corresponding to:

- a melt jet release in a flooded reactor pit at RPV failure: interaction leading to a meta-stable mixture of the fragmented and dispersed melt in water and vapor (and possibly non condensable gases such as hydrogen);
- a melt jet falling in a shallow pool of water in a partly filled reactor pit at RPV failure: interaction which may lead to the formation of a melt layer below the water layer, this occurs when the melt jet breakup length is larger than the pool depth. This results in a "stratified" melt-water configuration;
- a spread melt layer in an initial dry reactor pit flooded then with water for melt cooling: interaction which also results in a "stratified" melt-water configuration.

The "stratified" configurations have been studied experimentally and analytically. A steam explosion appears to be easily triggered in stratified configurations with a direct contact between melt and water (in the absence of crust on top of the melt). When the interface between the melt and the water is smooth and stable with little mixing, the explosion would result in moderate pressure loads, with no expected damage to the containment internal structures and leak-tightness. However, KTH experiments revealed that in the configuration with a melt jet falling in a shallow pool of water, the melt layer under the water was subject to important instabilities with the formation of an unstable mixed layer which may be at the origin of the triggering of explosions with larger efficiencies that those reported in earlier experiments. The issue is important in terms of safety since the configuration of a melt jet impingement on the reactor pit basemat through a water pool is considered as plausible in many cases.

It should be further noticed that experiments corresponding to the last type of interaction (flooding of a spread melt layer), never led to spontaneous steam explosion. As shall be discussed in the next paragraph, such experiments with prototypic corium are still on-going for the sake of analyzing cooling of melt interacting with concrete by water flooding. The interpretation is that the cooling rate is sufficiently slow to provide for the formation of a surface crust which sufficiently stabilizes the melt layer.

The current simulation tools for the evaluation of FCI, such as MC3D developed at IRSN, mainly consider the pre-mixing and explosion phase of a dispersed melt in a water-gas mixture (Figure 3). The triggering is considered as a stochastic event that cannot be modelled directly and for which no accepted criterion exists. A model for stratified configurations exists but it considers only smooth interfaces. Efforts should be placed in the development of a model describing interface instabilities between corium and water.

On the FCI topic, following the OECD SERENA2 conclusions, a Topical Opinion Paper on ex-vessel steam explosion is being elaborated in the OECD frame. The paper should make proposals on future research activities in the field. IRSN is supporting work in view of assessing better the risk (triggering, pressurization, effect on structures) in addition to the on-going ICE program.

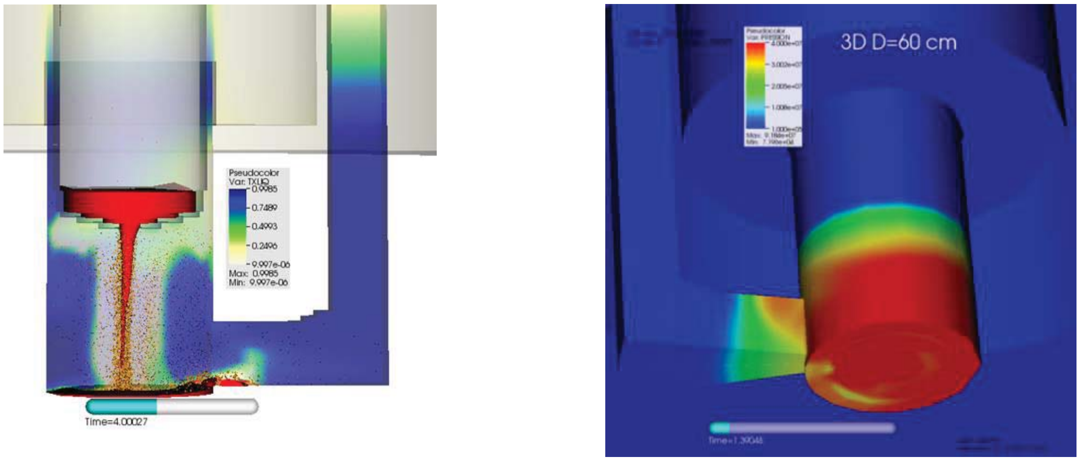

Figure 3. Examples of MC3D Calculations of the Corium-Coolant Pre-mixing after RPV Failure (Left) and of the Explosion (Right) Provided as Illustrations (see References [11,12] for More Detail on MC3D modelling)

4.5. Mechanisms of Corium Cooling during MCCI

In the safety assessment of operating French NPPs, calculations using MCCI models for dry conditions developed from recent R&D progresses [27,28] show that, except for very specific situations such as for the Fessenheim NPP (addressed by significant plant modification, *cf.* paragraph 2.4) no basemat melt-through can occur before the delay of 24 hours needed for implementing population protection measures. Nevertheless, uncertainties on these models do not allow excluding basemat melt-through after 24 hours. No simple experimental testing that would contribute to significantly reduce these uncertainties had been identified mainly because of the difficulty to perform analytical experiments using prototypical corium-concrete melt mixtures. A few years ago, IRSN thus decided to focus R&D efforts on the assessment of MCCI mitigation measures, consistently with the orientation and schedule of the on-going NPPs safety evaluations. Corium cooling by water injection below the corium has been recognized as a very efficient measure [29] but it requires controlling the water injection flow to avoid a too fast containment over-pressurization. Moreover, the technical feasibility of back-fitting such a mitigation measure in French operating plants is not established. Thus, even if R&D activities are still on-going on corium-concrete melts cooling by bottom water injection - which is also of interest for future designs -, more efforts are presently put on acquiring knowledge on corium-concrete melts cooling mechanisms by top water flooding during MCCI since such a mitigation measure can be more easily back-fitted in French operating plants.

Several mechanisms can lead to the cooling of a flooded corium-concrete melt. Bulk cooling will be the more efficient process at short term before a crust is formed on the melt top surface. Then cooling may be obtained by two different processes that are the water ingression [30, 31] through the crust and the melt ejection [32] due to gas generated by the concrete decomposition during MCCI (Figure 4). Associated models have been developed from analytical experimental programs and larger scale experimental programs performed with prototypical materials in the ANL facility (OECD MCCI [33] and MCCI2 programs, complemented recently by additional experiments under a specific EDF-IRSN-USNRC-GdF-Suez agreement).

These models have been implemented in the ASTEC code [10]. They include a dedicated debris bed layer that can be formed above and apart from the upper crust, thus allowing a proper account for the continuous cooling of already ejected debris by top water injection. A simple energy balance for the debris bed is applied assuming that the built-up debris remain at saturation temperature because of the much larger dry-out heat flux than in case of the heat flux extracted from the upper crust. Indeed, assuming that the debris bed porosity is larger than about 0.3 and the debris size a few mm, which are likely assumptions, the dry-out heat flux deduced from available correlations [34] reaches about 1 MW/m^2 which permits to extract the total decay power of a core inventory even for a large PWR reactor. In case of insufficient water inventory, this dry-out is supposed to start at the bottom of the debris bed. A preliminary validation of this debris layer modelling has been successfully achieved at IRSN vs. the data obtained recently in the ANL facility (CCI-7 and CCI-8 experiments). As far as water ingression is concerned, the detailed model proposed by Epstein [30,31] based on considerations on the material creep and cracking behaviour, has been implemented and taken into account in the heat flux continuity equation of the upper crust; this model that was validated against SSWICS tests is indeed considered to be sufficiently mechanistic. As to melt ejection, the PERCOLA model [32] for the melt ejection hydrodynamics has been implemented in ASTEC. This so-called "fountain" model is combining a double phase upwards flow through the hole (accounting for the impact of the melt viscosity) and a lateral liquid pouring. It must be underlined that, as a first step, a fixed hole size and density is assumed in ASTEC. A combined use of PERCOLA with available models for determining, as far as possible, the precise ranges of hole diameter and hole density corresponding to the MCCI situation is planned.

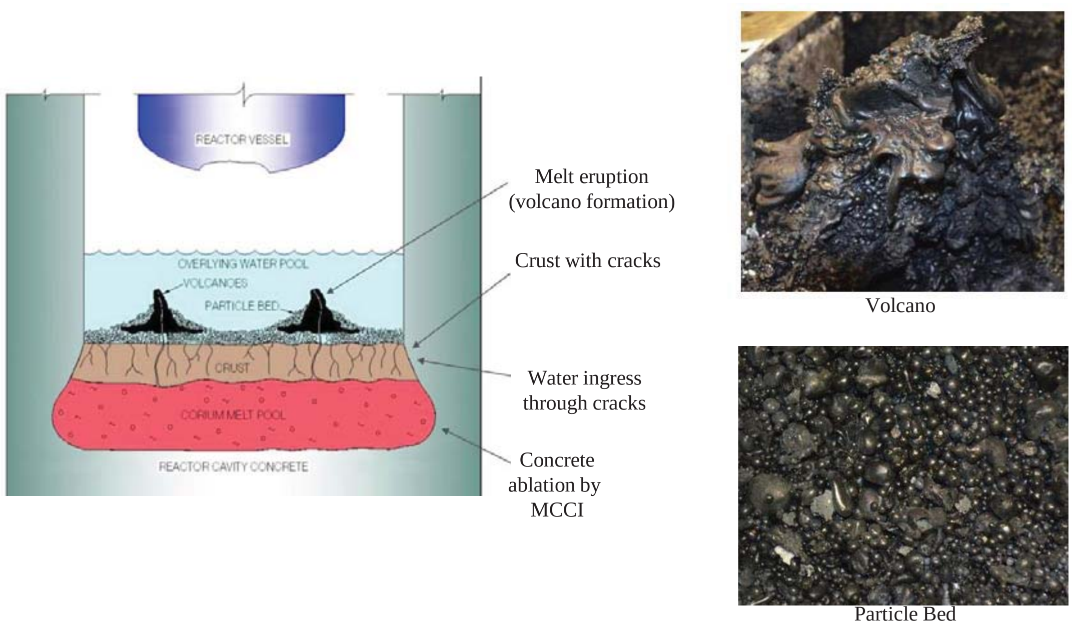

Figure 4. Schematic Illustration of Corium-Concrete Melt Cooling Processes with Top Flooding During MCCI (Left); Photos of a Volcano and a Particle Bed Formed During an ANL Top Flooding MCCI Test

Additional experiments are needed to study the MCCI over longer duration with more prototypic compositions (reinforcement bars and metal content in the melt) to provide for a better confidence in the extrapolation of experimental data to reactor situations. For that, experimental drawbacks including the crust anchoring on the experimental cavity walls or the control of the injected power to reproduce the decay heat would have first to be solved. These experiments are for both model development and validation purposes. They would be completed by analytical studies using the MC3D code. Moreover less idealized situations have been identified that can be safety relevant issues. As an example, one can mention the relocation of a debris stack formed by a lateral vessel rupture and corium spreading in a part of the reactor cavity pit initially flooded by a shallow water layer (around 1m depth). Current models cannot tackle directly this issue and significant R&D efforts would be necessary including possible experiments in upgraded or new facilities.

4.6. Quality improvements of thermodynamic data for modeling corium interactions in SA

Progress in understanding physical chemistry of high temperature corium melts and in quality of data used for thermodynamic modelling of SA phenomena is continuous. Recent reviews on the subject are provided in [35, 36]. Data updates are notably implemented in the NUCLEA thermodynamic database that are developed and maintained by IRSN and are considered for SA calculations notably using the ASTEC code. These updates are generally based on results of small scale tests performed in support of phase diagram determination, e.g. recently by ITU in Germany in the U-O-Zr system, by NITI in the framework of the ISTC Projects in some binary systems involving concrete components, by UJV Rev. in the Czech Republic within the SARNET network in the oxide part of the U-O-Fe-O-Si systems. Improved modelling for key systems for in-vessel (notably, the U-Zr-Fe-O system) and for ex-vessel (notably, the CaO-SiO2-CrO_x and Al_2O_3-FeO_x-CaO-SiO_2 systems) on basis of corium behaviour modelling have recently been implemented. The overall consistency of the database has and will be assessed by the analysis of its capabilities versus more global experiments with corium or systems containing UO_2 such as the VULCANO tests performed by CEA in France regarding the MCCI issue, or the cold-crucible tests done by NITI in the MASCA Programme, related to the in-vessel stratification issue for the oxide-metal corium. This work is essential to improve the modelling of corium behaviour in SA.

For RPV lower head behaviour, recent cold crucible experiments have confirmed MASCA results showing that components partitioning between oxidic and metallic melts results under certain conditions in density changes which impact the configuration of formed layered molten pools, i.e. with possible inversion of oxidic and metallic layers [37]. As illustrated in Figure 5, the NUCLEA database allows to satisfactorily calculate the compositions of the oxidic and metallic layers measured in the analytical tests of the MASCA programme devoted to the interaction between different Fe masses and a sub-oxidised corium. As discussed earlier, such inversion processes may induce transient focussing effects on the RPV walls. It is therefore of importance to model properly these effects. A significant impact of B_4C (for rather rich compositions in B_4C) on chemical equilibrium in the miscibility gap domain of the U-Zr-Fe-O system and on the densities of coexisting melts was also experimentally confirmed. Such an effect has to be considered to assess the corium behaviour in the RPV lower head for reactors containing B_4C as control material such as BWRs.

For corium behaviour in MCCI, the understanding of the effect of interactions of light elements such as Ca and Si with heavy U and Zr elements is progressing. This is of special interest since the velocity of the concrete erosion by the corium melt is determined by the corium properties (density and gravity effects, miscibility of phases, crust formation at interfaces).

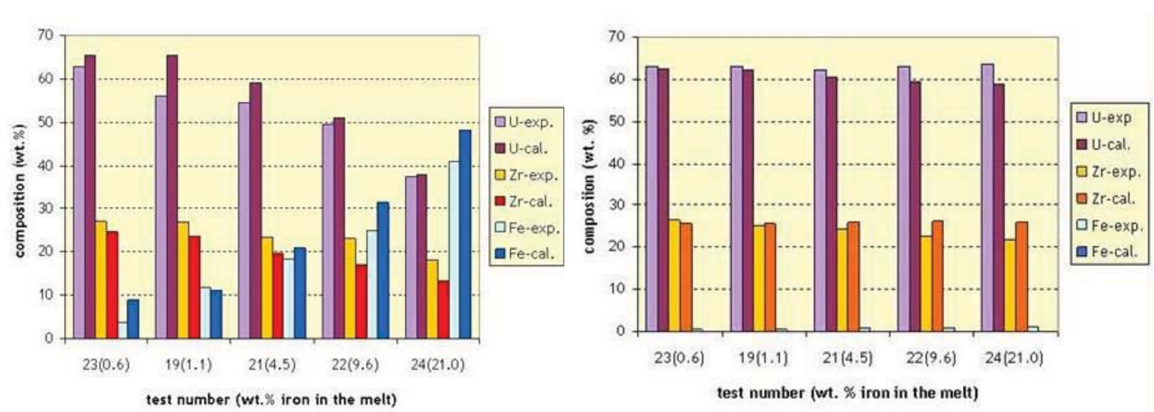

Figure 5. Examples of Comparison of Calculated (with NUCLEA database) and Measured (in Analytical MASCA Tests) Compositions of the Oxidic and Metallic Layers

5. CONCLUSIONS

Significant research efforts to improve knowledge of corium progression and of the induced risks of RPV and containment failure are underway at IRSN. Three major issues are investigated in relation to the demonstration of the robustness of possible mitigation strategies:

- RPV resilience with IVMR strategies for reactors using such strategies and for the design of future reactors;
- Ex-vessel steam explosion and induced risk of containment failure in operating reactors and related management of water supplies in the reactor pit;
- Ex-vessel corium cooling during MCCI by water injection on corium top-surface in operating reactors.

Such research is performed notably in the European Commission IVMR project launched in 2015, the French ICE project and the ANL CCI project conducted by EDF. EDF and IRSN will notably use the results of such research to reassess mitigation strategies involving a dry or a wet reactor pit.

Investigating the above mentioned issues is of high interest for any reactor design, existing or under development, in view of establishing firm technical grounds for the deterministic demonstration of the efficiency of mitigation features for the global safety enhancement of the NPPs.

NOMENCLATURE

BWR = Boiling Water Reactors
DCH = Direct Containment Heating
EFCVS = Emergency Filtered Containment Venting Systems
FCI = Fuel-Coolant Interaction
(LT)SBO = (Long Term) Station Black-Out
EVMR = Ex-Vessel Melt Retention
IVMR = In-Vessel Melt Retention
MCCI = Molten Core Concrete Interaction
NPP = Nuclear Power Plant
PWR= Pressurized Water Reactor
RCS = Reactor Coolant System
RPV = Reactor Pressure Vessel
SA = Severe Accident
SAM = Severe Accident Management
SAMG = Severe Accident Management Guidelines
SAMM = Severe Accident Management Measures
(I)-SGTR = (Induced)-Steam Generator Tube Rupture

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