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Science of nuclear safety post-Fukushima

Research at the CEA in the field of safety in 2nd and 3rd generation light water reactors

La recherche au CEA dans le domaine de la sûreté des réacteurs à eau légère de 2ème et 3ème génération

Philippe Billot

Commissariat à l'énergie atomique et aux énergies alternatives, Centre de Cadarache, Batiment 101, 13108 Saint-Paul-Lez-Durance cedex, France

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ABSTRACT

The research programs at the CEA in the field of safety in nuclear reactors are carried out in a framework of international partnerships. Their purpose is to develop studies on:

- The methods allowing for the determination of earthquake hazards and their consequences;
- The behaviour of fuel in an accident situation;
- The comprehension of deflagration and detonation phenomena of hydrogen and the search for effective prevention methods involving an explosion risk;
- The cooling of corium in order to stop its progression in and outside the vessel thereby reducing the risk of perforating the basemat;
- The behaviour of the different fission product families according to their volatility for the UO_2 and MOX fuels.

RÉSUMÉ

Les programmes de recherches du CEA dans le domaine de la sûreté des réacteurs nucléaires sont conduits dans le cadre d'un fort partenariat international. Ceux-ci visent à développer les études sur :

- Les méthodes permettant la détermination des aléas sismiques et de leurs conséquences ;
- Le comportement du combustible en situation accidentelle ;
- La compréhension des phénomènes de déflagration et de détonation de l'hydrogène et la recherche de méthodes de prévention du risque d'explosion ;
- Le refroidissement du corium visant à arrêter sa progression en cuve et hors cuve réduisant ainsi le risque de percement du radier ;
- Le comportement des différentes familles de produit de fission selon leur volatilité pour les combustibles UO_2 et MOX.

E-mail address: philippe.billot@cea.fr.

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1. Introduction

The mission of the French Commission of Atomic and Alternative Energies (CEA) is to place its expertise and competence at the disposal of its partners in order to meet the needs of R&D on 2nd and 3rd generation nuclear systems now used by French industrialists and research partners, along with our foreign participants whose demands continue to increase every day. The CEA thereby contributes to a higher level of safety in reactors by carrying out programs on accidents and by taking the results into account both in the design basis of reactors and outside of the design basis.

Nuclear reactors may be characterized by two key elements that justify a specific treatment in terms of safety and accident prevention:

- At the time of an incident leading to an emergency shutdown of a reactor, if it is not possible to resume normal operations quickly, it is nevertheless crucial to be able to continue cooling the fuel for a very long period of time in order to remove the decay heat given off by the fission products;
- If this cooling ceases to be done correctly, there is a high probability that the accident will lead to damaging the enveloping materials confining the radioactive products, namely the cladding of the fuel elements, the primary circuit of the core cooling system and the reactor containment, thereby causing a release of radioactivity into the environment.

Nuclear reactor safety relies from the very start on the concept termed, "defense in depth". The goal of this concept is to prevent core meltdown and radioactive release into the environment at all costs. It may be summarized in the following manner: Beyond all the precautions taken to prevent accidents, we nevertheless postulate, in principle, on their occurrence. We must therefore evaluate their consequences and make special provisions designed to limit their effects.

The Fukushima accident does not call into question the concepts and principles that have been put forward until now, but it has already proven rich in lessons to be learned in the logic of permanent, unceasing improvement that must always be on-going in the field of nuclear safety.

Reactor accident studies are currently carried out in the laboratories of the CEA, the IRSN and their partners. The major stakes involved in this research concern the prevention of accidents, and in the logic of defense in depth, the containment of their consequences. The R&D in this field must be understood more perfectly in order to comprehend the extreme complexity of the physical phenomena involved so that models applicable to the reactors can be developed. These models must, in particular, enable us to anticipate how an accident will unfold in its overall manner and permit us to evaluate the pertinence of the means implemented to limit its consequences.

In order to accomplish this, research programs are essentially carried out according to a triple procedure associating *experimentation, modelling and studies*, relying as much as possible on research work performed upstream by our "academic" research partners. Most of the time, these programs are conducted within a framework of broad national collaboration with the IRSN and international organizations.

2. Research and development on accidents and the limiting of their consequences

This is a question of verifying the aptitude of the safety systems in controlling the consequences of postulated accidents and in the logic of defense in depth, defining measures that must be implemented in order to limit consequences and even protect the population in the case of a failure in these systems.

2.1. The fuel in an accident situation

Here, this research is mainly justified by evolutions in fuel management, suggested by operators whose goal is to reduce electricity production costs. There are three main consequences: The development of new types of fuel; the use of increasingly complex simulation tools aimed at overcoming conservative ideas taken into account at an earlier date in the safety demonstrations; and finally the implementation of operating conditions necessitating the achievement of an experimental basis of the validation of these tools.

The research is particularly concerned with understanding the thermo-mechanical behaviour of the fuel at the time of two types of accidents that can lead to a loss of integrity in the fuel and to an evolution towards an accident ending in core meltdown: The accident involving a loss of primary coolant (LOCA) and accidents involving uncontrolled reactivity insertion (RIA).

2.1.1. The loss of primary coolant accident (LOCA) (Fig. 1)

The problem caused by this accident involves that of the reactor's capacity to cool the core after a break in the pipes of the primary circuit. It serves as a basis of design in all the major safeguard systems in this type of reactor and limits the reactor's range of operation. Generally speaking, the acceptable characteristic of the LOCA relies on the demonstration of the "cool-ability" of a core throughout the entire phase of the accident including the long term (the re-flooding and beyond that). What is primarily at stake is the resistance of the fuel rod cladding particularly during and after the core re-flooding given their rate of oxidation and hydriding developed during normal operation and in the course of an accident.

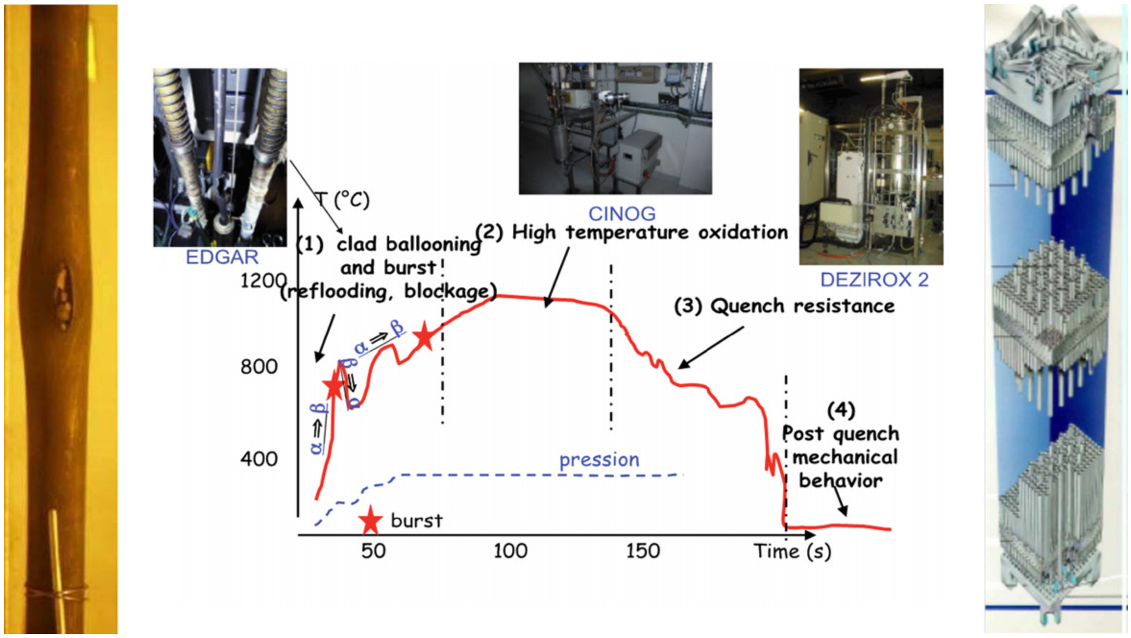

Fig. 1. Temperature kinetics in LOCA conditions.

At the CEA, considerable effort has been devoted to this subject over the past 20 years by working in partnership with the EdF, AREVA and the IRSN. It combines the analytical experimental approach and the modelling of physical processes that will enable us to:

- Provide behavioural laws governing ballooning-clad failure thanks to the implementation of dynamic tests conducted under internal pressure;
- Study the behaviour in oxidation at high temperature and its consequences on the "mechanical resistance to quenching" and "post quenching" of the fuel cladding. For example, the CEA has contributed to clarifying and quantifying the fragilizing role of hydrogen and revealed a significant effect on the residual ductility in the final cooling scenario;
- Model the behaviour of the cladding in LOCA conditions. The CEA in particular has developed physical models coupling the metallurgical evolutions of the cladding with its thermo-mechanical behaviour.

With regard to modelling, in the future it will be necessary to move on to detailed, three-dimensional models in order to consider a complete assembly and to take into account more effectively the effects of the interaction between the fuel rods.

2.1.2. Uncontrolled reactivity insertion accidents (RIA)

Here, the purpose of our research is to study the consequences of a massive and rapid deposit of power in the fuel rod following a reactivity accident: Clad failure, fuel ejection, fuel/coolant interaction. The selected reference accident is the ejection of a rod cluster control assembly. For the most part, the whole issue consists in defining the criteria of fuel resistance at the time of a rapid power and temperature transient (Fig. 2). This transient is characterized by a very short period (ranging from a few milliseconds to a few tens of milliseconds). The deposited energy in the fuel pellet can be great (on the order of about 100 cal/g), just like the rise in the associated temperature (up to approximately 2300 °C at the heart of the fuel with kinetics in the temperature rise on the order of 10,000 °C/s).

The work involved in this research necessitated the use of heavy test means, particularly, the CABRI reactor which is operated by CEA, Cadarache for the needs of the IRSN.

The *CABRI-REP-Na* program which was carried out by the IRSN in the CABRI reactor during the 1990s was subsequently used for characterizations of fuel rod tests in the Spent Fuel Studies Laboratory. These allowed us to examine the behaviour of fuel rods at high burn-up in the case of a reactivity accident resulting in the ejection of a control rod, specifically highlighting:

- The role played by hydrides in the risk of clad failure;
- The contribution of the fission gases on the grain boundaries;
- The importance of spalling on the oxide layer in the mechanical resistance of the cladding.

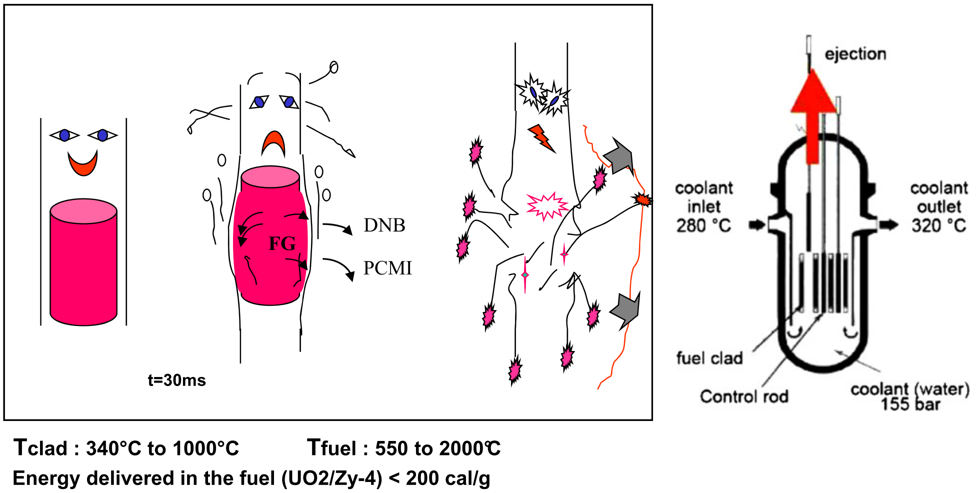

Fig. 2. Criteria of fuel resistance to RIA conditions.

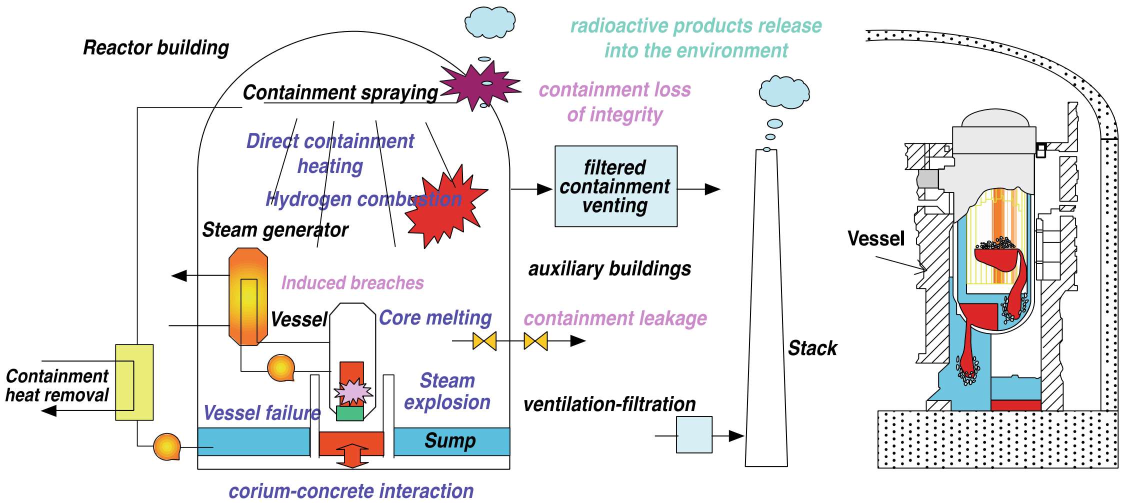

Fig. 3. Accidents with meltdown.

At present, a new program is being prepared which is to be piloted by the IRSN within the framework of the Organization for Economic Co-operation and Development (12 countries are involved, each of them with several participating organizations). This program will study the behaviour of different types of fuel pellets (UO_2, MOX, gadolinium pellets, doped fuel, ...) and different types of both current and advanced clad materials on the basis of experimentation in the CABRI reactor with characterizations to be carried out after tests on fuel rods from CEA, Cadarache.

2.2. Severe accidents (accidents involving core meltdown: Fig. 3)

For these accidents, resulting from an accumulation of failures within the facility, the major purpose of the research is to evaluate and attenuate the consequences. This is a matter of evaluating the risk of loss of containment consecutive to the complete or partial meltdown of the reactor core, as well as qualifying the prevention and mitigation devices.

This accident category has not been taken into account in the design of the 2nd generation reactors. However, for certain accident situations, improvements have been made at the time of each safety re-examination. The present programs of study enter into a continuous approach of improving the reduction of risks and limiting the consequences. In conformity with the demand of the French Nuclear Safety Authority (ASN) of 1993, severe accidents must now be taken into account in the design of any new reactor project and especially for the 3rd generation reactors.

The R&D programs deal with the following themes:

- The behaviour of corium (cladding materials, fuel and structures that have melted down in the reactor core) in the severe accident scenario: Corium in the vessel, vessel resistance, corium outside the vessel, the water/corium reaction;

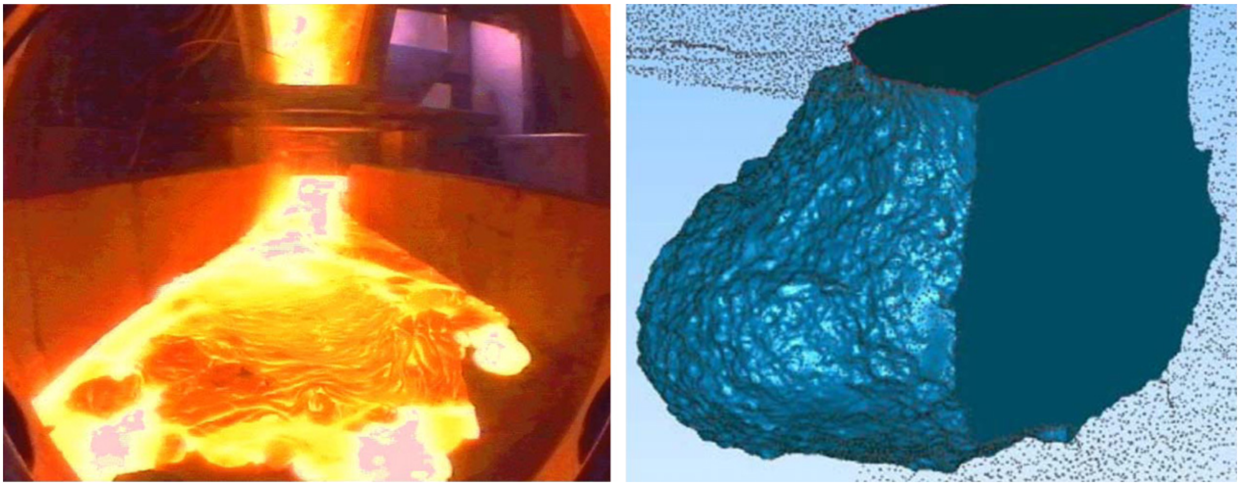

Fig. 4. Ablation of concrete by corium at 2500 K (VULCANO Test).

- The safeguard of the containment in accident situations: Thermal-hydraulics inside the containment, study of the hydrogen risk, behaviour of the structures;
- The FP source term (FP): FP release during the degradation phase, FP transport in the primary circuit, behaviour of the FPs in the containment and their release into the atmosphere;
- The development of numerical simulation tools and the production of studies (especially within the framework of the safety probabilistic evaluations).

These programs bring together the different actors from the nuclear community in national and international collaborations. At the European level, the excellence network, SARNET coordinated by the IRSN with active involvement of the CEA, contributes to sharing the available scientific resources in the field of severe accident study in order to make progress in handling questions that still remain unsolved due to their complexity.

2.2.1. Core meltdown and the progression of the accident

The different stages in the progression of the core meltdown accident are the following:

- Core uncovering;
- The partial or total meltdown of the core's constitutive elements, leading to a loss of geometry by local, then general collapse (debris beds);
- Corium formation (masses of fuel along with melted and mixed structure materials maintained in a state of meltdown by the decay heat of the fission products). According to the amount of water available in the vessel, this corium progresses more or less rapidly through gravitational pull towards the bottom of the vessel;
- Thermal ablation of the vessel floor and its possible perforation.

The research programs on corium today mainly deal with the re-cool-ability of a degraded core (in the form of debris or corium inside or outside the vessel) and also with the interaction of corium with concrete and with water.

The experimental means available mainly concern the PLINIUS platform of the CEA and the materials and thermal-hydraulics platform of the IRSN.

Located on the CEA-Cadarache site, the PLINIUS platform (Platform for Improvements in Nuclear Industry and Utility Safety) can provide prototypical corium in sizeable quantities for experimental purposes. The uranium used is spent uranium of which the physico-chemical properties are identical to those of uranium in nuclear fuels. According to specific needs, the corium produced is of rather varying compositions obtained from a mixed base mainly composed of uranium oxide, zirconia and iron oxides (UO_2-ZrO_2-Fe_xO_y). This platform is made up of several test facilities:

- The Versatile UO_2 Laboratory for Corium ANalysis and Observation: VULCANO allows us to melt 50 to 100 kilograms of corium through chemical heating. The bath is poured into a test section. The decay heat is simulated through inductive heating. The phenomenology of the spreading and interaction transient with the substrate during spreading are studied. This type of device thereby permits the study of corium solidification, the thermal-hydraulics of corium baths, corium progression in the debris beds and the long-term interactions of corium with concrete or ceramics (Fig. 4).
- COrium LIquids and MAterials: COLIMA enables researchers to melt several kilograms of corium by induction. The crucible is placed in a 1.5 m^3 containment that simulates the expected environment within the containment in the case of a severe accident occurring in a PWR (5 bars, steam, 150 °C), either in an oxidizing or reductive atmosphere. The

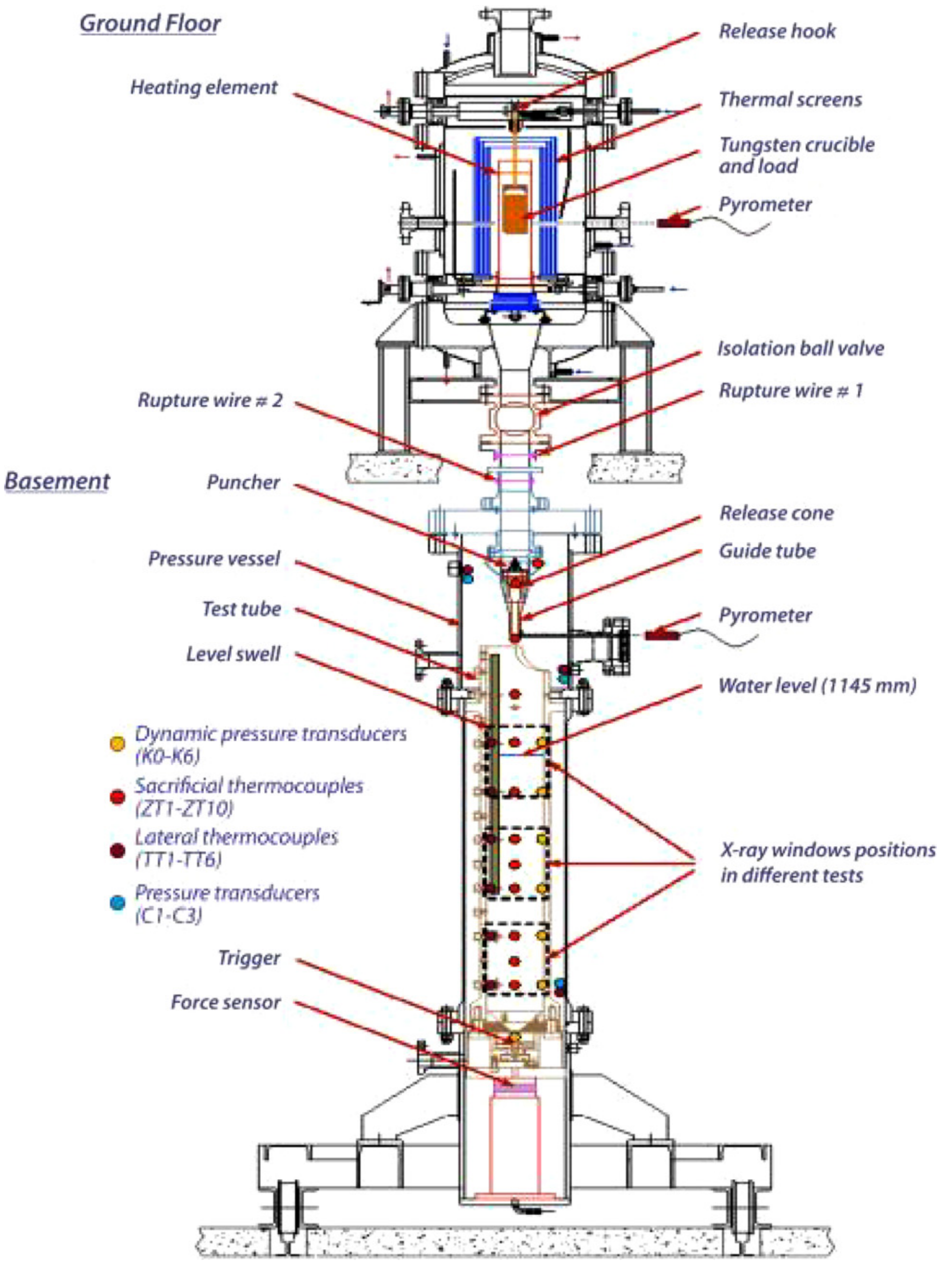

Fig. 5. The CEA KROTOS device devoted to the study of the corium–water interaction.

tests are aimed at the physico-chemical analysis of interactions between the fuel and other materials, the analysis of corium aerosols and the determination of physical properties.

- The VIscosite – Temperature Installation: VITI is a high temperature facility and its purpose is to study the properties of the materials, mainly their viscosity and superficial tension. It allows researchers to use the spent uranium contained in the simulated corium. The heat through induction produces a heat without contact and a characterization of the sample.
- KROTOS (Fig. 5) is a facility devoted to the study of the fuel–coolant interaction. The transfer between the melted corium and water can be so intense that the time scale of thermal transfers is shorter than the propagation of the pressure waves that cause the appearance of a shock wave. This shock wave amplifies because when moving through the corium–water mix, the drops of corium break up into fragments and exchanges are intensified. In this facility, 4.5 kilograms of prototypical corium can be melted and poured into a test section filled with water. The pre-mixing and explosion phases are thereby studied.

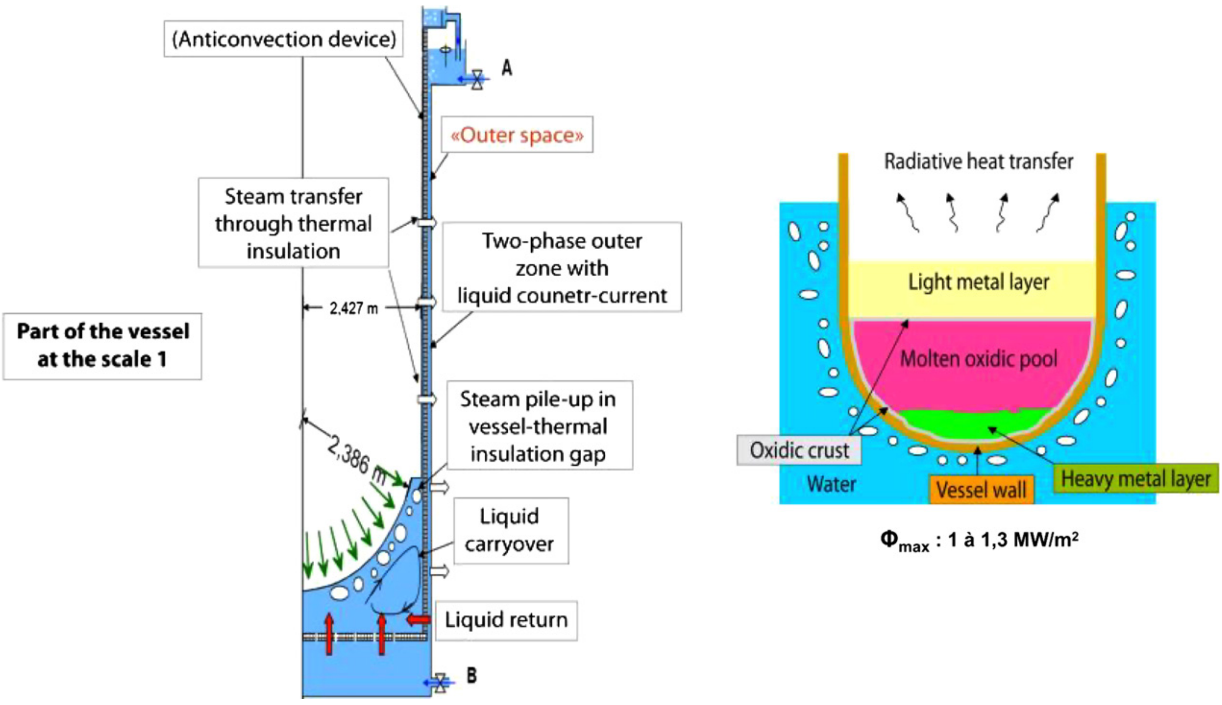

Fig. 6. Evacuation of the thermal flux by re-flooding the reactor pit.

2.2.1.1. Corium retention in the vessel The CEA essentially works on the external cooling of vessels through the flooding of the reactor pit, in the frame of bi-lateral co-operation with the EdF: Knowledge of the critical flux that can be evacuated at the wall of the vessel is a fundamental piece of data in assessing the capacity of maintaining all or part of the melted core in the vessel. The presence of insulation around the vessel is a penalizing situation when considering this objective.

The actions now underway at the CEA concern:

- The development of the LEONAR calculation code that will enable researchers to assess the probability of basemat perforation and to carry out sensitivity studies particularly those dealing with sequences of water injection into the reactor pit at different stages in the accident sequence.
- The CALO experimentation which permits scientists to carry out hydrodynamic tests aimed at demonstrating that the presence of insulation is not an insurmountable obstacle to the circulation of a two-phase flow around the vessel of a PWR or to the evacuation of the critical flux on the vessel wall (Fig. 6).

The major safety issues associated with these actions concern:

- The absence of restrictions to injecting water into the vessel or into the reactor pit in severe accident situations by demonstrating, in particular, the absence of a steam explosion risk outside the vessel as a consequence on the containment;
- The evaluation of the impact of cooling the in-vessel corium on the level 2 safety probabilistic studies.

2.2.1.2. Progression of the corium outside the vessel The perforation of the vessel would result in the transfer of all or part of the corium into the reactor pit. The arrival of corium triggers a "thermal" erosion of the concrete in the basemat, commonly referred to as the Corium–Concrete Interaction (CCI). Perforating the basemat would lead to a transfer of a considerable part of the core's initial radioactivity into the ground.

The R&D themes of potential interest focus on the corium–concrete interaction, both dry and underwater. Concerning the dry CCI, the experimental results obtained up-to-now do not yet allow us to predict accurately the time leading to the perforation of the basemat itself even if this precision has already been considerably improved. Serious thought is now being given to defining just how to progress in dealing the most effectively with this issue by combining experimentation and modelling. The CEA, in partnership with the EdF, is currently developing the TOLBIAC calculation code in order to analyze the phenomena that come into play during the corium–concrete interaction. It is therefore possible to quantify the kinetics of concrete ablation based on assessment reports and on the thermal fluxes for a wide range of configurations and different materials (siliceous concrete and silico-calcareous). Validation of these tools is being pursued by integrating the results obtained from tests currently underway in the VULCANO facility of the CEA using artificial concrete for parametric

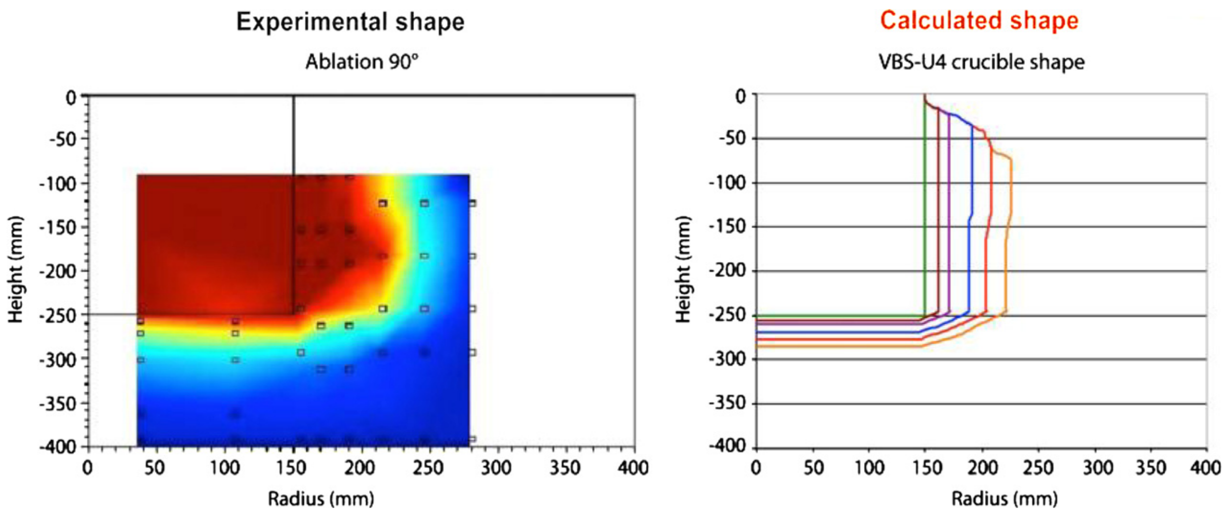

Fig. 7. Calculated and experimental anisotropic ablation profiles of a concrete rich in silica.

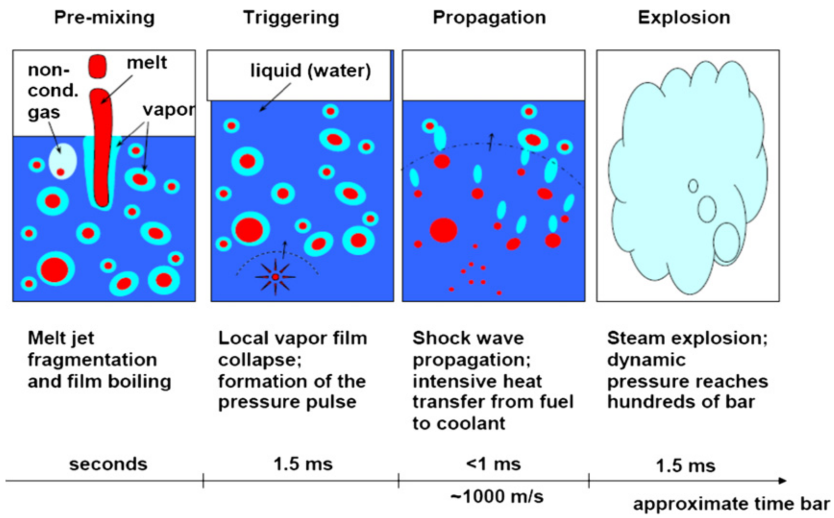

Fig. 8. Pre-mixing and fragmentation phase before the explosion.

studies or molten oxide and metal baths. Their goal is to determine the influence of the steel layer on concrete ablation (Fig. 7).

Work will presently focus on stopping corium progression, necessitating specific mitigation means (flooding the corium, re-cooling the basemat).

Finally, the contact of corium sufficiently dispersed with water either from the vessel or from the reactor pit (corium–water Interaction) might produce a steam explosion, the energy of which would be likely to affect the integrity of the containment (Fig. 8). The actions underway at the CEA (with the support of both the IRSN and the EdF) concern the realization of a program involving both experimentation and the modelling of these phenomena on the KROTOS facility in an international framework and the co-development of the MC3D software by the IRSN-CEA.

Generally, the experimental research carried out on the PLINIUS platform of the CEA has enabled us to make considerable progress in our understanding and modelling of phenomena linked to severe accidents in nuclear reactors: Aerosol release and transport above the corium bath, the spreading and solidification of molten corium melts, ablation of the concrete by corium as well as the experimental validation in prototypical materials of the COMET concept of in-reactor corium retrieval. It has also provided us with an experimental database for the validation of calculation codes. The R&D that remains to be done must now deepen our knowledge of scenarios involving the spreading of corium in the reactor core, the question of the "cool-ability" of debris and corium beds, the retention of corium in the vessel through external cooling, the studies devoted to mitigation systems with regard to the risk of perforating the basemat and the modelling of the consequences on the mechanical resistance of the reactor structures after a steam explosion.

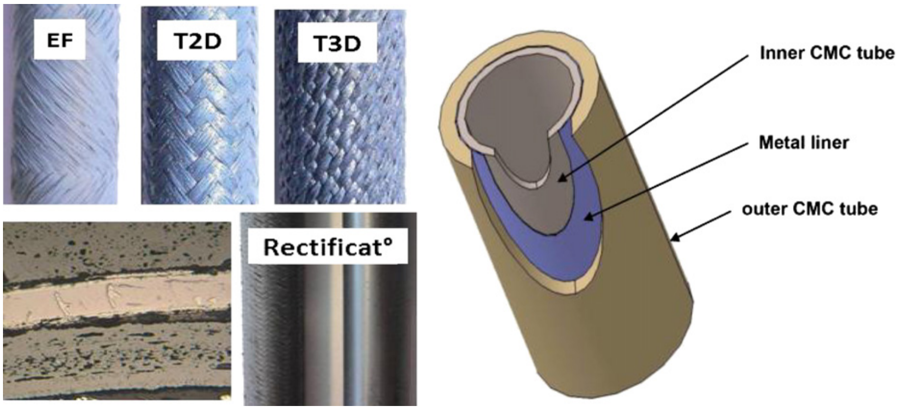

Fig. 9. The composite SiC/SiC fuel cladding.

2.2.2. The hydrogen risk

In a situation of prolonged loss of primary coolant leading to core meltdown, once the fuel has reached temperatures higher than 1200 °C, oxidation of the zirconium fuel cladding caused by steam, increases greatly and releases energy and hydrogen in the primary circuit (500 kg to 1500 kg of hydrogen for an EPR according to accident scenarios). When the hydrogen, released in the containment (through a possible break in the primary circuit) mixes with air and steam in a proportion ranging between 4% and 75% in volume, the initiation of combustion triggers an explosion either of a deflagration or detonation type. The detonation produces a pressure wave likely to lead to the failure of the containment. Mastery of the hydrogen risk is therefore an important element in the safety of the specific reactor type because it is associated with a risk of damaging the ultimate barrier that would thereby lead to the loss of radioactivity containment.

In order to cope with this risk, studies carried out have demonstrated the effectiveness of hydrogen catalytic recombiners which are currently placed in all reactor containments of the French nuclear infrastructure, although they cannot totally eliminate the risk of an explosion.

The goal of research carried out in the field of the hydrogen risk currently concerns perfecting our knowledge of the deflagration or detonation phenomena and their mechanical consequences on the structures. The research also deals with the recombination techniques or even other prevention methods. In addition to this, it involves intensifying our research on hydrogen concentration measurements (sensors, ...).

The scientific “locks” concern the distribution of hydrogen concentrations and the propagation of hydrogen flames in big geometries. Indeed, the flammability and the mode of hydrogen flame propagation depend on its distribution within the reactor containment, its stratification and the presence or absence of inert gases (steam, nitrogen, ...). R&D priorities have therefore focused on the predictive models of stratification and the mobilization of hydrogen performed by the re-combiners placed in the containments, the calculation of flame progression on a grand scale and the effect of explosions on the steel and concrete structures. This will lead to the improvement of accident simulation tools. Furthermore, the development of innovative mitigation and detection means has led us to propose:

- New types of composite ceramic SiC/SiC fuel cladding materials that produce low amounts of hydrogen serving as a replacement of the zirconium (Fig. 9);
- Monitoring means, analyzing the composition of the atmosphere inside the containment.

Working together in a collaborative effort, the CEA and the IRSN have developed numerical tools designed to simulate the different components involved in the hydrogen risk. The validation of these tools relies on the CEA and IRSN experimental facilities.

The MISTRA facility of the CEA at Saclay (Fig. 10) is a containment designed to further the development and validation of fluid mechanics models in a confined environment, in the case of a hydrogen release in various situations. At present, the experiments focus on the prevention means. The MISTRA facility, operating with helium to simulate hydrogen thereby permits researchers to:

- Take into account the different physical phenomena and their coupling;
- Carry out studies based on specific means of mitigating the hydrogen risk.

These characteristics correspond to a homothetic of a factor 10 in the containment of a 900 MWe PWR. With its condensers, MISTRA is the only experimental containment that can access the spatial distribution of the condensation, thereby enabling researchers to make a pertinent analysis of the hydrogen combustion modes.

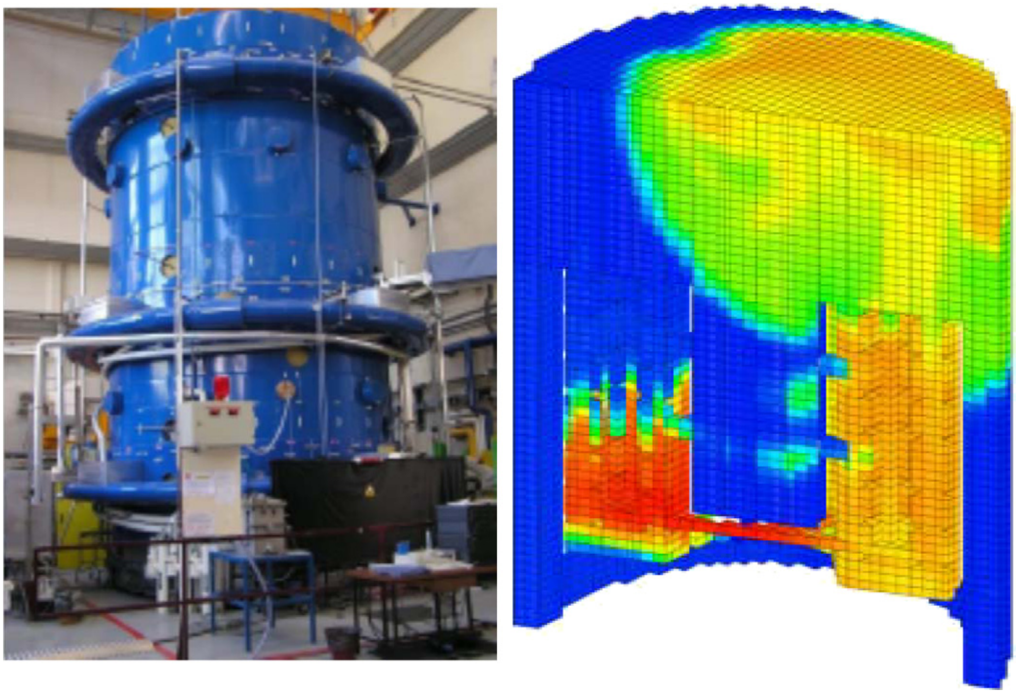

Fig. 10. The CEA MISTRA facility for the study of the hydrogen risk.

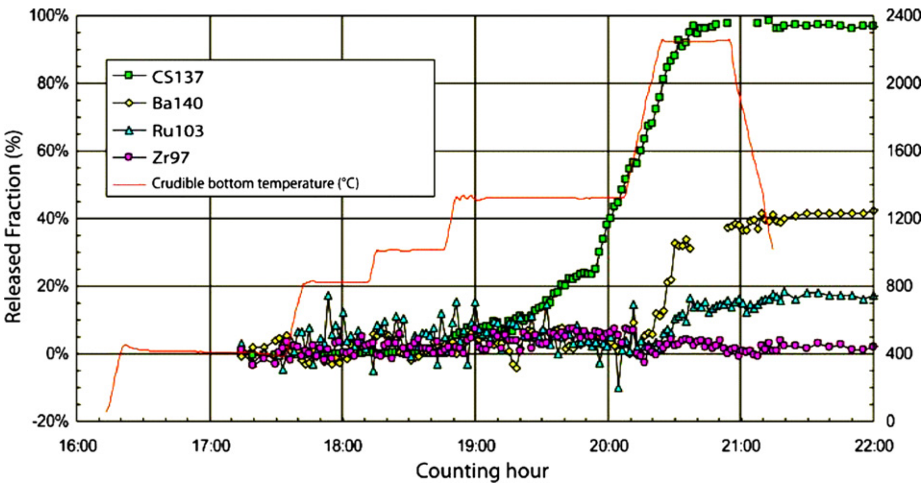

Fig. 11. The fission product families.

Two on-going international projects are underway on MISTRA. The numerical tools resulting from these experimentations have served and continue to serve in the elaboration of safety files pertaining to Basic Nuclear Facilities (reactors, hot labs, ...) and also to the storage and transport devices of radioactive matter.

2.2.3. The source term and the physico-chemistry of fission products (FP)

In the case of a break or partial or total meltdown of the fuel cladding in the reactor core, part of the radioactivity contained in the reactor core is likely to be transferred to the containment, or even into the environment in case of containment failure.

The main source of radioactivity is made up of fission products that can be classified into 4 families according to their degree of volatility. (See Fig. 11: Fission gas and volatile FPs such as iodine, caesium, antimony, tellurium, cadmium, rubidium and silver; the semi-volatile FPs that are: Molybdenum, rhodium, baryum, palladium and technetium; the slightly volatile FPs, namely ruthenium, cerium, strontium, yttrium, europium, niobium and lanthanum and the non-volatile FPs which are zirconium and neodymium.)

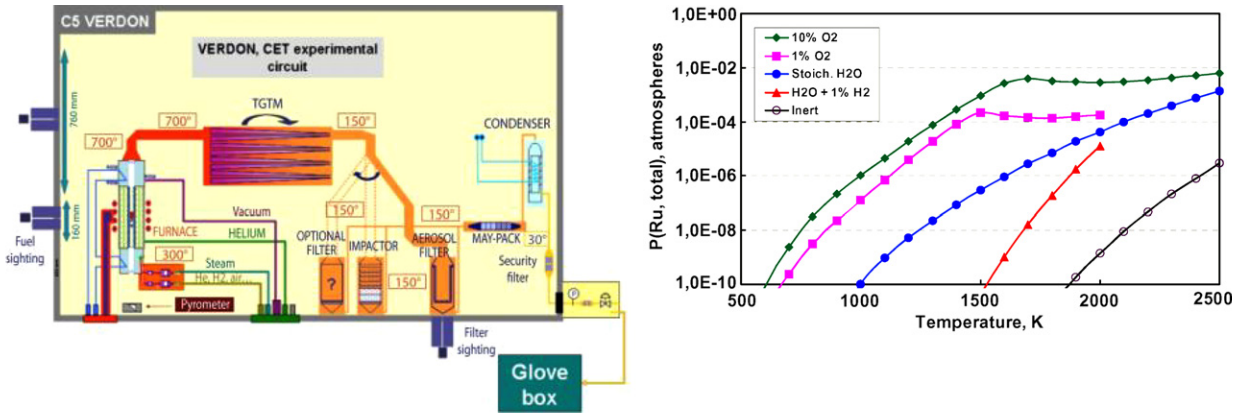

Fig. 12. The CEA VERDON facility designed for the study of fission product behaviour and release of ruthenium.

Major programs dealing with fission product behaviour during severe accidents have been carried out all over the world for many years (particularly the Phébus FP program done by the IRSN between 1993 and 2004. It consisted of integral tests using real materials and brought into play the totality of core degradation phenomena as well as the physico-chemical evolution of the fuel FPs inside the containment). Despite this fact, knowledge of the distribution of FPs and the associations between them has still proved to be insufficient in achieving validation of the models. During the period of 2011–2013, the program carried out by the CEA in an international framework at the VERDON facility in CEA-Cadarache (Fig. 11) will aim at eliminating uncertainties that remain concerning the release of FPs in certain types of fuels (UO_2 at high burn-up, MOX) and their transport in the primary circuit for scenarios involving air intake in the primary circuit. These results will complement the experimental basis of both national and international programs of the past.

The VERDON facility (Fig. 12) can take delivery and characterize the low power, re-irradiated reactor fuel samples coming from the experimental reactor, OSIRIS at CEA-Saclay in order to reconstitute its FP inventory in a short period and to heat the samples up to 2600 °C in an inductive oven under a controlled atmosphere that reproduces severe accident configurations thereby enabling us to study the fission product release and their transport in the primary circuit. The fission products are released from overheated fuel in a gaseous form. Afterwards, they are transported through the primary circuit under an atmosphere essentially composed of steam and hydrogen. Some of them condense on the walls of the primary circuit while others reach the containment either in an aerosol or gaseous form. It has been demonstrated, through the study of an accident scenario, that air intake in the vessel was possible after the re-localization of the corium at the bottom of the vessel and its failure. This dramatically modifies the environment of the degraded fuel and can lead to an accelerated release of ruthenium and to a re-volatilization of the volatile FPs already deposited on the walls of the primary circuit (Fig. 12). Particular interest has focused on the ruthenium due to its high radio toxicity when in volatile form.

3. External accidents: The seismic risk

In the field of “special risk” industries such as the nuclear industry, verification of building resistance to earthquakes is a regulatory requirement integrated into their initial design. Nevertheless, periodic safety re-evaluations lead to the development of analysis tools and methods that will enable us to determine in the most effective manner the design basis margins as well as the optimization of any possible need of reinforcement.

The R&D now underway involves comprehension and qualification tests for the structures and equipment at reference seismic levels allowing us to develop laws that govern the behaviour of the materials, to take into account the spatial incoherence in the seismic movement and to define equipment damage criteria. Theoretical research work and modelling have supplemented the tests in the justification of all structures under dynamic load. These studies are based on experimental means such as the TAMARIS platform which is made up of vibrating tables located at the CEA/Saclay site (Fig. 13).

The ambition now is to create a scientific cluster of knowledge and skills of European reference involving mastery of the seismic risk in order to work towards the development of tools and methodologies that will enable us to model the seism from the fault to the structure behaviour. This Institute, created by the CEA and the EdF will be asked to open its partnership not only to R&D organizations and universities but also to safety organizations and industrialists.

The primary focus of the research will be on:

1. Evaluation of the seismic hazard;
2. Study of the ground–structure interaction;
3. Behaviour of the structures and buildings;
4. Behaviour of the equipments;
5. Seismic insulation and reinforcement devices;

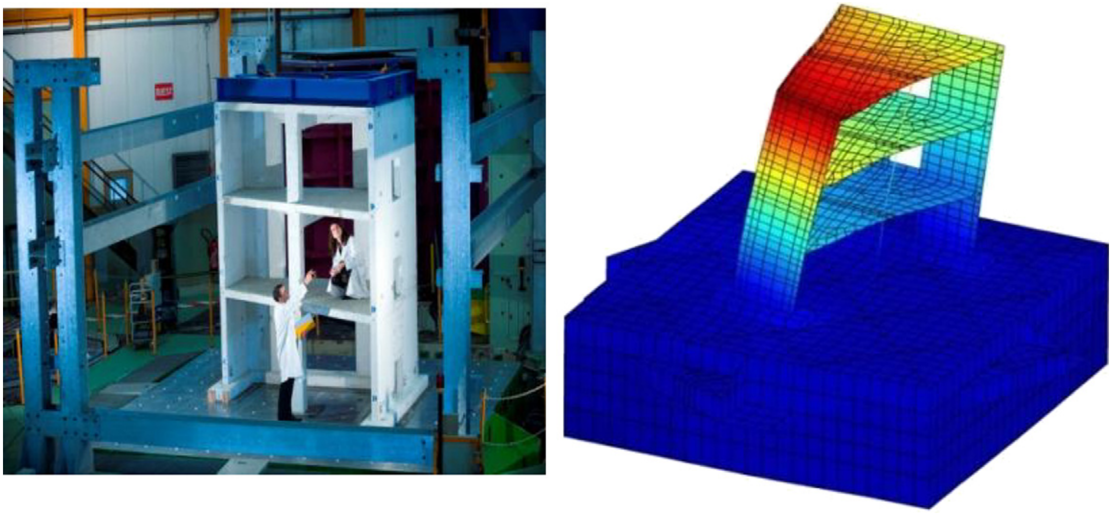

Fig. 13. Seismic behaviour of a nuclear building mockup on the TAMARIS platform and the modelling of the experiment.

6. Probabilistic methods;
7. Development of hybrid tests (tests of "under-structuration" between the numerical and physical structures).

These studies will be based on current experimental means as well as on their extension (especially, the EXtension TAMaris project) enabling us to test mockups close to scale 1 and to simulate the great displacements, the high accelerations as well as the behaviour of the multi-supported structures.

4. Conclusion

Nuclear safety research is carried out at the CEA for the benefit of the operators, industrialists and for its own facilities. The CEA also pursues this research to maintain and perfect its expertise so that it can place it at the disposal of the French government.

Nuclear safety research has three common points:

- A finalized research, relying heavily on research carried out "upstream" in university laboratories ranking highest in their field. Their results allow us on one hand to attain pertinent expertise and scientific backup and on the other to progress in reaching state of the art levels in the specific fields studied;
- Strong international partnerships;
- Experimental tools and software tools for the most part unique and joining together to create a synergy.

Several years of scientific and analytical work will be necessary before obtaining complete feedback on the Fukushima accident. For now, the initial elements allow us to confirm certain ways of improving work, mainly by acting as a technical and scientific backup to the industrialists and operators. The CEA had already been working on:

- The consolidation of knowledge about the sequence of events leading to core meltdown and the de-flooding of the pools: The issue now is to pursue the work already underway concerning the quantification of radioactive releases particularly in the long term and/or the presence of air;
- The acquisition of knowledge about the mechanisms and the procedures/devices that could stop the progression of the accident and limit the releases into the environment: This goal of this work is to reduce the gap between the reactor generations (cooling a degraded core in a vessel or after perforation of the vessel, trapping the radioactive products particularly the iodine and ruthenium, ...);
- Improvement of our knowledge about the hydrogen risk: Production (through oxidation of the cladding and by radiolysis), stratification model, mobilization through the re-combiners and explosion;
- Instrumentation on the follow-up of the accident;
- The pursuit of knowledge acquisition, following a logic of prevention in the field of fuel behaviour in accident situations to handle an evolution in a power excursion accident or a loss of coolant leading to an accident involving core meltdown.