Chapter 9
Research on Core Melt Accidents

When the nuclear power plants with pressurized water reactors (PWRs) and boiling water reactors (BWRs) were built, no specific design provisions were retained as regards core melt accidents. It was not until the 1990s, with reflection and research on "new generation" reactors (Generation III, e.g. EPR, AP1000, etc.), and following the Three Mile Island (TMI) and Chernobyl accidents, that core melt was taken into account in the design of third confinement barrier components[110]. The objective is that the human and environmental consequences of a core melt accident should only lead to counter-measures "limited in space and time" (no permanent relocation of communities, no permanent bans on the consumption of foodstuffs, etc.).

Nevertheless, from the first deployment of PWRs and BWRs, research on reactor core melt and its radiological consequences was begun, particularly in the United States. The aim was to assess the risk to the public from a major accident causing radioactive releases into the environment, at a nuclear power reactor of several hundred MWe, approximately 50 kilometers from a city (*Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants*, WASH-740 in 1957 and WASH-1400 in 1975 [1]). However, it was mainly the accident at the Three Mile Island power plant in 1979, and especially the discovery in 1985 that a large portion of the core had melted, that triggered a huge raft of research and development programs globally in this field.

After more than 30 years' research, vast progress has been made with understanding and modeling the complex phenomena that could occur during a core melt accident. Numerous programs run by the OECD/NEA (CSNI), the Phebus-FP (Fission Product)

110. Or components that could pose a risk to the containment, such as the main reactor coolant system.

program, the international ISTP program, and so on (we will come back to these) made major contributions to this progress. Models using the results of these programs have been integrated into codes for simulating these accidents and their impact on the reactor, e.g. the ASTEC [2] code, developed in France[111] by IRSN, in partnership until 2016 with its German counterpart GRS, and used in its safety assessments and to build the models used in level 2 probabilistic assessments[112]. Since 2004, under the European Commission's impetus, a research excellence network on severe accidents has been set up, known as SARNET [3, 4]. This network, directed by IRSN, brings together the work of around 250 researchers from 44 research organizations and laboratories in 23 countries (EU and non-EU). The network, which is now part of NUGENIA, provides a means of coordinating the researchers' work and pooling their efforts to establish the state of the art of knowledge, determining research priorities, setting up research projects and cooperating on the development and validation of accident scenario computer codes, particularly the ASTEC code (Figure 9.1), which has been made available to the network's members.

Since the early 2010s, as part of the safety reviews for the purposes of extending the service life of power reactors to 60 years and the complementary safety evaluation (ECS) conducted in the wake of the Fukushima Daiichi accident in Japan in March 2011, research in the field of severe accidents has focused more on studying physical measures to protect the containment and limit radioactive releases. These aspects are covered at national level by the ANR RSNR projects launched following the Fukushima Daiichi accident, at European level as part of the research and development framework program (FP7) and Horizon 2020 (H2020) and at international level in the context of a number of OECD projects. In particular, physical measures designed to contain the progression of core melt in the reactor vessel, keep the molten materials in the containment if it fails and reduce radioactive releases through filtration[113] are being examined.

We will not describe here the research and development being conducted to model the often complex thermohydraulic phenomena that lead to core uncovery[114], which were discussed in part in the previous section. We strongly recommend that readers wishing to understand in more detail the phenomenology of core melt accidents and the related research should refer to IRSN's publication *Nuclear power reactor core melt accidents – State of knowledge*[115] published in 2013, or to the reports by the OECD or the U.S.NRC on core melt accidents [5, 6].

111. Other codes have been developed abroad, including MAAP (Modular Accident Analysis Program) and MELCOR.
112. Level 2 PSAs are used to assess the nature and scale of radioactive releases outside the containment as a result of a core melt accident, with the corresponding frequencies, and contribute to assessing the overall safety of the facility. They make it possible to verify that the estimated frequencies of accidents that could lead to large releases to the environment are very low. See the document "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.
113. Where filtered venting of the containment is necessary to prevent its failure due to excessively high internal pressure.
114. The research and development on core melt described in this Chapter relates to an initiating event such as a loss-of-coolant accident, aggravated so that it leads to core melt.
115. Science and Technology Series – IRSN/EDP Sciences – 2013.

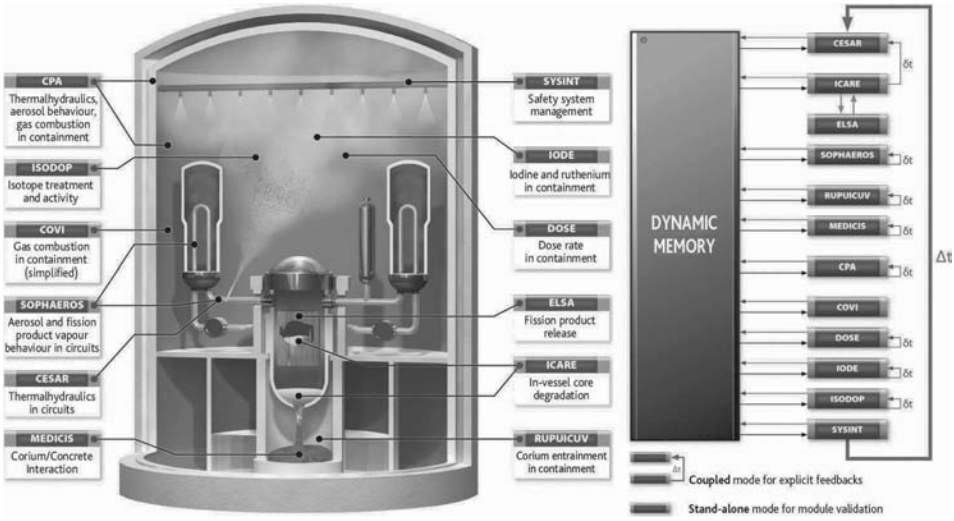

Figure 9.1 ASTEC code modules. © Didier Jacquemain/IRSN.

The different phases following core uncovery, if the accident is not brought under control, are:

- core heating and core melt inside the vessel,
- reactor vessel melt-through and erosion of the concrete basemat of the containment by the molten corium,
- loading of the containment by a sudden increase in internal pressure,
- loading of the containment by a slow increase in internal pressure,
- release of radioactive products outside the reactor (commonly referred to using the term "source term").

In the early 1980s, research focused mainly on studying the first phase.

9.1. *Heating of the core and core melt inside the vessel*

Many analytical tests were performed in the 1980s by different research laboratories, particularly in the United States and Germany, to study separately the various physical phenomena and find out the laws governing them.

Initially, the oxidation of the zirconium alloy cladding by steam was the subject of laboratory studies (see Section 3.2). This is an exothermic phenomenon that increases once the temperature has passed 1000 °C. The hydrogen produced spreads inside the vessel then in the reactor building if a break occurs in the reactor coolant system. The kinetics of cladding oxidation can be described by a parabolic law, expressing the fact that at a given temperature, the thickening of the oxide layer acts as a barrier to the transfer

of oxygen to the remaining metal, slowing down the oxidation. It greatly increases above 1580 °C as a result of a change in the zirconia's metallurgical phase, making it less resistant to oxygen diffusion within the metal itself. Correlations allow these phenomena to be taken into account with a good level of confidence.

The core of a water reactor is made up of many materials: several dozen metric tons of uranium oxide, plus around ten metric tons of zirconium, but also stainless steel, Inconel and materials designed to control the fission reaction. These are a silver, indium and cadmium alloy (AIC) in 900 MWe reactors, boron carbide (B_4C) and AIC in 1300 MWe and 1450 MWe reactors and in the EPR reactor – boron carbide only in BWRs. All these materials can interact with one another and form liquid phases at much lower temperatures than their respective melting points. Uranium oxide, which melts at 2840 °C, can be dissolved at much lower temperatures than its own melting point in alloys where it is combined with metals with lower melting points (such as the zirconium of fuel claddings).

Experiments on small samples of materials, particularly in Germany, the United States and Japan, have enabled these interactions to be observed. Phase and correlation diagrams have been drawn up as a result, describing the reaction kinetics. Between 1200 °C and 1400 °C, the materials absorbing the neutrons (AIC, B_4C) react chemically with their stainless steel cladding, forming a liquid mixture that in turn dissolves the zirconium. Above 1760 °C, the zirconium metal melts and dissolves the zirconia and the uranium oxide. Finally, the zirconia melts at 2715°C and also dissolves the uranium oxide. The mixtures obtained can also react with steam if they are not completely oxidized, producing hydrogen.

These phenomena of oxidation and chemical interaction between all the materials in a reactor core have been modeled, usually empirically, and the models have been integrated into simulation codes. In brief, the reactor core is split into meshes – representing at fuel rod scale the average behavior of a set of fuel assemblies over a given height – and for each mesh the code calculates the velocity and composition of the fluid flowing in the core, the temperature reached, the oxidized cladding thickness and the fraction of molten material and its composition. Above a certain fraction of molten material, trapped between the fuel pellets and cladding in the case of fuel rods or accumulated in a control rod, or if the remaining envelope of solid cladding is estimated not to be thick enough to contain the molten material, the liquid materials flow by gravity and solidify in the lower, cooler regions of the core. They then form metal plugs (mainly of AIC or a mixture of boron and steel) topped by ceramic plugs (Zr, U, O mixture).

Given the complexity of the physical and physico-chemical phenomena studied, the diversity of the materials present and the geometry of a reactor core, it was quickly realized that it was necessary to perform integral tests on a larger scale with components representative of a reactor core. The aim was to check that the interactions between the different phenomena would not produce effects that had not been predicted by the models stemming from the analytical tests.

Most of these tests were performed in experimental facilities, using a nuclear reactor to heat the fuel being studied. This experimental method, which is fairly difficult and very

expensive to carry out, has the advantage of keeping the fuel (and only the fuel) heated when it flows to the bottom of the experimental equipment once melted, as in a real accident.

In France, IPSN, and then IRSN, conducted two large-scale research programs in the PHEBUS experimental reactor described in Section 2.1: the Phebus-CSD[116] programs (on severely degraded fuel) from 1986 to 1989 and Phebus-FP (on fission products) from 1988 to 2012 (the last test was performed in 2004).

The **Phebus-CSD program** was used to study phenomena linked to zirconium oxidation by steam, the associated hydrogen production, physico-chemical interactions between the uranium oxide and zirconium, in a solid or molten state, and the effect on core degradation of AIC control rods. The program consisted of six tests: B9, B9R, B9+, C3, C3+ and AIC. The experimental devices each consisted of an assembly of 21 fuel rods that had not been irradiated, of the same type as those used in PWRs but shorter (a fissile length of 80 cm compared with 3.66 m in a real fuel rod), distributed in a square-pitch arrangement and held in place by two Inconel grids. The assembly was contained inside a steel tube, which in turn was inserted into the cell placed at the center of the reactor. A thick thermal screen made from porous zirconium was used to isolate the assembly from the steel tube, cooled by the circulation of pressurized cold water.

A mixture of steam, hydrogen and helium, the composition of which varied from one test to another depending on the test objectives, was injected at the foot of the assembly. The assembly was heated by gradually increasing the power of the PHEBUS reactor. A borated steel neutron screen placed between the PHEBUS reactor core and the test device was used to obtain an axial distribution of uniform power as in a power reactor core.

Each device was fitted with a very large amount of instrumentation for measuring all the physical parameters essential for interpreting the tests, particularly high temperature thermocouples (tungsten-rhenium, W/Re), which functioned at up to 2100 °C, and a mass spectrometer for measuring the amount of hydrogen being produced at every moment. An array of post-test examinations, both non-destructive (X-rays, gamma scanning[117]) and destructive (photo macrography, analyses with a scanning electron microscope), complemented the measurements taken during the experiments, so that the damage to the fuel rods could be quantified and the composition of the resolidified mixtures could be determined.

The six tests carried out produced many experimental results, which were used for developing and validating the core melt models integrated into the ASTEC simulation code. In particular, they made it possible to observe how the oxidation of the zirconium alloy cladding spreads axially with the temperature runaway above 1550 °C. The progression of the oxidation front was characterized by the occurrence of steam

116. Severely Degraded Fuel.

117. Device for measuring the intensity of the gamma radiation emitted by a source depending on its energy levels. Following calibration, analysis of the spectrum obtained can identify the radionuclide(s) responsible for the radiation and their masses.

starvation, indicating that all the steam available to the cladding had been absorbed by the cladding to oxidize the metal (the gas flowing beyond the front did not contain any more steam, only hydrogen). Very significant results were obtained for interactions between the core materials and the processes causing core melt (liquefaction of the zirconium on contact with the Inconel of the grids, dissolution of the outer layer of zirconia and the fuel inside the cladding by the molten zirconium, liquefaction of the control rods at around 1400 °C and damage to the adjacent fuel rods).

The **Phebus-FP program** [7] helped to reduce the uncertainty concerning the estimation of radioactive releases of a core melt accident at a light water reactor, and to increase IRSN's assessment and emergency response expertise in this field. The program consisted of five tests during which the main physical phenomena governing core melt, the transfer of fission products from the fuel to the reactor building, and their behavior inside it were studied. It was conducted in collaboration with a number of French and foreign partners (EDF, the European Commission and its Member States, the United States, Canada, Japan, South Korea and Switzerland). The definition of the tests, engineering and experiments, and the analysis and interpretation of the experimental results involved around 80 people over a period of approximately 15 years.

The program required major modifications to the PHEBUS facility. The reactor building was strengthened against earthquakes and enlarged to accommodate a sealed steel box containing a model of the reactor coolant system and containment of a PWR and all the necessary equipment and instrumentation (Figure 9.2). The experimental circuit reproduced, on a 1/5000 scale, the three vital components for this type of study of a 900 MWe PWR: the core, the reactor coolant system and the containment.

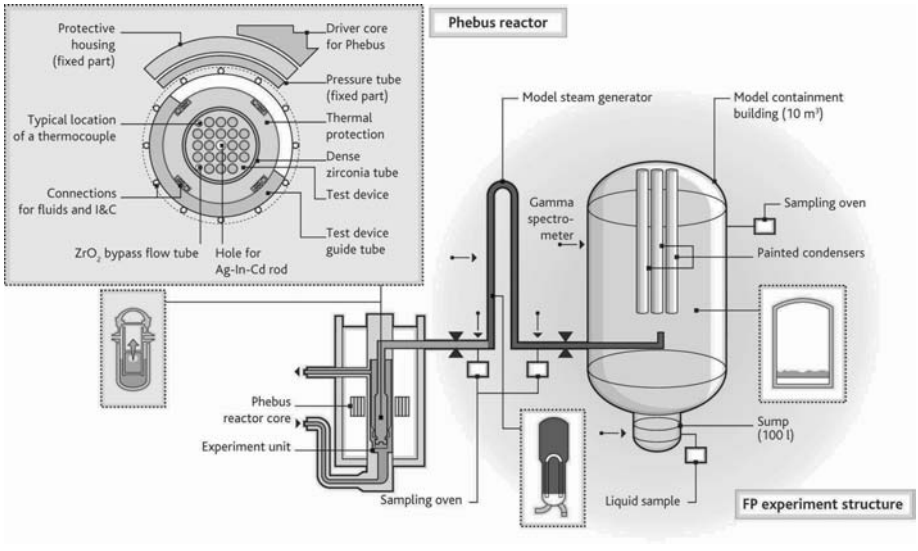

Figure 9.2 Schematic view of the experimental circuit used for the Phebus-FP program. © IRSN.

To represent the core, a similar device to the one designed for the Phebus-CSD program was used. This was an assembly consisting of 20 fuel rods with a fissile height of 1 m, and a rod in the center representative of the control rods used in power reactors, all arranged in the cell at the center of the PHEBUS reactor. The fuel used in the first test had not been irradiated before being used in the PHEBUS reactor, whereas the fuel used in subsequent tests came from the Belgian BR3[118] reactor and an EDF reactor; their burnup rates were between 23 and 38 GWd/tU. The instrumentation had been significantly improved, in particular with the use of ultrasonic thermometers that could measure temperatures of up to 2900 °C.

The reactor coolant system model consisted of a section cooled to 150 °C simulating a steam generator tube with a break at the outlet from this tube. The fluid leaking from the break was collected in a 10 m^3 tank fitted with a sump at its base, simulating a reactor containment. The external surface of the tank was thermally isolated and there were cooled cylinders at its center as a scale representation of the containment wall of a PWR on which the steam could condense. Some of the cylinders and a plate immersed in the sump were coated with identical paint to the paint used in the PWRs in France's nuclear power plant fleet.

The facility's strength was its instrumentation. As well as measuring flow rate, pressure and temperature at several points in the circuit, it could also measure hydrogen concentrations in the coolant system and the containment. It also included gamma-ray spectrometers targeted at the tube intake, several points along the wall and the outlet from the tube simulating the steam generator, the atmosphere and the walls of the tank simulating the containment, as well as the sump, to continuously identify and measure the different γ-emitting fission products in suspension in these areas or deposited on the walls. A Maypack composed of selective filters, monitored continuously during the tests by a gamma-ray spectrometer, was used to measure the concentrations of the different forms of iodine in the containment atmosphere (aerosols or gases, and among the gaseous species, molecular iodine or organic iodine). There was also an optical device for continuously measuring the aerosol concentrations in suspension in the containment.

Branch pipes at the intake and outlet of the tube simulating the steam generator, in the sump and on the containment wall, meant that samples could be taken at various points during the tests through filters, impactors (instruments for measuring the size and concentration of the aerosols), thermal-gradient tubes (instruments used for working out the chemical forms of different species in vapor form by recording the temperatures at which they condense on the walls) and in capsules. In the tanks simulating the containment, coupons collecting the aerosols deposited by settling were also exposed to the containment atmosphere for predetermined time intervals. After the tests, radiochemical analyses were performed on all the samples taken, allowing identification of the fission products and the different actinides carried or deposited, and enabling their masses to be calculated precisely.

118. Belgian Reactor 3.

Specific non-destructive examination techniques using transmission and emission tomography[119] were also developed for the program and provided valuable information about the state of degradation of the fuel rods after the tests, and about the quantities of some radioactive products emitted by those rods.

The international Phebus-FP program, which ran from 1988 to 2012, consisted of five tests. The main parameters studied in these tests are summarized in the table below.

	Fuel type	Material simulating the control rods	Steam flow rate (g/s)	pH of the sump water	Sump/atmosphere temperature difference[120] (°C)
FPT[121]0	Very lightly irradiated	AIC	0.5–3	5	–18/–37
FPT1	~20 GWd/tU	AIC	0.5–2.2	5	–18/–38
FPT2	Ditto	AIC	0.5 (+ boric acid)	9	–18/+11
FPT3	Ditto	B_4C	0.5	5	–18/–3
FPT4	Debris bed of UO_2 at 38 GWd/tU and of ZrO_2	None	0.2–0.5 H_2O 0–0.25 H_2	N/A	N/A

Apart from test FPT4, each test consisted of three successive phases:

- the first phase of re-irradiation of the fuel, during which the PHEBUS reactor was operating at full power for approximately a week. The fuel being studied was cooled by pressurized water circulation. The aim was to recreate in the fuel the short-lived fission products that had disappeared since the fuel was unloaded from its original reactor, such as iodine-131 (half-life of 8 days[122]), which was essential for studying the releases;
- a "degradation" phase lasting a few hours, during which, by gradually increasing the PHEBUS reactor power, the temperature of the uncooled test fuel increased until the materials liquefied (at between 2300 °C and 2500 °C), causing the fission products to be released and transported into the circuit and then into the tanks simulating the containment. At the end of this phase, the PHEBUS reactor was shut down;

119. Technique for reconstructing the volume of an object from a series of radiographic measurements (transmission) or gamma-ray spectrometry measurements (emission) performed on each unit from the outside.
120. On the left of "/", temperature difference during the degradation phase; on the right of "/", temperature difference during the long-term study of iodine behavior. A positive difference means that water is evaporating from the sump surface, which is likely to encourage the transfer of the volatile iodine produced in the sump water to the containment atmosphere.
121. Fission Product Test.
122. The concentration decreases by half during the half-life.

– a "containment" phase, lasting for several days during which the variables of interest for understanding the deposition of radioactive products and the iodine chemistry in the tank simulating the containment were measured.

The first group of three tests (FPT0, FPT1 and FPT2) was used to study the effect of the burnup rate and the oxygen potential (atmosphere rich in steam or conversely in hydrogen, surrounding the fuel rods) on oxidation and damage to the fuel rods, and on the release and transporting of fission products in the circuit. These three tests were performed with an AIC fuel rod. Boric acid was also injected with the steam during test FPT2 (during normal reactor operation the boric acid serves to control its reactivity; in an accident situation, the boric acid can react with some fission products such as cesium, and significantly alter the chemical balance with the different iodine species, encouraging the formation of the most volatile).

Test FPT3 was used to study the influence that the presence of boron carbide control rods has on core degradation and fission product behavior. This type of material is used in EDF's most recent PWRs (1300 MWe, 1450 MWe and EPR) in addition to AIC, as well as in BWRs.

Test FPT4 focused on the advanced phase of an accident, studying the release of low-volatile fission products and transuranium elements from a debris bed of fuel and oxidized cladding brought to melting point. It consisted of a single temperature-increase phase until a molten material pool was obtained. The fission products and actinides emitted were sampled through a bank of filters situated above the debris bed and activated in succession.

Tests FPT0, FPT1, FPT2 and FPT3 were used to reproduce fuel rod damage states that had never yet been achieved experimentally, with significant melting of the fuel (by up to 50% of its mass) and the formation of corium pools, as illustrated in Figure 9.3. The numerous results obtained contributed significantly to improving knowledge of core damage mechanisms and substantially extended the scope of simulation code validation. For example, the cladding dislocation criterion was revised to take account empirically of the mechanisms causing the zirconia outer layer of the cladding to fail, releasing the molten zirconium inside and causing oxidation and hydrogen production to stop locally. The tests also significantly improved knowledge of the effects of physico-chemical interaction between the fuel, the zirconium and the materials forming constituting the control rods, leading to liquefaction of the fuel at around 2200 °C, a much lower temperature than predicted by models. The core degradation models in simulation codes such as ASTEC have benefited extensively from these new results and their prediction capabilities have been greatly improved by them.

The originality and quality of the tests performed at the PHEBUS facility was such that the Phebus-CSD B9+ and Phebus-FP FPT1 tests were selected by the OECD/NEA for the performance of international exercises to compare core melt codes (ISP 28 and ISP 46).

Abroad, large-scale programs have also been run, in chronological order:

– the SFD (Severe Fuel Damage) program, consisting of four tests conducted between 1982 and 1985 in the PBF (Power Burst Facility) reactor of the Idaho National Laboratory in the United States;

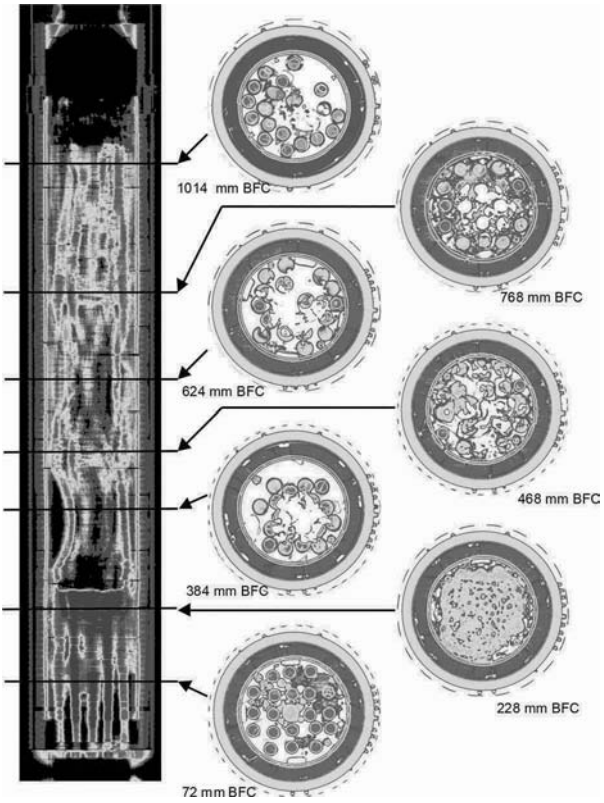

Figure 9.3 State of degradation of a fuel assembly after test FPT1; post-test non-destructive examinations of the tested fuel cluster: X-ray (left) and X-ray transmission tomography (right). (BFC: Bottom of Fissile Column). © IRSN.

- the DF (Damaged Fuel) program, consisting of four tests conducted in 1984 in the ACRR (Annular Core Research Reactor) reactor of the Sandia National Laboratory in the United States;
- the OECD/NEA LOFT-FP (Loss of Fluid Tests–Fission Product) project, consisting of two tests conducted from 1984 to 1985 in the LOFT reactor of the Idaho National Laboratory in the United States;
- the FLHT (Full Length High Temperature) program, consisting of four tests conducted between 1985 and 1987 in the NRU (National Research Universal) reactor of the Chalk River Laboratories in Canada;
- the CORA program, consisting of around 20 tests conducted between 1987 and 1997 at a non-nuclear facility of the German research center in Karlsruhe.

For experiments performed in nuclear reactors, the experimental devices consisted of assemblies of between ten (DF) and several tens (PBF) of fuel rods (or indeed 100 in the

case of the LOFT-FP project) of similar manufacture to the rods used in power reactors. They were generally shorter, like those used in the Phebus-CSD and FP programs, because the cores of the research reactors hosting the experimental devices were not as high, with the exception of the FLHT program which, as the name suggests, used full length rods with a fissile height of 3.66 m in the large NRU reactor. In some tests, rods containing AIC or boron carbide were included in the assembly, as well as fuel rods that had previously been irradiated for a significant period to obtain representative burnup rates.

In the CORA program tests, the fuel rods were simulated by zirconium alloy cladding filled with alumina and heated internally using a tungsten core with an electrical current running through it. The lower cost of these tests meant that many parameters could be studied: temperature increase rate, flow rate of the injected steam, ambient pressure, cladding materials, effect of the presence of AIC or boron carbide, geometry of the different reactors (PWR, BWR, VVER), effect of quenching at the end of the test, etc. However, this heating method meant that it was not possible to study the corium pool formation phase.

All the tests contributed to a better knowledge of cladding oxidation, hydrogen production and core melt phenomena. They delivered a wealth of experimental results covering a broad spectrum of situations, which are extremely useful for developing models and validating simulation codes such as the integrated ASTEC, MAAP and MELCOR codes.

In 2016, as shown by a benchmark exercise run by the OECD/NEA for an accident scenario similar to TMI-2 [8], the integrated codes for core melt accidents are able to simulate with a good degree of precision the progression of PWR core degradation up to the reflooding phase. On the other hand, a benchmark exercise run by the OECD/NEA for the damaged BWR reactors at the Fukushima Daiichi plant [9] identified major discrepancies in the calculation of the progression of core degradation in the three damaged reactors and of hydrogen production, revealing current uncertainties as regards representing degraded configurations for BWRs. Partly these difficulties are due to the fact that core melt accident experiments have focused mainly on PWR fuel assembly configurations. There has been very little study of BWR fuel assembly configurations; major uncertainties exist with regard to molten mixture relocation phenomena in the presence of Zircaloy channels for water in the assemblies and the cruciform B_4C-steel control rod plates specific to BWRs. More detailed analyses are under way in order to understand the differences between these very different configurations.

Interestingly, none of the tests conducted abroad reached the temperature or fuel degradation levels observed in the Phebus-FP tests, making them particularly valuable. On the other hand, some of the tests performed abroad (LOFT, CORA and QUENCH programs – QUENCH succeeded CORA at the same facility from 1997) have been able to study the effects on cores molten to varying degrees, of reflooding with large flows of water or steam.

The most important finding from these tests is that reflooding a core that has already been extensively damaged may, at least initially, considerably accelerate cladding oxidation, hydrogen production and core melt. When the accident in Unit 2 at the

Three Mile Island power plant (TMI-2) happened, there is no doubt that water entering the molten core caused any cladding that had not yet melted to rupture and the debris bed observed in the upper part of the core to form. Experimental results tend to show that core reflooding can halt the progression of an accident only if it happens very early on, i.e. when the maximum temperature in the core has not yet exceeded 1800 °C, and if the flow of water is sufficient (more than 2 g/s per rod). Despite all the experimental results obtained, models are unable at present to reliably predict whether or not reflooding a damaged core can cause runaway oxidation of the cladding and aggravate the situation.

To find out more about this phenomenon, since 2010 IRSN has been conducting test programs with the PRELUDE[123] and PEARL facilities at its laboratories in Cadarache, involving reflooding experiments on debris beds consisting of steel balls heated by induction inside a cylindrical cavity (diameter 11 to 29 cm in the case of PRELUDE and diameter 50 cm in the case of PEARL). The PEARL tests performed in a large-diameter section were used to observe the two-dimensional flows. An experimental and theoretical research program on the reflooding of solid debris beds in more complex configurations, in particular with the observation of oxidation phenomena, is planned for 2018.

If core reflooding is ineffective, the accident continues and the molten materials flow to the lower head of the reactor vessel, where they accumulate, as seen in the case of the TMI-2 accident. If these materials come into contact with water at the lower head, vast quantities of steam and hydrogen are produced. Despite the many examinations carried out after the TMI-2 reactor accident, particularly during the OECD/NEA TMI-VIP program, and the tests of dropping corium into water performed on a large scale in the FARO facility at the European Ispra Joint Research Centre in the 1990s, the flow mechanisms of the corium at the lower head and its mixture with the water present are still not clearly understood. This difficulty is overcome by using pessimistic hypotheses for the quantity of materials transferred, their fragmentation on contact with water, the oxidation reactions, etc. when evaluating the progression of a core melt accident. Nevertheless, all the experts agree that a steam explosion violent enough to destroy the reactor vessel could not occur under these conditions (OECD/NEA project entitled SERENA[124], described further on).

If the corium debris at the lower head is not cooled effectively, it will melt again, forming a pool in contact with the vessel walls, threatening the vessel's integrity. Various test programs have been undertaken to study this configuration. These test programs either use simulator materials, like the BALI tests conducted by CEA in Grenoble, at full scale in 2D geometry with water, to study natural convection in the pool, or they use representative materials (UO_2, zirconium alloy, steel), like the OECD/NEA test programs RASPLAV (1994–2000), MASCA (2000–2003) and MASCA 2 (2003–2006) conducted in Russia. During the latter test programs, corium (a mixture of UO_2-ZrO_2-Zr and steel) of various compositions and in different atmospheres (inert or oxidizing) was brought to

123. Preliminary Study on the Experimental Reflooding of a Debris Bed.
124. Steam Explosion REsolution for Nuclear Applications.

melting point and kept between 2200 °C and 2650 °C. In RASPLAV, a 200 kg mass of corium, with mass composition of 76.6% UO_2, 14.3% Zr and 9.1% ZrO_2, heated by tungsten walls forming a thin semi-circular section of a reactor vessel, was used. In the smaller scale tests using corium in crucibles cooled on the outside, heating was achieved by electrical induction and the more varied corium compositions studied meant that coriums with a high metal content could be used and in some cases simulators of FPs (a stable species of the radioelements of interest in order to study their distribution between the various metallurgical phases of the corium). In parallel, smaller scale tests were conducted with coriums of identical composition, allowing the physical properties of the material mixtures to be determined. Tests using pools of molten salt, eutectic or binary mixtures, completed the studies of the effect that the material properties have on the interface temperatures and their impact in correlations for predicting the distribution of flows of heat extracted from a pool to the walls of the vessel containing it. The RASPLAV tests confirmed that using traditional correlations for the distribution of heat from a pool at its edges (established using various simulator liquids that are easier to manage) with the physical properties of the corium was entirely justified.

The experiments performed in MASCA and MASCA 2 reveled complex phenomena of interaction between materials and of stratification between the metal and oxide phases in the presence of steel. Under certain conditions (particularly where the pools initially contain a significant amount – around 50% – of unoxidized zirconium), a three-layer configuration was observed with a high density metal layer containing uranium, topped by a layer of refractory oxides (mainly uranium and zirconium oxides), itself topped by a low density metal layer (iron, chromium, nickel, zirconium, etc.). In the stratified pool configurations where there is no reflooding, a very intense flow of heat can be transmitted to the walls around the upper metal pool (heat flow concentration or focusing effect), if its thickness is below a threshold of around 50 cm. The tests also revealed the transient nature of the stratification phenomena where steam is present, since the metal layers oxidize and gradually mix with the oxide pool.

These phenomena are modeled in the ASTEC simulation code but many uncertainties remain concerning the transient phenomena, particularly oxidation of the uppermost layer of metal and the transfer of oxygen between the layers when steam is present. To reduce these uncertainties, an experimental program known as CORDEB (Corium-Debris) was launched in 2012 by the Aleksandrov Scientific Research Technological Institute (NITI) in St Petersburg. It should run for at least seven years in the MASCA facility, in partnership with EDF, AREVA, CEA, IRSN. Examining the corium pools produced during the Fukushima Daiichi accident could shed further light, of great relevance, on the subject.

9.2. Reactor vessel melt-through and basemat erosion by the molten corium

The OECD/NEA TMI-VP project to examine in detail the damaged TMI-2 reactor vessel and its contents, mentioned above, established that the internal face of the lower head, lined with stainless steel, had locally reached a temperature of 1100 °C for

30 minutes, while the pressure inside the vessel was 10 MPa. The cooling of the corium once water circulation had been restored in the reactor coolant system prevented reactor vessel melt-through, though in 2016 the reasons for this are still not clear. Once of the hypotheses advanced to explain why the reactor vessel remained intact was that direct contact between the corium and the lower head was limited by the presence of water at the lower head when the corium flowed in.

The OECD/NEA experimental project known as OLHF[125], conducted by the Sandia National Laboratories in the United States with the involvement of CEA and IRSN, obtained experimental results on a 1/5 scale model for the creep-induced failure mechanisms of a PWR lower head at a representative range of temperature/pressure parameters. However, there are uncertainties involved in applying the results to a full-scale reactor, since a number of welded tubes pass through the lower head. A test conducted with these tubes showed a significant reduction in the rupture elongation of the vessel.

Following vessel melt-through, the molten corium runs into the reactor pit and, if there is no proper cooling, causes the chemical decomposition due to heat, and therefore erosion, of the concrete reactor pit walls and basemat. This could ultimately put the integrity of the containment at risk and lead to massive radioactive gaseous and liquid releases into the environment. The interaction between the corium and the concrete also generates large amounts of gas (steam, which is reduced to hydrogen and carbon monoxide and dioxide due to oxidation of the rebars or the metal phases present in the corium), which can increase the pressure inside the containment and also threaten its integrity.

During the erosion of the concrete by the corium, which can last for several days, the composition of the pool changes. It contains some dense oxides (mainly U-Zr-Fe-O_x), low-density oxides (SiO_2, CaO) and non-oxidized metals (Fe, Cr, Ni, Zr). These are gradually oxidized by the steam or carbon dioxide whereas they can only reduce much more marginally other less stable oxides. The pool solidifies on contact with the cooler concrete, temporarily forming crusts that are mechanically unstable since the underlying concrete wall is breaking down. Stratification between the metal and oxide phases can also occur, at least temporarily, changing the distribution of the heat flows. Whether this phenomenon occurs depends on the configuration of the pool within the vessel when the vessel fails (if it is stratified or evenly mixed) and the position of the break (e.g. if it is level with a metal layer). In the longer term, it depends on how the composition of the metal phase changes, the density ratio between metal and oxide phases, the flow rate of the gases generated by decomposition of the concrete, and the residual quantity of unoxidized metal; all these parameters changing during the interaction. Incorporation of the metals from molten rebars in the concrete can contribute to the formation of a metal layer.

125. OECD Lower Head Failure.

In France, these phenomena were first studied experimentally in small-scale analytical tests using simulator materials with low melting points. The tests were conducted by CEA on behalf of IPSN (then IRSN) and EDF:

- the BALISE program studied liquid entrainment at the interface between two non-miscible liquids with different densities and viscosities (water and liquid paraffin, silicon oils, etc.), percolated through a nitrogen flow (1999–2000);
- the ABI program measured the heat transfer coefficients at the interface between two non-miscible liquids with very different densities (Wood metal and gallium to simulate the corium, water and oils of different viscosities to simulate the concrete), percolated through a nitrogen and argon flow (2007–2008);
- the ARTEMIS program simulated the erosion phenomena in one-dimensional then two-dimensional geometry using a salt ($BaCl_2$) to simulate the corium and a binary eutectic mixture (LiCl-$BaCl_2$) to simulate the concrete (2003–2008);
- the CLARA program measured heat transfers in a pool of water in the presence of a flow of gas injected through the walls using water and additives to vary the viscosity (2008–2012).

Given the importance of the combination of convection and thermal and physico-chemical phenomena, tests with real materials were necessary. At Cadarache CEA built the VULCANO[126] facility, commissioned in 1997[127], which uses a rotary plasma arc furnace to melt prototypic corium mixtures (a UO_2 and ZrO_2 mixture, also incorporating oxides from the decomposition of the concrete to study the later phases of the interaction) with a mass of around 40 kg, which are then poured into an experimental section. For the programs conducted from 2003 with the participation of EDF and IRSN, to study interaction between the corium and concrete (CCI[128]), corium pools were poured into cylindrical concrete crucibles of varying compositions. High frequency induction heating was used to reproduce the residual heat released in the fuel throughout the interaction. The instrumentation in place was used to determine the axial and lateral progression of the concrete ablation front.

One specific characteristic of the facility is that it can also be used to study the erosion caused by metal/oxide pools and the impact of stratification. A mass of steel of approximately 25 kg representing the molten structural materials was melted in three separate induction furnaces then poured into the experimental section containing the molten corium. In the test section, a system of magnetic screens meant that it was possible to heat only at the level of the oxide phase of a stratified mixture.

126. Versatile UO_2 Laboratory for COrium ANalysis and Observation.
127. The facility was initially used to study corium spreading in the core catcher installed under the EPR reactor vessel. The tests performed there were also used to validate the CROCO corium spreading simulation code developed by IRSN.
128. Corium-Concrete Interaction.

The following tests were devoted to studying the interaction between corium and concrete:

- nine tests with all-oxide corium (VB-U series),
- four tests with corium and steel (VBS-U series),
- four tests that were more analytical (VBES-U series).

The results revealed the importance of the type of concrete (siliceous or limestone) to the concrete's ablation profile, confirming the results of tests carried out by the Americans (see further on). They also contributed to the development of models for evaluating a conservative approach taking account of the uncertainties that still exist concerning the time to penetration of the concrete containment basemats of French reactors if a corium/concrete interaction were to occur. It has therefore been possible to demonstrate that basemat penetration should not occur within 24 hours of the start of an accident[129], allowing time for measures to be taken to protect the public.

In Germany, tests were performed in the 1990s at the BETA facility in the Karlsruhe research center. The corium was simulated by a molten mixture of 400 kg of alumina and iron obtained through an exothermic oxidation/reduction reaction, to which several dozen kilograms of zirconium were added. This mixture was poured into a concrete crucible.

In the United States, three research programs were conducted at facilities of the Argonne National Laboratory: the MACE (Melt Attack and Coolability Experiments, 1989–2000) programs on behalf of EPRI (Electric Power Research Institute) and the OECD/NEA MCCI[130] programs (2006–2010) in which France (IRSN, EDF and CEA) was a partner. The objective of these two programs was to conduct large-scale and smaller-scale integral tests using representative materials, to study:

- concrete erosion phenomena in a two-dimensional geometry in the absence of any cooling;
- the various cooling mechanisms identified during the MACE experiments when water was poured on the surface of the corium;
- new technological concepts for cooling the corium and halting the concrete erosion.

Two experimental devices were used, SSWICS (Small Scale Water Ingression and Crust Strength) and CCI (Corium Concrete Interaction). In the first, a mass of around 75 kg of UO_2 and ZrO_2, mixed with siliceous or limestone concrete, brought to a temperature of more than 2000 °C by an exothermic oxidation/reduction reaction, was melted in a cylinder of diameter 30 cm with inert walls made from magnesium oxide. Water was poured on the corium to study the cooling mechanisms and their effectiveness, in particular the mechanism of water penetration into a top crust that has cracked

129. In the case of Fessenheim, after basemat thickening.
130. Molten Core-Concrete Interaction.

due to thermomechanical stresses. Gas could also be injected through the bottom to simulate the gases released during the corium-concrete interaction.

In the second device, a mixture of 400 to 590 kg of UO_2 and ZrO_2 was melted using the same method in a larger cavity with a square cross-section of 50 cm by 50 cm (one test was performed in a 50 cm by 70 cm cavity with a single ablatable side wall to study any effects of scale). Tungsten electrodes were arranged on two opposite sides of the cavity so that an electrical current could be applied to the corium pool to generate a release of heat representative of the residual heat. The other two sides and the bottom were made from concrete; the type of concrete (siliceous or limestone) was one of the parameters of the study. On this test device, the "Joule effect" heating technique made it impossible to study stratified pools with two separate phases, oxide and metal, because the stratification of the metal would cause an electrical short circuit between the electrodes. Following a dry corium-concrete interaction phase to study the concrete ablation mechanisms, water was added at the top to study the cooling mechanisms.

In all, 21 tests were carried out during the two OECD/NEA MCCI programs. They provided important results that significantly improved understanding and modeling of concrete ablation and corium pool cooling mechanisms in a corium-concrete interaction. As far as concrete erosion kinetics in the absence of cooling are concerned, it was observed that the ratio between the final erosion thicknesses in the axial and lateral directions depends on the type of concrete. Whereas the erosion is almost uniform in the case of limestone concrete, it can be three times greater laterally in the case of siliceous concrete. As regards cooling mechanisms, it was shown that cracking of the upper crust and water penetration is much more effective when the cooling above the pool starts early, i.e. while the concrete content inside the pool is still low and the crust is more fragile. Eruption and fragmentation of the corium ejected into the water through the crust covering the pool were also observed. They help to cool the pool by turning a compact mass into a bed of debris several millimeters in diameter. This is even more effective when the concrete is a limestone type (more gas is released ejecting more corium through the cracks in the crust). At the end of phase 2 of the MCCI program, a test with siliceous concrete but with cooling at the start of the corium-concrete interaction meant that it was possible to observe ejections with this type of concrete for the first time. This encouraging result prompted EDF to continue funding tests (CCI), as part of its work program on extending the operating life of France's nuclear power plants, to study cooling mechanisms where water is injected early on in the corium-concrete interaction at the surface of the pool, with support from IRSN, the U.S.NRC, NRA and CEA.

Experiments have also been conducted in Germany to test a different corium cooling concept (COMET) by injecting water through a special device consisting of several porous concrete layers. This device could prove more effective than pouring water on the corium because then the liquid and steam are not circulating against the flow. These experiments were conducted at the Karlsruhe center, using up to 1300 kg of alumina and iron melted using an exothermic oxidation-reduction reaction and kept at temperature by induction heating. Following the erosion of a sacrificial layer of concrete, the pool is cooled by water injection *via* a layer of highly porous concrete through porous concrete blocks, the section of which was optimized to maximize cooling efficiency while avoiding a too fast

production of steam, which would increase pressure inside the containment. The layer of highly porous concrete through which the water was injected on to the blocks was linked to a tank on a higher level, so water was flowing by gravity only. During the OECD/NEA MCCI-2 test program and specifically a VULCANO VWU[131]-1 test, cooling with a device following the same principle was also studied using prototypic materials. The results led to the conclusion that this type of cooling system was effective, meaning that it could be implemented in new reactor concepts.

Knowledge of concrete erosion and corium cooling phenomena has advanced significantly thanks to all the experimental results produced in the last 20 years. More robust models based on a large database of experimental data have been integrated into simulation codes, particularly the ASTEC code. However, it is still not possible to model in any detail the phenomena occurring at the interfaces between the corium and concrete and therefore to understand from a mechanistic point of view the differences between erosion kinetics due to the concrete type. IRSN, in partnership with universities and the CNRS, has tried to use direct simulation techniques to advance this area of knowledge, e.g. by studying the material flows at the corium-concrete interface or the consequences of bubbles passing through an interface between two non-miscible liquids, in terms of the mass of liquid entrained and the effects on heat transfer through convection. Without any local information for validation purposes, these efforts have not delivered any very significant advances for the erosion models and so the current models are still very empirical.

Moreover, without sufficient experimental data, uncertainties remain about the possible effects of a stratified pool. Two mechanisms could limit the life of such a pool: oxidation of the metal phase and mixture of the materials due to the gases released by decomposition of the concrete. Further tests are therefore necessary to study these phenomena and reduce the uncertainty. The impact of the rebars on the erosion mechanisms is also unresolved because results from the MOCKA tests conducted at Karlsruhe in the last few years, with simulator materials at high temperatures, are fairly contradictory and to date no equivalent tests have been performed with prototypic materials.

Nationally, research is also continuing to study the possibility of cooling and solidifying the corium by covering it with water or injecting water into a COMET-type device, with a view to designing systems that would avoid basemat penetration in existing reactors. They are being carried out in France as part of the MIT3BAR[132] project in response to the call for RSNR projects launched in 2012 by ANR in the wake of the Fukushima accident, and are coordinated by CEA in partnership with EDF and AREVA.

Examination of Unit 1 of the Fukushima Daiichi power plant could also help to improve knowledge of the interaction between corium and concrete and the effectiveness of cooling mechanisms, by providing information about a situation potentially involving large corium masses at reactor scale. Indeed, accident calculations performed

131. VULCANO Water-Uranium.

132. Assessment and Mitigation of the Risk of Third Containment Barrier Breakthrough at Nuclear Power Plants.

using core melt accident simulation codes like ASTEC, MAAP and MELCOR suggest that the core of Unit 1 was seriously degraded, that the vessel failed and that large quantities of molten corium flowed on to the concrete under the vessel, eroding a significant thickness of it. To understand more about the corium-concrete interaction phase in this reactor, it is necessary to work backwards and analyze the in-vessel degradation phase which, depending on the code used, produces some very different results in terms of composition, temperatures and the physico-chemical properties of the debris and corium [9].

► Possibility of in-vessel corium retention

One countermeasure envisaged by some licensees to try to avoid vessel melt-through in a reactor where the core has melted is to flood the reactor pit in order to cool the pit by organizing a two-phase flow of water around it. The effectiveness of this action depends on many different factors, including the reactor's size and power, the moment when the materials relocated to the lower head, the fraction and composition of the corium in the lower head, which determines the distribution of the flow of heat to the lower head internal wall, and the geometry of the vessel pit and characteristics of the thermal insulation around the vessel. The presence (or not) of specific geometric features (discontinuous thickness between the vessel shell and the lower head, penetrations through the lower head for instrumentation), which determine the flow of heat that can be evacuated by the outer face in the water, should not be ignored.

In a deterministic approach, a robust demonstration of the possibility of keeping the corium in the vessel, known as in-vessel retention (IVR), requires checking that the flow of heat through the vessel can be evacuated at any point on the vessel's external surface. This heat evacuation must be performed without drying out the wall, particularly in areas where the flow of heat is greatest, and this applies to all feasible configurations (stabilized or transient) of the debris bed[133], which are generally inhomogeneous. This depends on the distribution of heat on the vessel wall's internal surface and on the distribution of the critical flow at each point on the vessel's external surface, which depends on a number of parameters as explained above. The distribution of heat on the vessel wall's internal surface is the result of natural convection mechanisms within the different layers or piles of corium at the vessel's lower head, whereas the absolute values are dependent on the length of time before relocation to the lower head and the change in mass of the relocated corium, i.e. ultimately the total residual heat to be evacuated through the lower head wall. The physico-chemical balance of the materials that compose the corium will change and the situation can gradually evolve so that a stratified pool develops in which a non-miscible metal layer floats at the surface of a pool of oxides, which may itself be composed of two phases with different densities[134]. This

133. "Debris bed" is the term used to refer to the corium from the damaged core that has relocated to the lower head. It may contain solidified clumps, and pieces of the structures at the bottom of fuel assemblies (grids, etc.) that have fallen into the corium.

134. The behavior of the corium at the lower head is explained in brief here; it is described in more detail in Section 5.1.2 of the publication "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.

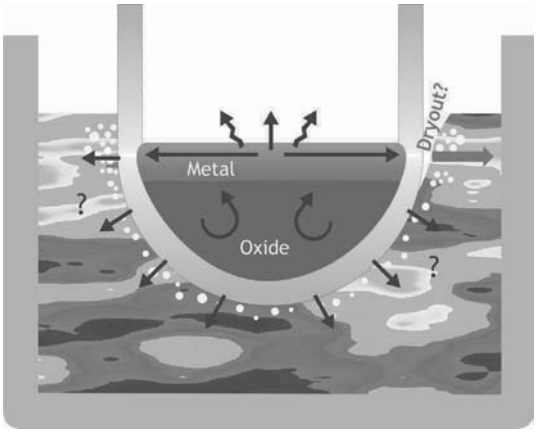

Figure 9.4 Heat flow distribution and risk of dryout © Georges Goué/IRSN.

configuration may be temporary when there is steam in the vessel because the gradual oxidation of the metal phases can produce a more favorable situation in terms of heat distribution, of a homogeneous pool.

If water has not been injected into the vessel – a likely situation in the first few hours of a core melt accident, which may in turn be due to a loss of coolant – the transfer through radiation from the metal layer floating on the oxide pool to the very overheated upper structures of the core is reflected in the fact that the surface temperature of the metal layer is higher than the melting point of steel (the interface temperature between the metal layer and the internal face of the vessel). The metal layer therefore transfers more heat to the vessel due to the imbalance between the interface temperatures. This results in a heat flow concentration, known as a "focusing effect", which causes a heat flow peak in the vessel at the location of the metal layer (Figure 9.4). The thinner the floating metal layer and the larger the mass of oxides under it (from which the residual heat is being released), the greater this peak is. If water is present, the formation of a metal crust on the pool's surface will reduce the interface temperature imbalance and limit this concentration.

Furthermore, given the big thickness of the vessel wall and the intensity of the heat flows to be evacuated through it, the vessel will partially melt starting from its internal face, and its thickness will be reduced to a few centimeters at drying flows of around 1.5 MW/m^2. The corium crust may then become unstable on contact with the liquid steel film, causing the transfer of the underlying molten steel to the corium pool and causing transient heat flows that are locally very intense when the liquid corium once again comes into contact with the wall[135]. From a mechanical point of view, a few centimeters of residual steel will be enough to keep the vessel intact once external cooling is provided and the vessel has been depressurized.

135. This phenomenon has been observed experimentally in tests conducted in the SCARABEE reactor on the total instantaneous blockage (TIB) of fuel assemblies in fast neutron reactors.

Although in-vessel retention appears possible for small, low-power reactors, the highest power level at which it would still be possible without being in danger of causing even more serious consequences is still the subject of uncertainty. If vessel failure should occur when the reactor pit is completely flooded, the ejection of the molten corium in the lower head could cause a steam explosion on contact with the water, which could damage the reactor containment (see Section 9.3.1 further on) and the components contributing to its leaktightness.

This is the reason why France is playing an active part in the European In-Vessel Melt Retention (IVMR – H2020) project, coordinated by IRSN with the participation of AREVA, CEA, EDF and 19 other European study and research bodies, with the aim of assessing whether a strategy of this kind could be successfully applied to Europe's existing 1000 MWe reactors (pressurized water reactors, boiling water reactors, VVER reactors). A common methodology will be defined for the project, which is expected to finish in 2019, for analyzing in-vessel retention capabilities and to complete experimental data in order to reduce uncertainty related to phenomena about which still little is known.

In the context of IVR, the effectiveness of core reflooding at a particular moment determines the mass of corium that will relocate to the lower head and therefore the absolute value of the residual heat to be evacuated from the lower head. IRSN has begun some research on this theme, specifically concerning:

- modeling of the geometric characteristics of the different possible debris bed configurations (porosity, exchange surfaces, permeability, etc.);
- modeling of the thermohydraulic phenomena likely to occur during reflooding for these different configurations;
- modeling of the oxidation phenomena of the metal-oxide mixtures in corium and of hydrogen production;
- impact studies of how all these phenomena interact.

The development and validation of the models rely in particular on the PRELUDE tests and the tests currently being performed by IRSN on a larger scale at the PEARL facility as part of IRSN's analytical experimental program on debris bed reflooding (PROGRES[136]) (see Section 9.1).

Concerning focusing effect risks, IRSN is working to identify the conditions in which the layer of metallic materials can float for a long period of time on the oxide layer, and for this it is studying the kinetics of transition between these layers, taking particular account of crust formation at the interfaces and the oxidation of the metal phases. The experiments on this subject are being run as part of the CORDEB program mentioned in Section 9.1 and will continue at the facility used for this program once it is finished (2016–2019), as part of the European IVMR program coordinated by IRSN.

136. Analytical Axperimental Program on Debris Bed Reflooding.

Finally, concerning interaction between the corium and water, research activity has been combined within the ANR RSNR ICE[137] project, which will be discussed in Section 9.3.1 further on.

An in-vessel retention strategy was adopted in Finland in 1995 when the power of the VVER-440/213 reactors at the Loviisa nuclear power plant was increased to 510 MWe, by installing a system of mobile insulation to create a cooling space around the vessel and by modifying the reactor pit ventilation system to create a cooling system. Experimental programs (COPO[138] in Finland and ULPU[139] in the United States) and numerical simulations were used to check that the cooling system performance was sufficient, taking into account bounding heat flow distributions at the lower head.

As far as new reactors are concerned, the in-vessel retention strategy has been adopted for the AP600. Given the power level of this reactor (600 MWe), the margins were considered adequate, even in the absence of water injection into the vessel, to validate the demonstration, allowing the U.S.NRC to certify the reactor in 1999. The same approach was used for the AP1000, which was certified by the U.S.NRC in 2005. It should be noted that there are no penetrations through the lower head for instrumentation in the AP600 and AP1000 reactors.

Three types of information were taken into account by the U.S.NRC:

- deterministic data;
- assessments giving sufficient confidence that if the vessel were to fail – particularly due to the focusing effect – there would be no cliff edge effect, or more specifically the reactor pit and the basemat could then provide their confinement function, withstanding any overpressure that might be caused by interaction between the molten materials and water[140];
- probabilistic data, integrating the risk of vessel melt-through and early loss of containment[141].

Interestingly, IVR is also the strategy chosen by the designer of the VVER 1000, which still provided an external core catcher (TianWan reactor in China and Kudankulam reactor in India).

137. Corium-Water Interaction.
138. Corium Pool Facility.
139. An IVR-related full-scale boiling heat transfer facility at University of California, Santa Barbara.
140. For the AP1000, this was the subject of an analysis presented in the document *Analysis of In-Vessel Retention and Ex-Vessel Fuel Coolant Interaction for AP1000* – NUREG/CR-6849, ERI/NRC-04-201 – August 2004. For the AP600, refer to the report *AP600 Final Safety Evaluation Report Related to the Certification of the AP600 Design* – NUREG-1512, 1998.
141. According to US regulations, probabilistic assessments must show that the frequency of massive early releases (less than 24 hours after an accident) is less than 10^{-6}/year.reactor. Given the uncertainties surrounding the models for corium progression to the lower head, for the AP1000 the designer assumed a conditional probability of vessel failure of between 4% and 30%, and postulated a systematic failure of the containment with a massive release into the environment. Since the calculated frequency of containment failure is 1.9×10^{-8}/year.reactor, with 38% of failures leading to early releases, the designer concluded that its design was well below the limit.

9.3. Dynamic loading of the containment by a sudden increase in internal pressure

This kind of phenomenon is likely to cause a relatively early loss of integrity of the containment, followed by a massive radioactive release into the environment. In line with the classification proposed by Norman C. Rasmussen in the early 1970s (Figure 9.5), we will distinguish between steam explosions (failure mode "α") and hydrogen explosions (failure mode "γ"); direct heating of the containment gases (Direct Containment Heating [DCH]) should also be taken into account.

9.3.1. Steam explosions

Steam explosions can occur when two fluids come into contact, where one (in this case the liquid corium from the core melting) is at a higher temperature than the boiling point of the other (the coolant). It is a thermodynamic reaction. When it enters the water, the corium jet fragments and causes intense vaporization of the water. This vaporization can create a shock wave that fragments the corium even more finely, further increasing the contact area between the two fluids and, as a result, the heat transfer. The process can propagate in the corium-water mixture, triggering an explosion. Simple contact

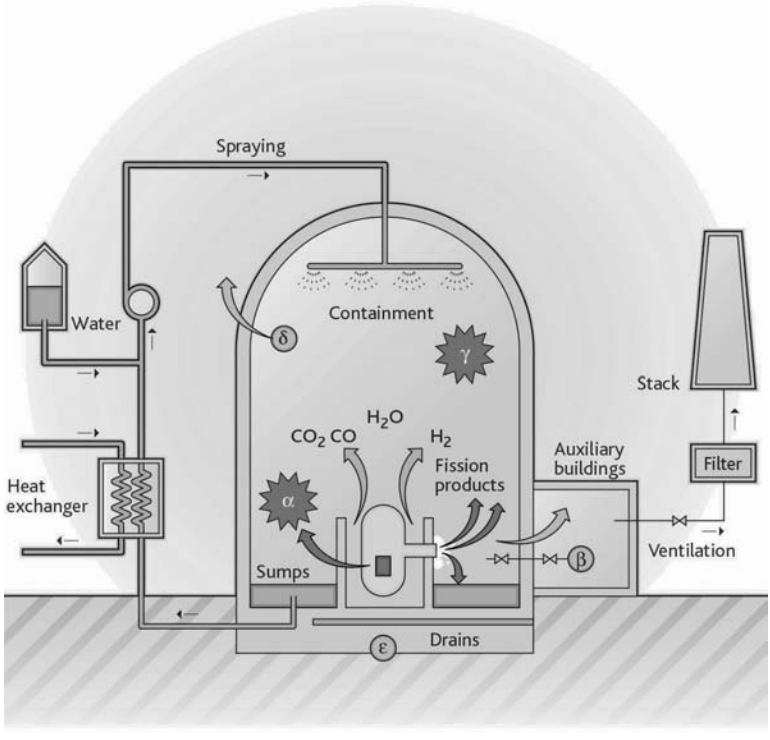

Figure 9.5 Schematic diagram of the possible containment failure modes according to the Rasmussen report [1].

between the two fluids is not enough to cause such an explosion. If the hot fluid fragments into quite large pieces (measuring millimeters or centimeters), the heat transfer will not be fast enough or intense enough, and will merely cause steam production characterized by a slow pressure increase in the system. If the hot fluid stays liquid on contact with the cold fluid due to the formation of a steam film and significant confinement restricting steam production in the first few moments, this is likely to trigger a very forceful explosion.

Fine fragmentation and explosion phenomena were initially studied in the 1970s as part of analyses of core melt accidents likely to occur in fast neutron reactors with sodium as coolant. From the late 1980s, this research was extended to PWRs. In France, analytical test programs were conducted by CEA on behalf of IPSN (then IRSN) and led to the development of heat transfer models.

The BILLEAU tests (which ended in 1997) were used to characterize heat transfer on the surface of solid metallic spheres heated to 2200 °C and plunged into water. The TREPAM program (which ended in 2003) consisted of measuring the heat flow on the surface of a tungsten filament plunged into water, for different filament diameter values (10 to 250 µm), filament temperatures (1080 °C to 2630 °C), water pressure values (0.1 to 21 MPa), water undercooling values (0 to 80 °C below the water saturation temperature) and plunge speeds into the water (0.2 to 46 m/s).

Abroad, tests have been conducted with more representative materials, sometimes on a larger scale. These include the following programs:

- CCM[142], run by the Argonne National Laboratory (United States) in the early 1990s, which studied the fragmentation of molten mixtures containing 2 to 12 kg of UO_2, ZrO_2 and steel at 2800 °C;
- SUW[143] and WUMT, run at Winfrith (United Kingdom) at the same time, with a uranium-based thermite[144];
- FITS[145], run at the Sandia National Laboratories (United States) with an aluminum-based thermite;
- ALPHA, conducted by the Japan Atomic Energy Research Institute (JAERI), Japan, with the same material;
- FARO, run by the European Commission's Joint Research Centre at Ispra (Italy), which ended in 1999 and studied, in the TERMOS vessel, the fragmentation of molten mixtures of 18 to 176 kg of UO_2 and ZrO_2 and the steam explosion resulting from dropping this into water, under pressures of 0.2 to 5 MPa;

142. Cold Crucible Melting.
143. Scale-Urania-Water.
144. The corium is represented by a mixture consisting of an oxide and a molten metal obtained by means of an exothermic oxidation-reduction reaction like the reaction between iron oxide, Fe_2O_3, and aluminum (aluminothermic).
145. Fully Instrumented Test Series.

- KROTOS, originally run by the same research laboratory, then by CEA after 2004 in a shock tube (one-dimensional geometry), to study fragmentation on contact with water of a liter of various molten materials (tin at 1000 °C, alumina from 2300 °C to 2800 °C and a UO_2-ZrO_2 mixture at 2800 °C), and the steam explosions that occur as a result;
- PREMIX, run by the research center at Karlsruhe (Germany), which studied the fragmentation of 16 to 60 kg of aluminum-based thermite producing molten alumina brought to 2300 °C;
- ECO[146], conducted at the same laboratory, which studied the explosion resulting from contact between 6 to 18 kg of molten alumina brought to a temperature of 2300 °C and one to two meters of water.

The results of these various research programs provided a large amount of experimental data. However, the results were fairly diverse: some tests produced explosions but others did not, for no apparent reason except the physical properties of the materials used (the molten alumina had a greater tendency to cause explosive interactions than the UO_2–ZrO_2 mixture, the thermal conductivity of which is lower).

On the basis of the analytical tests, IRSN and CEA together developed the MC3D[147] steam explosion simulation code. The code uses an eulerian description with a mixed finite volume (mass and energy balances) and finite difference (momentum balance) method with a three-dimensional (cartesian or cylindrical) structured mesh. It simulates both the fine fragmentation and the explosion phases. The complexity of the phenomena to be described during fragmentation requires very detailed modeling with precise numerical schemes. The fuel behavior is described using a three-field model:

- a continuous liquid field,
- a field for droplets from fragmentation of the continuous field,
- a fine solid fragments field.

The Rayleigh-Taylor instabilities at the interface between the corium and the water are also modeled.

To advance the modeling of the phenomena, the OECD/NEA launched the international SERENA project (2001–2005) [10], to which France (IRSN and CEA) contributed alongside Germany, Japan, Korea, the United States and Russia. Its objective was to determine the state of the art of knowledge about steam explosions, to compare simulation codes with one another, to assess their ability to predict explosions and to identify the areas where uncertainty needs to be reduced.

The OECD/NEA's international SERENA-2 program [11], which succeeded it (2005–2012) included 12 tests conducted by CEA at the KROTOS facility, which was moved from Ispra to Cadarache at the end of the FARO program, and by the Korea Atomic Energy

146. Experiments on energy COnversion during a steam explosion.
147. Multiphasic and thermohydraulic 3D simulation code.

Research Institute (KAERI) at the TROI[148] facility in Daejon, South Korea. The objective was to gain a better understanding of the mechanisms that trigger steam explosions and the effect of the type of materials used to represent corium, and to check the models' capacity to reproduce effects related to the geometry of the interaction zone.

The tests run at the KROTOS facility consisted of injecting 5 kg of molten corium (UO_2–ZrO_2 mixture but also more realistic compositions including the metallic phases or iron oxide) in a tube (one-dimensional geometry). An X-ray cinematography system was used to film the fragmentation of the jet in real time and the vaporization of the water around the corium fragments. In the TROI facility, larger corium masses of 20 kg (identical composition to those at KROTOS in the frame of the SERENA program) were injected into a container of similar height and width (two-dimensional geometry representative of a PWR reactor pit). The tests were defined after preliminary calculations had been made with the various simulation codes. The analysis of the results was discussed by international experts. The program also included code benchmarking exercises and reactor-scale applications.

The SERENA and SERENA-2 programs advanced understanding of the fine fragmentation of a corium jet in water and the explosions that can be caused by this. They revealed the influence of corium fragment solidification kinetics on the magnitude of the explosion (rapid solidification restricts the fine fragmentation process, and therefore heat transfers and water vaporization). In general, the simulation codes developed globally give a good overall prediction of the mechanical energy evaluated in tests using molten alumina and overestimate the mechanical energy evaluated during tests with molten corium. An analysis of the images taken during tests at the KROTOS facility shows that instabilities of the steam film surrounding the corium jet are correctly reproduced by the MC3D code's models. The code also correctly predicts the attenuation of the shock wave when the explosion occurs in a two-dimensional geometry like those studied in the TROI facility.

However, the effects related to corium oxidation during the interaction are still not properly understood. The research, particularly on fragmentation of the jet and oxidation of the corium, is continuing with theoretical and experimental analytical studies as part of the ICE project (2013–2017), launched in response to ANR's call for RSNR projects mentioned above and coordinated by IRSN in partnership with CEA, EDF and Nancy University.

9.3.2. Hydrogen-related risks

Hydrogen, which is produced mainly by oxidation of the zirconium in the cladding during the core degradation phase, but also by the oxidation of the other metals in the corium pool or in the basemat during the interaction phase between the corium and the concrete, can accumulate in the containment. The amount of movement of the atmosphere in the containment has an impact on the hydrogen distribution, which may therefore not be uniform. Locally, the hydrogen can reach high concentrations that

148. Test for Real cOrium Interaction with water.

exceed the threshold for the gaseous mixture to be flammable; hydrogen detonation in a containment is an event to be avoided.

The hydrogen distribution in the containment is the result of thermohydraulic phenomena which are complex to study due to the number of fluids in the containment atmosphere to be taken into account and the containment's compartmentalized geometry: convection caused by temperature differences between the atmosphere and walls and the possible use of spraying, condensation of steam on the containment walls with the water droplets produced by the spraying, diffusion, turbulence, the effect of the hydrogen recombiners, etc. [12, 13].

To assess the risk of hydrogen combustion in the containment, IRSN uses its own validated simulation codes (ASTEC, TONUS and the FLUENT code pending the implementation of P^2REMICS) and criteria from its own research programs, enabling it to determine which configurations are potentially dangerous. These tools enable it to predict hydrogen distribution and combustion, taking account of the effect of mitigating means of core melt such as recombiners, spraying and the venting system.

To predict hydrogen distribution inside the containment, IRSN uses two approaches: modeling based on several compartments in which fluid concentrations and temperatures are uniform, and more detailed multi-dimensional modeling with a large number of meshes. The first approach, which is economical in terms of calculation time, is used in the ASTEC code. The second is used in the TONUS code developed by IRSN in collaboration with CEA[149], and in FLUENT. The two approaches are complementary: the first approach is used to identify scenarios requiring detailed analysis using the second approach.

The validation of these simulation codes is reliant upon a number of experimental programs being conducted in France or abroad, at well-instrumented facilities simulating the geometry of a reactor containment on different scales:

- the PANDA facility (Figure 9.6a), at the Paul Scherrer Institute (PSI) in Villigen, Switzerland, consisting of four compartments with a total volume of 460 m^3, allowing the shape of fluid flows to be studied among other things;
- the THAI[150] facility (Figure 9.6b), run by Becker Technologies, in Eschborn, Germany, consisting of several compartments with a total volume of 60 m^3, with thermally insulated walls, designed for studying the distribution and combustion of hydrogen, and the operation of catalytic hydrogen recombiners;
- the MISTRA facility (Figure 9.6c) run by CEA in Saclay, with a cylindrical internal volume of 100 m^3, suited to the study of steam condensation on the walls and spray droplets;
- the TOSQAN facility (Figure 9.6d) run by IRSN in Saclay, with a cylindrical internal volume of 7 m^3, dedicated to the study of condensation and spraying, and of thermal and mass exchanges with the sump water.

149. EDF is developing the SATURNE code.
150. Thermal-hydraulics, Hydrogen, Aerosols and Iodine.

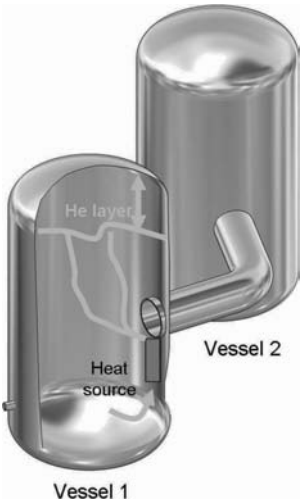

Figure 9.6a Diagram of the PANDA facility. © PSI.

Except in the THAI facility, for safety reasons hydrogen is simulated by helium, a gas with very similar physical properties but with the advantage of not being combustible. Tests carried out at the THAI facility have confirmed the similarity of hydrogen and helium distributions in the test volume.

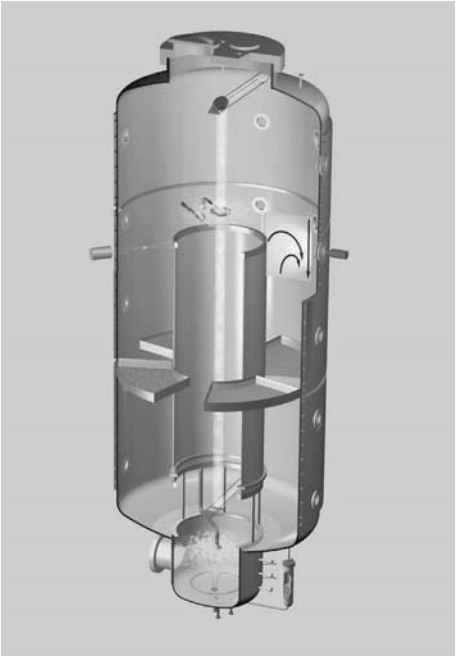

Figure 9.6b Diagram of the THAI facility [14].

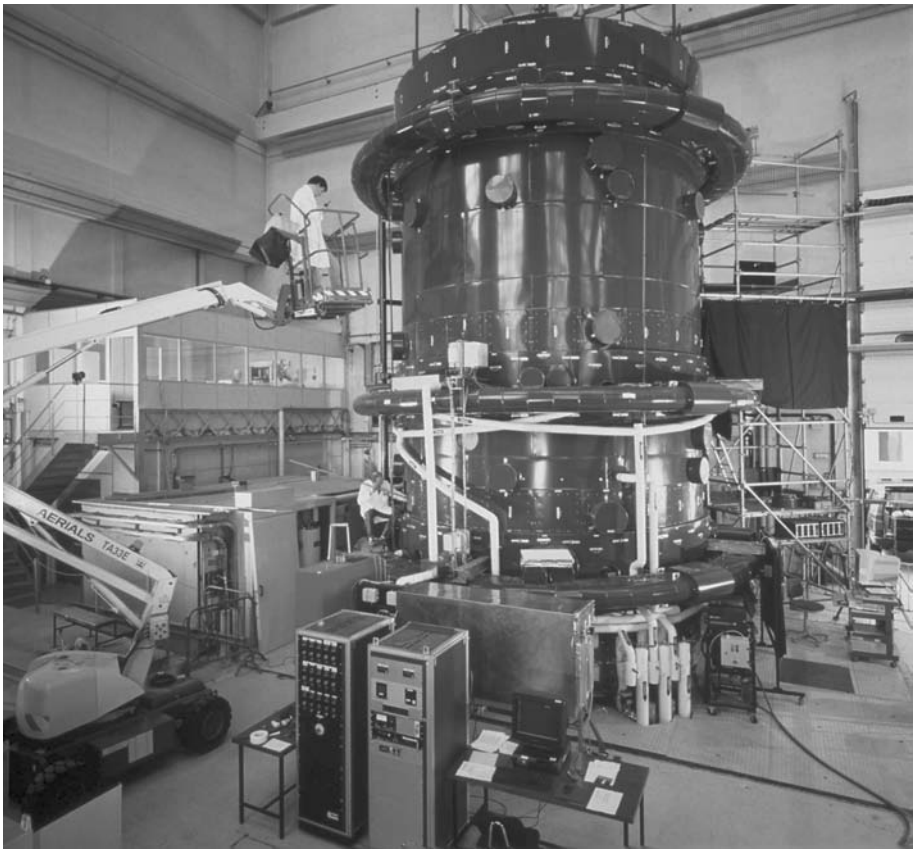

Figure 9.6c View of the MISTRA facility. © A. Gonin/CEA.

Many national and international projects and programs have collected a large amount of experimental data, enabling the scope of validation of the simulation codes to be extended. In particular these include the international OECD/NEA programs SETH (SESAR THermalhydraulics, 2000–2004) at the PANDA facility, SETH-2 (2007–2010) using both the PANDA and MISTRA facilities, THAI (2007–2009) and THAI-2 (2011–2014) at the THAI facility. The SETH-2 program studied the phenomena that cause the destabilization of stratified configurations, due to the effect of heat released by the hydrogen recombiners and spraying; because of its low density, the hydrogen is initially concentrated in the upper part of the containment.

Effects of scale have also been studied as part of the ERCOSAM[151]-SAMARA program (2010–2014), supported by the European Commission and by ROSATOM[152] (Russia), in

151. Containment thermal-hydraulics of current and future LWRs for Severe Accident Management.
152. Federal Agency on Atomic Energy.

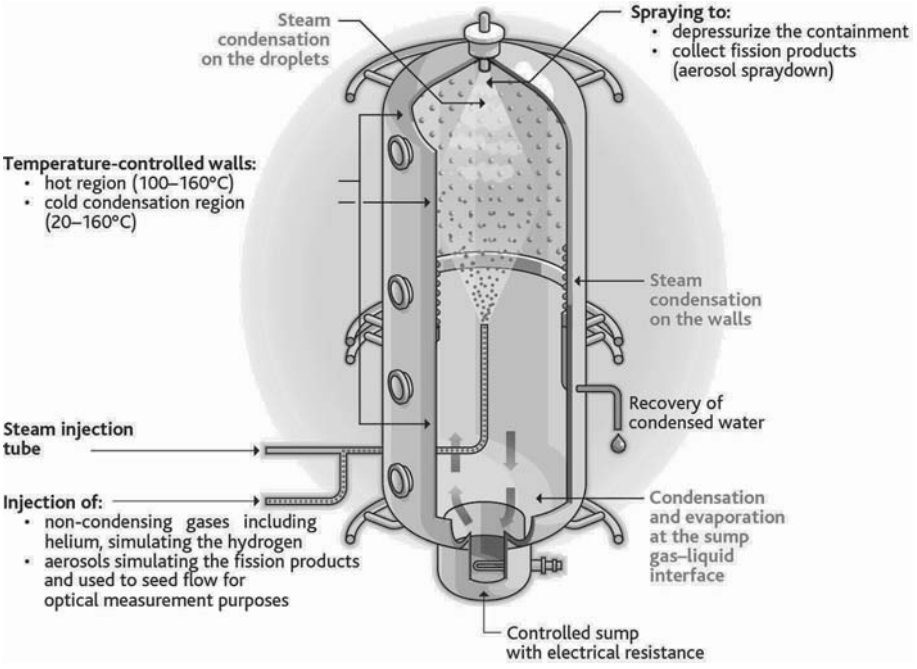

Figure 9.6d Diagram of the TOSQAN facility. © Stéphane Jungers/IRSN-source IRSN.

particular including tests at the PANDA, MISTRA and TOSQAN facilities already mentioned, as well as at the SPOT facility (60 m^3) in Russia. The extrapolation of the results to large scales was performed on the basis of preliminary calculations. It could be confirmed in the future by means of tests at the 2000 m^3 Russian KMS facility (currently at the design stage).

Concerning combustion, it should be noted that a ternary mixture of air, water and hydrogen (Figure 9.7), considered to be flammable, can lead to different combustion regimes, which depend mainly on the maximum hydrogen concentration reached and the distribution of this gas between the compartments:

- laminar deflagration,
- rapid deflagration,
- the deflagration-detonation transition (DDT).
- detonation – a phenomenon that can have disastrous consequences and that should therefore be avoided.

In a mixture known to be flammable, combustion may be triggered by an energy source of a few millijoules. Consequently, in the presence of electrical power sources or hot points, it seems probable that ignition would occur rapidly once the gas mixture enters the flammability domain. In contrast, a much more powerful energy source (at least 100 kJ) is required to trigger a stable detonation. This explains why direct detonation of the gaseous mixture in the

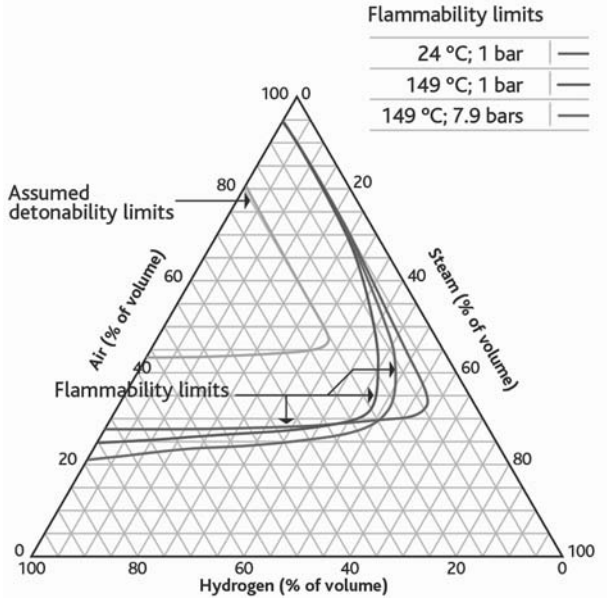

Figure 9.7 Shapiro diagram for hydrogen, air and steam mixtures (please note that the detonability limits shown have been called into question by later studies). @ DR.

containment can be ruled out; the only mechanism considered likely to cause detonation is flame acceleration and the deflagration to detonation transition.

Two types of criteria have been defined and adopted:

- the criterion "σ" related to flame acceleration. The quantity σ is the mixture's expansion factor, a ratio of fresh and burnt gas densities at constant pressure. It is an intrinsic property of the mixture in question; the critical value σ^* above which flame acceleration is possible depends on the initial temperature of the gases and the stability of the flame; it has been determined using the results of many experiments at different scales and in different geometries;
- the criterion "λ": similarly, prerequisite conditions have been defined for assessing the possibility of a deflagration–detonation transition (DDT). They are based on comparison of a characteristic dimension of the space being studied with the detonation cell size (denoted λ), which characterizes the mixture's reactivity.

Flame propagation after ignition, in an air, steam and hydrogen mixture, has been a study theme of many experimental programs [15]. The purpose of these tests is twofold: to study the transition between combustion modes and to produce a database of experimental data for validating simulation codes.

On this theme, IRSN is engaged in a long-term collaboration with the CNRS Institute for Combustion, Aerothermal Engineering, Reactivity and Environment (ICARE) in

Orléans. Besides determining the flammability limits of the mixtures, work done as part of this collaboration has enabled the flame acceleration criterion "σ" to be refined.

An analytical test program has therefore been run since 2004 at ICARE's ENACCEF[153] facility (Figure 9.8). This facility was designed to represent, on a 1/24 scale, a steam generator bunker and the containment dome of a 900 MWe PWR. ENACCEF consists of a dome with a volume adjustable between 781 and 958 liters (representing the free volume of the containment above the steam generator), connected to a cylindrical tube that is 3.2 m high, has an internal diameter of 0.17 m (representing the steam generator bunker) and can contain obstacles, in which flame acceleration can be studied. The instrumentation it contains includes photomultiplier and pressure sensors to measure the progression of the flame front and the pressure generated along the tube. In addition, the composition of the gaseous mixture is analyzed at several places in the test section; laser Doppler velocimetry (LDV) and particle image velocimetry (PIV) measurement techniques are used to measure the velocity field of the flow before the flame arrives.

The tests have been used to refine the expression of the critical values of expansion factor "σ" for homogeneous mixtures and extend it partially to heterogeneous mixtures.

Tests on a larger scale had also been conducted previously in the 62 m-long RUT facility at the Kurchatov Institute in Moscow, Russia. This facility consists of a straight canyon connected to a cavity, which is itself connected to a second, shorter canyon curved at the end. All three zones are rectangular in cross-section and may be obstructed

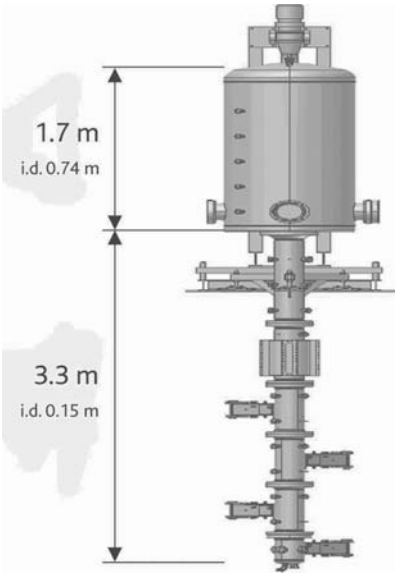

Figure 9.8 Diagram of the ENACCEF facility [16].

153. Flame Acceleration Enclosure.

by obstacles. This geometry can be used to study both monodirectional flame acceleration in the canyons and more complex 3D effects or interactions in the cavity. The mixtures used contained hydrogen, air and sometimes a diluent (steam). A test program run on behalf of IPSN in partnership with German researchers at the Karlsruhe center (FzK) studied the different combustion regimes and obtained values for the criteria "σ" (flame acceleration) and "λ" (detonation cell size), which are used for predicting different combustion regimes in the ASTEC and TONUS simulation codes[154].

This program was completed by the HYCOM (integral large scale experiments on HYdrogen COMbustion for severe accident code validation) program, run from 2000 to 2003 with support from the European Commission. Its aim was to study flame acceleration in mixtures of hydrogen and air (without water), particularly validation of the "σ" criterion. The effect of the expansion of the burned gases (piston effect) and the effect of compartmentation were studied in the 12 tests conducted in the RUT facility. The effects of pressure vents, concentration gradients and changes in cross-section were studied in 46 more analytical tests run at the smaller-scale DRIVER and TORPEDO facilities at the Karlsruhe center in Germany. These facilities consisted of two cylindrical tubes with diameter 174 mm and 520 mm respectively and length 12 m and 6 m respectively. The program, in which IPSN (then IRSN) played an active part with the support of EDF, also included a reactor-scale simulation code benchmark exercise, studying a containment geometry similar to that of EPR.

Research on hydrogen distribution in a reactor containment (effects of spraying and recombiners – see below), on combustion (deflagration to detonation transition on a large scale) and explosion is continuing as part of the MITHYGENE[155] program (2013–2018). The program was launched in response to ANR's call for RSNR projects mentioned earlier, and is coordinated by IRSN in partnership with CEA, ICARE, AREVA Expansion, the German Jülich research center (KfJ), EDF and Air Liquide.

To reduce the hydrogen content of a containment in a core melt accident, passive catalytic hydrogen recombiners (Figure 9.9) have been installed in all the containments of French reactors.

The behavior of recombiners in situations representative of core melt accidents has been the subject of a number of test programs, conducted primarily by manufacturers (Siemens, AECL, etc.). IPSN ran the H2PAR[156] test program (1998 to 2000) in partnership with EDF, to study the possible poisoning of catalytic sites in recombiners by fission products, particularly the halogens. The facility consisted of an oven that could bring approximately 10 kg of a UO_2–ZrO_2 mixture up to a temperature of 2900 °C. Inert compounds representative of the main fission products were included in this mixture. The aerosols and the gases released by the molten mixture spread inside a 7.6 m^3 containment made from Terphane, in which a mock-up representing a hydrogen recombiner used in French reactors was installed. The tests also provided an opportunity to study the risk of the hydrogen in the recombiner igniting. The atmosphere in the

154. The equivalent German code, COM3D, uses equivalent criteria.
155. Hydrogen Mitigation.
156. Hydrogen Passive Autocatalytic Recombiners.

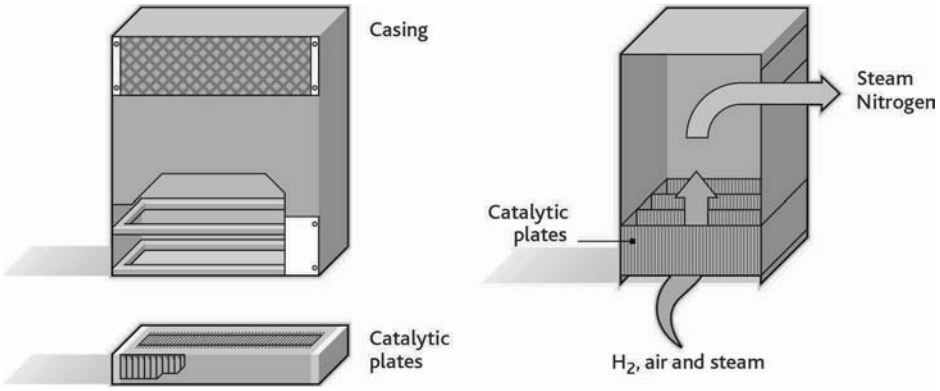

Figure 9.9 Diagram of a passive catalytic hydrogen recombiner. © IRSN.

containment contained a mixture of air, hydrogen and steam; the proportions were one of the parameters studied. The impact on recombination performance of various geometric parameters (number of catalytic plates, height of stack) and chemical parameters (several catalytic plates replaced by neutral chemical plates) was also studied.

The KALI H2 test program, run by CEA at the end of the 1990s in partnership with EDF, also provided an opportunity to study the effects of humidity, exposure to smoke from cable fires and carbon monoxide on recombiner performance. The results of this program revealed that humidity has little impact on recombiner performance. They also showed that recombination was deactivated by carbon monoxide in low-oxygen atmospheres and that there was a loss of performance of approximately 5% for recombiners exposed to cable fire smoke. The fire on May 10, 2011 in the containment of reactor 2 at the Ringhals nuclear power plant in Sweden also revealed the possibility that catalytic recombiner plates could be perforated by chlorinated products from the combustion of plastics. Investigations are therefore being carried out by IRSN and the Jülich research center to study the effect of cable fires on recombiners. The first results of this study have revealed that the start of recombination is delayed in the case of catalytic plates exposed to smoke from cable fires [17].

During the Phebus FPT3 test, specimens from recombiner plates made by various manufacturers were exposed for several dozen minutes to the radioactive aerosols and steam in the tank simulating the containment. The low oxygen content of the tank (for safety reasons) slightly complicated the analysis of the results, which nevertheless not revealed unexpected phenomenon.

To investigate the effect of a lack of oxygen, of carbon monoxide or of the presence of dust on the startup and effectiveness of recombiners, IRSN is working with the Jülich research center in Germany, which has the REKO experimental platform with a number of facilities on different scales (Figure 9.10). On the basis of the experimental results, IRSN is developing the SPARK simulation code, which can be used to take account of all the physico-chemical phenomena governing recombination. This collaboration has

Figure 9.10 View of the REKO platform. © Forschungszentrum Jülich.

already enabled limits to be defined for hydrogen ignition by recombiners. These results were confirmed by the test program conducted as part of the OECD THAI and THAI-2 projects. The findings of these various tests allow the improvement of the models integrated into the simulation codes used at IRSN (including ASTEC) and as a result to refine the assessments of hydrogen explosion risk in the annulus space between dual containments, which IRSN is required to produce for 1300 MWe and 1450 MWe reactors, and in fuel storage buildings (and the ITER facility).

9.3.3. Direct containment heating

Direct containment heating (DCH) was originally studied experimentally and analytically by the Americans in the 1990s for the PWRs at the Zion, Surry and Calvert Cliffs sites. If a reactor vessel fails under pressure, corium could be ejected and fragmented in the reactor pit and the fragments could be carried into the containment. The metallic components of corium (such as zirconium and iron) would oxidize on contact with the steam and air in the containment, releasing heat and producing a quantity of hydrogen

that would be added to the amount already produced by oxidation of the cladding at the start of the accident but not recombined. Direct containment heating due to radiation and convection, and hydrogen combustion, would cause a sudden increase in pressure inside the containment which, depending on the mass of corium ejected and hydrogen produced, would threaten its integrity.

The American studies concluded that the corium dispersal and the amplitude of the resulting overpressure were very dependent on the geometry of the reactor pit. Analytical tests in which the corium was simulated by water and the geometry studied (reactor pit and communication channels with the free volume of the containment) was that of a 900 MWe reactor, were conducted at KAERI in South Korea on behalf of IPSN in the late 1990s. The results enabled an empirical model to be created that was integrated into the ASTEC simulation code, giving the fraction of corium ejected from the reactor pit as a function of the geometric parameters and ejection conditions.

In the early 2000s, IRSN launched a new research program in partnership with German researchers in the DISCO[157] facility at the Karlsruhe research center, consisting of two experimental devices. The first was a plexiglass mock-up enabling the flows to be seen, with the corium simulated by water, gallium or Wood metal[158], an alloy with a low melting point. The second was used for more representative experiments, where the corium was represented by a mixture of 10 kg of molten alumina and iron obtained using an exothermic oxidation-reduction reaction, brought to approximately 2700 °C. Several geometries were studied in these devices: those of an EPR, a P'4-type reactor (1300 MWe), a VVER-1000 and a German KONVOI PWR. The program, which was funded by IRSN[159], consisted of 60 analytical tests at low temperature and 12 integral tests at high temperature on a 1/16 scale model of the containment of a P'4 reactor representing the geometry of the reactor pit, the compartments in communication with the reactor pit and the dome.

The experimental results were analyzed using the MC3D simulation code (Section 9.3.1). The analysis established a correlation giving the corium fraction dispersed axially in the reactor pit as a function of the pressure in the vessel at the moment when it fails. Another finding is that the pressure increase in the containment is mainly due to combustion of the hydrogen produced during dispersal of the corium and, to a lesser extent, of the hydrogen already in the containment when the vessel failed. The rate of hydrogen production after vessel failure is correlated with the fraction of corium dispersed axially in the dome.

These results have significantly improved the direct containment heating model in the ASTEC simulation code, enabling it to be applied with greater confidence to the geometry of French PWRs.

157. DISpersion of simulated COrium.
158. Wood metal is an eutectic alloy of bismuth, lead, tin and cadmium. Its melting point is approximately 70 °C.
159. Except in the case of the KONVOI reactors.

9.4. Radioactive releases

The amplitude and nature of radioactive releases depend on three main processes:

- the emission of radioactive products by the fuel when it is damaged,
- the transporting of these radioactive products into the reactor's circuits,
- their behavior in the containment.

The volatility of the radioactive products depends on their physical state, which in turn is determined by their chemical nature. Chemical reactions therefore play a key role, bearing in mind that they occur at a very wide range of temperatures (from 50 °C to 3000 °C) and involve a large number of elements: fission products, the fuel and the core's structural materials (zirconium, steel, materials that absorb neutrons such as AIC and/or boron carbide, etc.). These materials release a considerable mass of steam and aerosols, which can be much higher than the mass of all the fission products put together. The nature of the materials used in the walls of the circuits and containment also plays a part.

The research (analytical experiments) conducted specifically on the above three processes is discussed in more detail in the sections that follow. The findings concerning each of these processes from the more general (integral) tests in the Phebus-FP program will then be explained.

The reference information [18] provides a summary of the knowledge acquired about radioactive releases in the event of a core melt accident.

9.4.1. Emission of radioactive products by fuel

The emission of fission products by the fuel while it is being heated has been studied in France as part of the HEVA[160] program (eight tests in the 1980s) and VERCORS[161] program (17 tests from 1989 to 2002) run by CEA at its Grenoble center on behalf of IPSN and EDF. At around the same time, similar programs were being run abroad:

- the HI[162] tests (six tests) and VI tests (seven tests) at the Oak Ridge National Laboratory (ORNL), United States,
- the VEGA[163] tests (10 tests) by the Japan Atomic Energy Research Institute (JAERI), Japan,
- the AECL-CRI tests (more than 300 tests) by Atomic Energy of Canada Limited (AECL), at the Chalk River Laboratories, Canada.

All these tests consisted of heating, in an oven, specimens of fuel that had been in a reactor, to identify the fission products released and measure the quantities, using in-line gamma-ray spectrometry and other radiochemical analysis techniques on samples. The main parameters studied were the type of fuel, its burnup rate, whether or not there was

160. Helium, Steam.
161. Realistic Verification of the Containment of Reactors.
162. Horizontal Induction.
163. Verification Experiments of radionuclides Gas/Aerosol release.

cladding present, the maximum temperature reached during the test and the composition of the gas flowing around the fuel: a mixture of steam and hydrogen in variable proportions, inert gases, air.

In France, the first six tests in the VERCORS program (VERCORS-1 to 6) were conducted between 1989 and 1994 on samples of UO_2 fuel heated to 2300 °C. Then, two test series, VERCORS-HT (three tests) and RT[164] (eight tests), at higher temperatures up to the melting of the fuel, followed between 1996 and 2002. These tests used UO_2 and MOX fuels in the form of fuel rod segments or a debris bed. The device used for the HT tests included a cooled, instrumented channel downstream of the fuel, allowing the behavior of the fission products emitted during their transfer in a reactor coolant system to be studied, and in particular their deposition on the walls.

The fuel specimens generally consisted of a fuel rod segment containing three fuel pellets and their cladding, sampled from fuel rods that had spent several years in an EDF reactor. In most of the tests, the segment was re-irradiated for several days in a research reactor (the SILOE reactor in Grenoble, then the OSIRIS reactor at the Saclay center once the SILOE reactor had been shut down) in order to restore the inventory of short-lived fission products such as iodine-131 that would have disappeared after the rod was unloaded from the core of the origin reactor.

The tests enabled a database of very precise experimental data to be obtained on the emission kinetics of fission products and actinides in different configurations. This database was used to develop and validate empirical models integrated into the ASTEC simulation code. A more sophisticated simulation tool, MFPR[165], based on a more detailed description of the physical and physico-chemical phenomena, was also developed by researchers at the Nuclear Safety Institute of the Russian Academy of Science (IBRAE) in Moscow in collaboration with IRSN; it is used to analyze the results of the tests.

It emerged from all this research that it is possible to distinguish broadly speaking between the following:

- the volatile fission products such as the isotopes of krypton (Kr), xenon (Xe), iodine (I), cesium (Cs), rubidium (Rb), tellurium (Te), and antimony (Sb), which are almost entirely released by the fuel when it reaches 2300 °C;
- the semi-volatile fission products, such as the isotopes of molybdenum (Mo), barium (Ba), yttrium (Y), rhodium (Rh), palladium (Pd), and technetium (Tc), the emission rate of which lies between 10 and 100% and depends heavily on the oxygen potential of the fluid flowing along the fuel being studied[166];
- the low-volatile fission products such as the isotopes of strontium (Sr), niobium (Nb), ruthenium (Ru), lanthanum (La), cerium (Ce), europium (Eu), neptunium (Np)

164. Release of Transuranics.
165. Module for Fission Product Release.
166. The molybdenum oxides are much more volatile than the metal molybdenum, which explains why greater quantities are emitted in steam; the opposite is the case with barium, which is emitted more readily in its metal form in hydrogen.

and some actinides such as uranium (U), of which 10% maximum is generally emitted, with the notable exception of ruthenium, which is more volatile when the fuel is exposed to air rather than steam;

– the non-volatile fission products and the actinides such as the isotopes of zirconium (Zr), neodymium (Nd) and plutonium (Pu), of which no emissions were really detected during the different tests.

The VERCORS facility was dismantled and a new facility that performed the same functions, VERDON, was built by CEA on the Cadarache site. The facility is used to study not only the release, but also the transporting and deposition of the fission products in thermal-gradient tubes. Tests with MOX and UO_2 fuel with high burnup rates, in a mixture of hydrogen and steam, and a test in the presence of air, were performed in this new facility as part of the International Source Term Program (ISTP, 2005–2013).

The emission of fission products during core reflooding was studied experimentally in the LOFT-FP tests already mentioned in Section 9.1. They revealed significant releases of volatile species (iodine and cesium isotopes) during runaway oxidation of the cladding. Tests were also carried out in the early 2000s as part of the ISTC 1648 research program run by the Scientific Research Institute of Atomic Reactors (NIIAR) in Russia, on sections of irradiated fuel rods from a VVER reactor heated to 1700 °C.

The emission of steam and of aerosols released by control rods containing a silver, indium and cadmium alloy (AIC) has been modeled by means of measurements made in the 1990s during specific tests (EMIS[167] program) conducted by CEA in Grenoble on behalf of IPSN on material specimens brought to melting point. AIC specimens were also vaporized during certain VERCORS tests (EMAIC[168] tests) to study their influence on the chemistry of the fission products released.

Once emitted by the core, the radioactive products and elements from the degradation of the core structures undergo rapid cooling. The vapors condense, forming aerosols (homogeneous nucleation), or condense on the aerosols already formed (heterogeneous nucleation) and on the reactor coolant system walls. The vapor of certain fission products such as cesium and iodine can also react chemically with the materials in the RCS walls (Ni, Cr, Fe) and be fixed, to a greater or lesser extent reversibly. The heating of the wall due to the heat released by the fission products or the partial pressure reduction of these elements in the fluid in the RCS can cause some of the deposited fission products to be re-emitted.

9.4.2. Transporting of radioactive products into the reactor's circuits

The mechanisms governing the transporting – and deposition – of aerosols in a circuit has been the subject of many experimental and theoretical studies, often conducted outside the nuclear field. This work has allowed basic models to be established that have

167. Fission Product Emission.
168. Silver, Indium and Cadmium Emission.

been integrated into severe accident simulation codes such as ASTEC. It involves the agglomeration of the aerosols contributing to an increase in their size, thermophoresis and diffusiophoresis[169], phenomena responsible for deposition in the presence, respectively, of a temperature gradient between the carrier fluid and the wall or of condensation of the carrier fluid on the wall, gravitational settling or impaction on a wall if the carrier fluid changes direction, Brownian or turbulent diffusion, etc.

Analytical test programs have been used to check these models, particularly the TUBA and TRANSAT programs run by IPSN in the 1990s to study thermophoresis and diffusiophoresis, and the DEVAP[170] program run at the same time by CEA on IPSN's behalf at the Grenoble research center, consisting of measuring deposits by condensation of cesium iodide and hydroxide injected into a tube with a controlled thermal gradient. The temperature profile of the wall of this kind of tube decreases between the inlet and the outlet, which is representative of the thermal conditions in the reactor coolant system (RCS) of a PWR in accidental conditions. The FALCON program in the United Kingdom in the early 1990s also studied experimentally the deposition of simulated fission products (I, Cs) in controlled thermal-gradient tubes and the interaction between vapors and aerosols.

The aerosols deposited can go back into suspension if there is a sudden increase in flow rate of the fluid in the RCS, e.g. following an attempt at core reflooding. The phenomena involved were studied experimentally during the 1990s in the STORM facility at the Ispra Joint Research Centre in Italy.

Tests were also performed in the ARTIST[171] facility at PSI, Switzerland, as part of an international program, to study the retention of aerosols in the secondary system of a steam generator. The results obtained made it possible to evaluate more precisely the releases in the event of steam generator tubes rupture during reactor operation.

Given the uncertainties concerning the behavior of iodine in the reactor coolant system, confirmed during the Phebus-FP tests (see further on), IRSN built an experimental device named CHIP at its laboratories in Cadarache, to study the behavior of different chemical species in a reactor coolant system. The device consists of:

- several generators of chemical species in vapor or gas form: mixture of hydrogen and steam (H_2/H_20), simulators of highly volatile, reactive fission products such as diiodine (I_2), cesium (Cs), molybdenum (Mo), reagents present in large quantities during core melt accidents, such as boron (B), silver (Ag), indium (In), cadmium (Cd) from the reactor's control rods melting;
- a heating system to bring the various reagents to a temperature between 1500 °C and 1600 °C;

169. Movement of the aerosols caused by diffusion in a gaseous environment of water molecules to a cold wall on which they condense.
170. Volatile Fission Product Deposits in the Vapour Phase on Reactor System Surfaces.
171. Aerosol Trapping in a Steam-Generator.

– a controlled thermal gradient circuit made from the same metal as the reactor coolant system, into which the reagents are injected; this circuit includes high-temperature (500 °C to 900 °C) sampling devices, and devices for sampling at lower temperatures (approximately 150 °C) for separating the aerosol and gaseous species of iodine at both temperature levels;

– a filter for recovering at the circuit outlet the aerosols formed during the reagent cooling process so that they can be analyzed.

Tests were conducted by IRSN as part of the International Source Term Program (ISTP, 2005–2015) in order to quantify the fractions of volatile iodine produced in the reactor coolant system for reaction systems representative of the conditions of a core melt accident in a water reactor. An analysis of the chemical species that make up the aerosols collected in the filter also provides more information about the chemical species of iodine that are transported in a condensed form. Analysis of these results provides a better understanding of the chemical reactions involved in the production of volatile iodine and their kinetics, and also improves the iodine transport models in the ASTEC simulation code, so that potential radioactive releases in the event of a core melt accident can also be characterized.

9.4.3. Radioactive product behavior in the containment

The aerosols emitted within the containment are also subject to the phenomena of agglomeration, deposition and possibly re-suspension. The basic physical phenomena are the same as those at work in the reactor coolant system. Some components of the aerosols may be hygroscopic and absorb the steam in the containment, which increases their mass and accelerates their deposition by settling. The main deposition phenomena of aerosols in the containment are settling and diffusiophoresis. Spraying, which could be used to cool the atmosphere in the containment and reduce the pressure there, can also accelerate the deposition of aerosols.

The phenomenon of aerosol spraydown in particular has been studied at IRSN, during an analytical test program at its CARAIDAS facility in Saclay. This program was used to study and model the condensation of steam on water droplets and its evaporation, aerosol collection by the droplets when they fall and the absorption of gaseous iodine by the droplets. Experiments on the effects of spraying have also been conducted on a larger scale at the TOSQAN facility already mentioned. The results were used to develop and validate an aerosol deposition model integrated into the ASTEC simulation code.

A large amount of the radioactive products released by the core ultimately ends up in the sump water following their spraydown, leaching of the containment walls by the steam condensates or their deposition by settling. Consequently, the sumps are intensely radioactive. Moreover, with the notable exception of silver iodide (AgI), the majority of the metal iodides (CsI, RbI, CdI_2, InI) are water soluble and form I^- ions. Due to the action of the radiolysis products of the water and a large number of radiolytic and thermally activated chemical reactions, the I^- iodide ions can be oxidized to become volatile I_2 iodine, which is then released into the containment atmosphere. The formation of volatile iodine depends on many parameters, the most important being water pH. At a

basic pH, the formation rate is very low, making this a relatively simple countermeasure to implement in the event of an accident (addition of soda). The water in the sumps also contains organic materials originating mainly from the immersed paintwork. The iodine reactions with the organic radicals can produce volatile organic iodides such as methyl iodide CH_3I, which are also released into the containment atmosphere.

These different chemical reactions have been studied in France at IRSN's laboratories during analytical tests on specimens using a γ irradiator (IODE program in the late 1980s–early 1990s).

Because the iodine emanations from the sump also mean that mass is transferred from the water to the containment atmosphere, more global tests were then performed in the 1990s at CEA's CAIMAN facility on behalf of IPSN in partnership with EDF. This consisted of a 300 liters containment with a 25-liters sump in the field of a cobalt-60 source delivering a γ dose rate of 10 kGy/h. Plates coated with the same type of paint as used in French PWRs could be placed in the sump water and in the containment atmosphere. The water in the sump could be brought to 90 °C and the atmosphere to 130 °C. The iodine placed in the water in soluble form was mixed with a tracer (iodine-131). A system of selective filters, known as Maypack[172], was used to identify the different iodine species produced (I_2 and CH_3I) and their physical forms (aerosols or gases) and to measure the quantities using a gamma-ray spectrometer. The SISYPHE program, conducted at IPSN in support of the Phebus-FP program in a similar 10 m^3 containment, also contributed significantly to the development of models for iodine compound mass transfers in sump evaporation conditions.

The different programs provided a better understanding of the phenomena governing the production of volatile iodine species in sumps and enabled the modeling of those phenomena to be improved. They also identified the role of paints, the iodine absorption rate of which increases significantly with temperature. However, the reaction is not irreversible and some of the absorbed iodine revolatilizes in an organic form. With this research, data were gathered on the phenomenon and enabled initial correlations to be established for use in the simulation codes used to evaluate potential radioactive releases in the event of a core melt accident.

In Canada, a practically identical facility to the CAIMAN facility, the Radioiodine Test Facility (RTF) was used by the AECL at the Chalk River laboratories to run many experimental programs on volatile iodine production in sumps. In particular we will mention the Phebus-RTF tests conducted as part of the international Phebus-FP program. They were used to quantify the effects associated with the presence of insoluble silver in

172. Measuring device invented by F.G. May in the 1960s, in which a sampled gas circulates in succession through a membrane trapping aerosols then two filters: the first consisting of a knit mesh, which is a silver-coated braid that traps molecular iodine, the second consisting of active charcoal (impregnated with potassium), which absorbs the other forms of gaseous iodine, including the organic iodine. Measurement of the concentrations deposited at the different stages gives the desired result. Improvements were subsequently made by various IRSN researchers, particularly by replacing the active charcoal with zeolite and measuring the deposits in real time by γ scanning when the iodine is radioactive.

sump water, observed during the first Phebus-FP tests. When silver is significantly in excess that reduces the concentration of I^- ions and thus volatile iodine production in the sumps, even when the pH is acid.

Analytical tests were performed in the context of two international programs run by CNL[173] (formerly AECL) under the guidance of the OECD/NEA, in which France took part: the Behavior of Iodine Project (BIP, 2007–2011) and the Behavior of Iodine Project – Phase 2 (BIP, 2011–2014). Their objective was to study organic iodine production in more detail and establish more accurate models than those obtained to date, so that results obtained in the laboratory could be extrapolated with greater confidence to the scale of a reactor.

Given the uncertainties over iodine behavior, IRSN built a new facility, EPICUR, at its laboratories in Cadarache to study in more detail the physico-chemical processes of volatile iodine production; the tests began in 2005. The facility consists of a 5-liters tank that can be brought to a temperature of 120 °C and placed in the field of six cobalt-60 sources emitting an average γ radiation dose rate of 2.5 kGy/h. The iodine is traced by the iodine-131 and a Maypack scanned in real time by a gamma-ray spectrometer allows accurately measuring the production kinetics of the volatile iodine and whether it is molecular or organic iodine. A series of 30 tests was conducted as part of the ISTP international program (2005–2013), allowing the formation of organic iodine from painted surfaces, whether immersed or not, to be studied; the tests revealed the importance of organic iodine production from painted surfaces when they are in the tank atmosphere.

The tests were also used to study the formation and stability of the iodine oxides (IO_x) formed under radiation in the containment atmosphere. Indeed, the iodine in the containment atmosphere in its molecular and organic forms can react with the radiolysis products of the air, ozone and nitrogen oxides, to form iodine oxides and nitroxides, which are not volatile. These aerosols are likely to be deposited in the containment or in filters if the containment is vented. Preliminary tests at a laboratory in Germany (PARIS program), to help with understanding the results from the Phebus-FP program, had already showed the importance of these reactions, even for low molecular iodine concentrations. The IOx formation and decomposition mechanisms are still being researched as part of the OECD/NEA STEM[174] and STEM-2 programs being run by IRSN in order to develop the associated models.

The EPICUR facility is also being used, particularly as part of the OECD/NEA STEM program (2011–2015), to study the stability of iodine aerosols subject to radiation at different humidity levels, the effect of aging of paints on their ability to adsorb iodine and release volatile organic iodides, and lastly the mechanisms for long-term revolatilization of iodine deposited on the walls or in filtration media. The analysis of the results should reduce the uncertainties about predicting potential iodine releases in the event of a reactor core melt accident.

173. Canadian Nuclear Laboratories.
174. Source Term Evaluation and Mitigation.

In the early 2000s, tests conducted by IPSN at the laboratory in Saclay (RECI[175] tests) showed that heating the air loaded with aerosols passing through hydrogen recombiners can lead to decomposition of the metal iodides (typically CsI, AgI, InI and CdI_2) deposited on these aerosols by forming volatile iodine compounds such as I_2, HOI and HI. Tests on a larger scale conducted under more realistic conditions at the THAI facility mentioned earlier (THAI program) confirmed this phenomenon and enabled decomposition rates of around a few percent to be measured.

In parallel with all these analytical research programs, programs were run using large-scale research facilities. As we have seen, chemistry plays a key role in the behavior of the fission products emitted during core melt and evaluating the resulting releases. For this reason it was considered essential to check, by means of "integral" experiments, i.e. experiments that are as close as possible to reality, that the phenomena and their potential interactions are properly accounted for in the models used in computer codes for calculating releases. In the 1980s and 1990s, the Americans and Canadians had already conducted experiments in a reactor using irradiated fuel (PBF tests, LOFT-FP tests and FLHT tests mentioned earlier), in which appropriate instrumentation was used to identify and quantify the fission products emitted by the fuel. However, none of the experiments modeled the reactor containment. Moreover, the information obtained about the chemical forms in which the main fission products (Cs, I) were emitted where very imprecise.

9.4.4. Contribution of the Phebus-FP program on the various processes involved in radioactive releases

To fill these gaps, IPSN (then IRSN) ran the Phebus-FP program (1988–2012) at the PHEBUS experimental facility, already described in detail in Section 9.1.

The program's main findings concerning fission products and/or the validation of the associated models are as follows [7]:

- the models for the emission of fission products and actinides when fuel degrades, deduced from the VERCORS tests, are confirmed except for barium which is released in small quantities, probably as a result of the formation with zirconium of a less volatile chemical compound;
- the models for the transporting and deposition of aerosols allows their observed behavior to be reproduced overall, both in the reactor coolant system and in the containment;
- cesium is transported in the form of cesium molybdate ($CsMoO_3$), a less volatile species than the cesium hydroxide (CsOH) commonly assumed by experts to be the form and used until then in all the tests of separate effects to study the behavior of cesium;

175. Recombiner and Iodine.

- a fraction of the iodine released into the containment at a break in the reactor coolant system is gaseous, whereas the equilibrium thermodynamics calculations predict that all the iodine emitted by the fuel should at 150 °C be in a condensed form, cesium iodide (CsI); the reactor coolant system therefore constitutes a further source of gaseous iodine in the containment, which needs to be taken into account when calculating releases;
- this fraction of a few percent in tests performed with a silver, indium and cadmium (AIC) alloy as the control rod material, reaches nearly 100% in tests where the AIC is replaced by boron carbide;
- even when the sump water is acidic, there is very little iodine gas emanation from the sump in tests run with AIC since silver, as studied subsequently in more analytical tests, fixes the I^- ions.
- after rapidly decreasing, the volatile iodine concentrations in the containment stabilize after a few days at a value that is approximately identical regardless of the test. This reflects an equilibrium between the various chemical reactions that produce and destroy volatile forms of iodine in the containment: reactions in the containment atmosphere with the products of radiolysis of the air (ozone, nitrogen oxides), transfers between the atmosphere and paints on the walls, transfers between the atmosphere and the sump water, etc.

Iodine's unexpected behavior observed during the Phebus-FP tests (reactor coolant system as the main source of gaseous iodine in the short-term, concentration not changing in the containment in the long-term) led to a reorientation of the research programs studying the elementary phenomena (ISTP, STEM, BIP, BIP2, THAI, THAI-2), which until then had focused on studying radiochemical and chemical reactions in the aqueous phase (IODE, CAIMAN, RTF, etc.), because at that time the sump water was considered to be the only source of volatile iodine production. Very analytical and theoretical research was also begun as part of the process of setting up a laboratory "without walls", C3R[176], involving IRSN, the Laboratory for the Physical Chemistry of Combustion Processes and the Atmosphere (PC2A) in Lille and CNRS.

9.4.5. Aspects to be explored

A) Iodine-related and ruthenium-related aspects

In some accident configurations (after the vessel has failed, during an accident in a reactor that is shut down with the vessel open for reloading fuel, or during a spent fuel pool uncovery accident), fuel could degrade in the presence of a steam and air mixture. The Canadian AECL-CRL tests already mentioned revealed that substantial amounts of ruthenium were released under these conditions whereas, with a steam and hydrogen mixture, very little ruthenium was released as we saw earlier. Ruthenium is highly radiotoxic, and particularly its two isotopes ruthenium-103 and ruthenium-106, which have half lives of 39 days and 369 days respectively. The VERDON-2 test conducted in

176. Laboratory of Chemical Kinetics, Combustion and Reactivity.

2012 at CEA's VERDON facility as part of the ISTP program, on a MOX fuel irradiated at approximately 60 GWd/t, confirmed this phenomenon and enabled it to be better quantified. When it is transported in the circuits, ruthenium changes into less volatile species. However, as shown by analytical experiments performed in Hungary and Finland in the early 2000s, ruthenium tetroxide (RuO_4) in gaseous form, despite being unstable at room temperature, could continue to exist in small quantities and could accumulate in the containment. Tests conducted at the EPICUR facility as part of the ISTP program studied the behavior under radiation of this oxide in the containment and enabled it to be modeled.

Studies of the complex behavior of iodine and ruthenium in the circuits was pursued until 2015 as part of the OECD/NEA STEM program and continue beyond that date with the MIRE[177] and STEM-2 programs. If necessary, other tests could be performed beyond that at the CHIP facility to study and model in more detail the effect of the control rod materials on the chemical forms in which iodine is transported in the reactor coolant system.

B) Filtration of atmospheric releases

The course of the Fukushima Daiichi accident attracted attention to the value of fitting containment depressurization systems with very efficient filters, in particular that will capture all the volatile forms of iodine (diiodine [I_2], organic iodides such as methyl iodide [CH_3I]) even under core melt accident conditions[178] and for periods of several days. Research carried out recently at the PSI in Switzerland tended to show that the effectiveness of trapping all the volatile forms of iodine by aqueous solutions could be considerably increased by using an alkaline thiosulfate and quaternary amine combination. Moreover, in response to the call for RSNR projects launched by ANR in 2012 in the wake of the Fukushima Daiichi accident, IRSN is coordinating the MIRE research program (2013–2017) in partnership with EDF, AREVA and various universities (Lille, Marseille, Nancy, Nantes) on improving the retention of all forms of iodine in filtration media[179]. Lastly, the Passive and Active Systems on Severe Accident source term Mitigation (PASSAM, 2013–2016) research and development project coordinated by IRSN and supported by the European Commission brings together eight partners including EDF to explore possible improvements to the filtration systems in service and to study innovative devices that would be much more efficient.

C) A "water-borne release" counter-measure to be explored

IRSN is conducting studies of a geotechnical barrier as a "waterborne-release counter-measure" in a core melt accident situation. If the event of a core melt accident in a pressurized water reactor that does not have a core catcher, the molten materials can enter into direct contact with the reactor building's concrete basemat. As seen earlier, interaction

177. Mitigation of Releases into the Environment in the Event of a Nuclear Accident.
178. For example, where the electrical systems for heating the filters for removing the steam saturating the iodine trapping sites on certain materials such as active charcoal, are unavailable.
179. The containment depressurization systems of French nuclear power plant reactors are fitted with sand filters.

between corium and concrete is a subject of research aimed at establishing the precise conditions of erosion and the time until basemat melt-through. There is a risk – which has not been completely eliminated though it has been greatly reduced since EDF thickened the basemats of the two Fessenheim reactors seen as critical – that the molten materials and sump water will go into the soil, the ground water, then the nearest river or the sea.

In 2009, IRSN began a study of the feasibility and effectiveness of a vertical geotechnical barrier in the ground under the foundations of nuclear power plants, extending downwards until it reaches a relatively impermeable geological formation, to act as a retention system to prevent ground water from flowing to the site boundaries. A geotechnical barrier like this would offer a grace period of several months following a core melt accident, and also a practical way of limiting pollution outside the site, through the pumping out, storage and treatment of the contaminated water inside this enclosure.

Creating a geotechnical barrier of this kind would not be easy but would generally be possible with existing reactors. However, it would mean that there would have to be a layer of sufficiently watertight materials (clay, compact chalk, rock with few fractures, etc.) at an accessible depth in the ground in order to create a closed volume.

The investigations carried out by IRSN suggest that the feasibility of geotechnical barriers – using existing geotechnical barriers formed for the purpose of installation construction – has been demonstrated for all French nuclear power plant sites except Fessenheim, for which further studies are required, and hydrogeological studies have led to the conclusion that these systems are likely to be effective at most sites, though the Belleville and Cruas sites would still require specific study.

This matter is the subject of discussions with EDF in the context of ASN's formal requests. It could ultimately lead to research and development. For its part, IRSN plans to look into the design (from a safety and radiation protection point of view) of a system for pumping out the water collected in a geotechnical barrier during a core melt accident.

References

[1] N. Rasmussen *et al.*, Reactor Safety Study. An Assessment of Accident Risks in US Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), Washington DC, US Nuclear Regulatory Commission, 1975.
[2] Special Issue on ASTEC, *Nuclear Engineering and Design*, Vol. 272, 119–298, 2014.
[3] Major Achievements after 4.5 years of SARNET, *Progress in Nuclear Energy*, Vol. 52, 1, 1–157, 2010.
[4] ERMSAR 2013 Conference of the SARNET Network, *Annals of Nuclear Energy*, Vol. 74, 1–223, 2014.
[5] See the OECD/NEA website containing the NEA/CSNI reports on core melt accidents: http://www.oecd-nea.org/nsd/docs/.
[6] See the U.S.NRC website containing the NUREG reports on core melt accidents: http://www.nrc.gov/reading-rm/doc-collections/nuregs/.
[7] Special Issue. Phebus FP Final Seminar, *Annals of Nuclear Energy*, Vol. 61, November 2013.

[8] Ability of Current Advanced Codes to Predict In-Vessel Core Melt Progression and Degraded Core Coolability – Benchmark Exercise on Three Mile Island-2 Plant – Final report, NEA/CSNI/R(2015)3.
[9] Benchmark Study of the Accident of Fukushima Dai-ichi Nuclear Power Plant (BASF Project) – Phase 1 Summary Report, NEA/CSNI/R(2015)18.
[10] OECD Research Program on Fuel Coolant Interaction. Steam explosion resolution for nuclear applications – SERENA, Final Report, NEA/CSNI/R(2007)11, 7 September 2007.
[11] OECD/SERENA Project Report. Summary and conclusions, NEA/CSNI/R(2014)15. February 2015.
[12] State-Of-the-Art Report on containment thermalhydraulics and hydrogen distribution, NEA/CSNI/R(1999)16.
[13] A. Bentaib, H. J. Allelein *et al.*, International Standard Problem ISP-47 on containment thermalhydraulics, Final Report, NEA/CSNI/R(2007)10.
[14] OECD/NEA THAI project final report – Hydrogen and fission product issues relevant for containment safety assessment under severe accident conditions, rapport NEA/CSNI/R(2010/3).
[15] State-Of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation Transition in Nuclear Safety, NEA/CSNI/R(2000)7.
[16] H. Cheikhravat, « Étude expérimentale de la combustion de l'hydrogène dans une atmosphère inflammable en présence de gouttes d'eau ». Thesis from the University of Orléans, September 2009.
[17] A. Bentaib, N. Meynet, & A. Blayer, Overview on hydrogen risk R&D activities methodology and open issues, *Nuclear Engineering and Technology*, Vol. 47, 1, 26–32, 2015.
[18] L.E. Herrantz, T. Haste & T. Kärkelä , Recent advances in the source term area within the SARNET European severe accident research network (SARNET2), *Nuclear Engineering and Design*, Vol. 288, 56–74, 2015.