# Chapter 5
# Development of the core melt accident

## 5.1. Development of the accident in the reactor vessel

### 5.1.1. Progression of the core melt in the reactor vessel

#### 5.1.1.1. Introduction

If the reactor core remains dry for a considerable length of time, the temperature of the fuel rods rises and may locally reach levels that cause significant and irreversible core degradation. The mechanisms of this degradation are both chemical and mechanical. Depending on the local temperature levels, degradation may result in more or less severe hydrogen production, fission product (FP) release, and molten corium formation and propagation towards the lower head. These phenomena have been studied in many national and international research programmes [1, 2, 11, 12, 15, 22]. The main degradation mechanisms that appear as the core temperature rises, as well as their consequences, are described in Section 5.1.1.2. This description is followed by a presentation of the main experimental programmes that have increased the state of knowledge of the degradation mechanisms, as well as a description of the modelling and computer codes that capitalise on that knowledge. The main mechanisms involved in the evolution of the fuel rod and core degradation are schematically shown in Figures 4.1, 4.2 and 4.3 of the previous chapter.

### 5.1.1.2. Physical phenomena

#### 5.1.1.2.1. Cladding oxidation and hydrogen formation

At temperatures above approximately 1300 K, the zircaloy in the cladding is exothermically oxidised by the steam. This reaction plays a major role in aggravating the core degradation, as the thermal power that it releases can become significantly higher than the residual power. The equation of this oxidation reaction is as follows:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

with an enthalpy reaction $\Delta H$ between – 600 and – 700 kJ/mole of zirconium and 0.0442 kg of hydrogen produced per kg of oxidised zirconium.

This oxidation produces a zirconia ($ZrO_2$) layer on the external surface of the cladding. The mass of oxygen absorbed by the cladding and the thickness of the oxide formed are governed by a parabolic time law. The square of the increase in the mass of oxygen fixed by the zirconium $\Delta M_0$ is proportional to the time interval $\Delta t$, that is to say:

$$(\Delta M_0)^2 = K_0(T)\Delta t$$

The reaction rate $K_0(T)$ varies as an exponential function of the inverse of temperature (Arrhenius law) and, at temperatures above 1600–1700 K, the energy supplied to the cladding by the reaction cannot be evacuated by convection with the steam; the reaction rate then increases rapidly, resulting in the cladding temperature exceeding the zirconium melting temperature (2100 K). Numerous experimental and theoretical studies have focused on this phenomenon, which is now well understood. The hydrogen produced can escape from the RCS (through a break, if there is one) into the containment building atmosphere; this results in an explosion risk regarding which the strength of the containment must be assessed. Knowing how to predict hydrogen production is therefore an important aspect of the safety studies, as we have already discussed in Section 4.3.

In the case of the 1300 MWe PWRs, the control rods are partly composed of boron carbide $B_4C$ (Section 2.3.2.1). This can also oxidise at temperatures above 1600 K, producing hydrogen. Little hydrogen is produced through this reaction, however, in comparison with the volume of hydrogen produced by the zirconium oxidation reaction. In the case of 900 MWe PWRs, the Ag-In-Cd alloy in the control rods does not oxidise.

#### 5.1.1.2.2. Meltdown of materials and interactions with the intact rods

The control rods melt at lower temperatures than the fuel rods, either through meltdown (the Ag-In-Cd alloy melts at temperatures above 1100 K) or, in the case of the 1300 MWe reactors, through a chemical reaction resulting in their liquefaction (with steel, the $B_4C$ forms a liquid eutectic mixture at temperatures above 1500 K). The $B_4C$ may also oxidise once the steel cladding and the zircaloy guide tube have disappeared. $B_4C$ oxidation is an exothermic process, effectively accelerating control rod degradation. It also produces hydrogen (always through steam decomposition, as for Zr oxidation), and part of the boron is in gaseous form ($HBO_2$).

After they melt, the control rod materials (including some steel) flow into the core and come into contact with the fuel rods, thereby weakening the cladding of those that are still intact through chemical interactions (forming eutectic liquids). It should be noted that the spacer grids made in Inconel may also react with the zirconium cladding. The major dissolution reactions include the Ag-Zr and Fe-Zr interactions, both of which form a eutectic liquid whose melting point is considerably lower than that of zircaloy. Experimental studies have been conducted on these interactions; the existing knowledge of these interactions and their modelling is satisfactory. Some uncertainties still remain regarding the influence of $B_4C$, however, as it seems to cause the cladding of the fuel rods to degrade at a lower temperature than that indicated by the models developed using the current state of knowledge.

### 5.1.1.2.3. Cladding failure

The increased fuel temperature and the formation of fission gas within the pellets increase fuel rod internal pressure. The zircaloy cladding begins to distort when the temperatures exceed 1000 K, due to the degradation of their mechanical properties. In some cases, the pressure inside the fuel rods can exceed the pressure inside the reactor vessel. This overpressure within the fuel rod causes the cladding to swell as a result of creep. This phenomenon, which is called "ballooning", can cause a mechanical failure in the cladding before they are oxidised. Some major distortion, referred to as "flowering", has also been observed. This is the result of the fuel pellets growing in volume, causing additional stresses in the cladding. There are sufficient experimental data on these phenomena, and their modelling is satisfactory.

During a core melt accident, not all of the fuel rod cladding suffers from mechanical failures before they oxidise. The oxidised cladding that has not failed mechanically may lose its integrity as a result of other mechanisms occurring at higher temperatures. These other mechanisms are much less well-known, however. The current hypotheses used to take them into account are based on experimental findings; consequently, it is accepted that the zirconia layer breaks above a certain temperature (typically around 2300 to 2500 K). Another mode of failure may occur when the thickness of the zirconia layer is less than a certain value (approximately 300 mm). The rupture mechanism involved is still poorly known, and it is modelled using a correlation deduced from the results of integral experimental programmes such as Phebus and CORA (Section 5.1.1.3.1), which use a rupture temperature that varies according to the thickness of the zirconia layer. In order to improve our understanding of the mechanism involved, it would be necessary to perform experiments that are both difficult and costly. Such experiments are not planned, as most users of the computer codes consider that the modelling described above is adequate for representing this mechanism in the computer codes used to simulate core melt accidents. It should nevertheless be remembered that the zirconia layer rupture criterion is a key parameter in these codes, as it defines the threshold for liquid zircaloy relocation towards the lower parts of the core.

### 5.1.1.2.4. Zircaloy melting and fuel dissolution

When the zircaloy melting point is reached, the $UO_2$ fuel is partially dissolved by the liquid metal (which does not flow out of the cladding as long as the zirconia layer remains intact). This may result in the mechanical integrity of the fuel rods being lost and the fragments produced in certain areas of the core accumulating long before the $UO_2$ melting point is reached (approximately 3100 K). The resulting fusion-dissolution, mechanical degradation and relocation of core materials within the reactor vessel (melts of molten materials and local accumulations of fragments) determine how the distribution of the degraded materials in the reactor core evolves during the course of the accident, and these must be taken into account in the modelling in order to realistically predict the degraded condition of the core. This can then be used to predict which areas are likely to be cooled if water is injected (reflooding) and which areas cannot be cooled because molten materials have accumulated, thereby preventing water from reaching them. Many experimental studies have been conducted in order to study changes in the distribution of the degraded materials in the core during the course of a core melt accident and considerable knowledge has been gained as a result, but the modelling is not yet satisfactory, undoubtedly due to the complexity of the phenomena involved. Despite the progress made (the development of mechanistic models based on detailed analyses of tests conducted on fuel rod clusters), some experimental results are still difficult to explain or interpret using the existing models, particularly the finding that fuel pellet dissolution exceeds the possible values based on the phase diagrams. It also remains difficult to model the simultaneous phenomena of fuel pellet dissolution and cladding oxidation. An ISTC (International Science and Technology Centre, a European Commission body) project named THOMAS, led by IBRAE (the Nuclear Safety Institute of the Russian Academy of Sciences in Moscow) resulted in the development of a model capable of computing the oxidation of a large corium pool (with natural convection processed in 2D or 3D) and the formation and dissolution of solid crusts at the edge of the pool.

### 5.1.1.2.5. Corium flow

The flow of the molten materials through the degraded core and their solidification in its colder areas may result in considerable localised reductions in coolant flow cross-sections (Figure 5.1 clearly shows this phenomenon), which directly affects coolant flow and the cooling of the degraded core. This flow depends on various factors including the viscosity of the molten mixture, which is a function of its oxidation degree. In the 2100–2900 K temperature range, the viscosity of a U-Zr-O mixture is an increasing function of the oxygen content. Knowing how to calculate the oxidation of such mixtures is thus particularly critical in determining the corium flow. The current understanding of this phenomenon is incomplete, notably because, in most of the experiments conducted (Phebus, CORA and PBF, described in Section 5.1.1.3.1), the corium globally flows in a single direction (a one-dimensional flow). It is likely, however, that corium radial flow would be just as important if not predominant[1] in the case of an accident affecting a reactor core. Various corium flow models provide partially satisfactory results, i.e. they

1. This is illustrated by the example of the Three Mile Island core melt accident, although this accident scenario is a special case (Section 7.1).

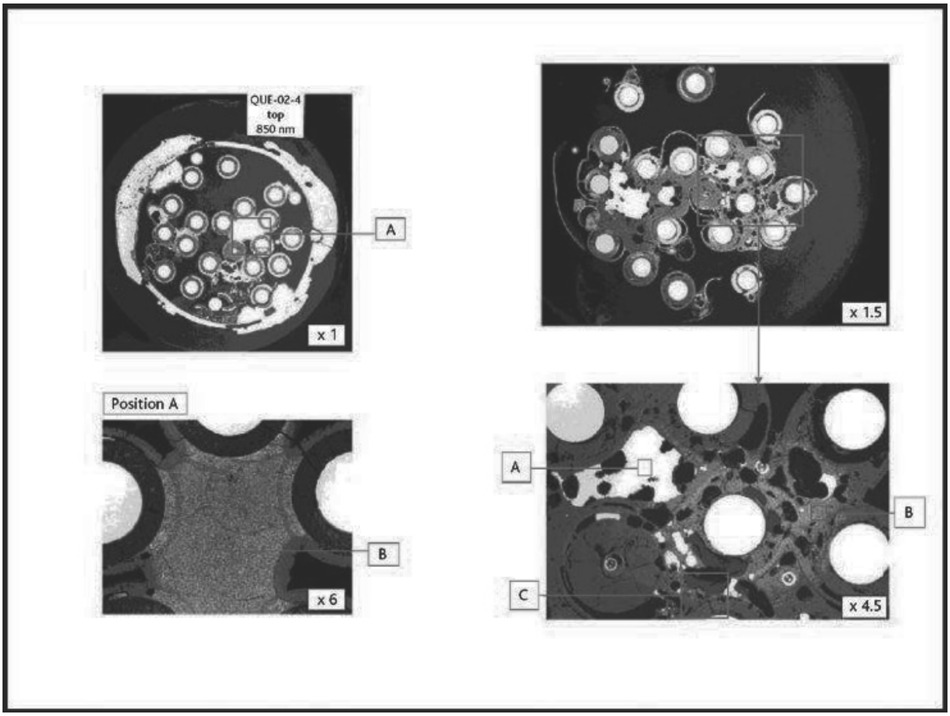



**Figure 5.1.** Photographs of two cross-sections of the fuel assembly after the QUENCH-02 test. These show the distribution of the solidified molten materials (melts) between the rods and the impact of these melts upon their degradation. The solidified melts are visible between the rods. They either totally block the spaces between the rods (position A, lower left) or partially block the spaces, leaving porosities (photograph, lower right).

generally predict the place in which the molten materials accumulate, but they are not capable of accurately predicting the resulting porosity (experimentally, it is found that the molten materials still do not occupy all of the available space). For the moment, however, it is not possible to improve them on the basis of existing experimental data. In the experiments conducted with corium melts, the corium flow is closely linked both with the localised temperatures reached in the tested fuel assemblies and with the degree of oxidation in the corium. These data cannot be determined sufficiently accurately from the experimental results or computed from the models, due to the complexity of the phenomena involved. Furthermore, in most of these experiments, corium progression is one-dimensional. The flow models have not been validated using sufficiently large-scale two- or three-dimensional test data. Uncertainties also remain regarding the physical properties of corium, notably its solidus and liquidus temperatures, as well as regarding the apparent viscosity (the viscosity of the liquid-solid mixture) where the solid and liquid phases are present simultaneously. These properties directly influence the corium flow.

### 5.1.1.2.6. Oxidation of the molten mixtures

When the corium flows through the core, it contains zircaloy that has not been completely oxidised. On contact with the steam, the zircaloy continues to oxidise. No experimental measurements of a U-Zr-O liquid mixture oxidation rate are available. However, during integral tests such as QUENCH (reflooding of an assembly of already-oxidised rods), substantial hydrogen production has been observed over a very brief period in the case of scenarios involving reflooding or a local increase in steam flow rate. This observation is particularly important when assessing the hydrogen explosion risk, as reflooding may lead to the instant hydrogen release rate within the containment temporarily exceeding the capacity of the hydrogen recombiners in the containment building concerned.

The QUENCH tests were the first that could be used to understand this effect, but they had two disadvantages. Firstly, the use of $ZrO_2$ pellets instead of $UO_2$ pellets forms a corium that is a Zr-O mixture instead of the U-Zr-O mixture that would be formed in a reactor core melt accident. Secondly, it is difficult to distinguish between the oxidation itself and the other phenomena (flow, cooling, etc.) in these integral tests; as a result, it is not possible to determine whether the materials oxidise while they are flowing or afterwards. The most likely explanation regarding the prolonged and intense oxidation of zircaloy is that of a relatively slow melt consisting of the very hot U-Zr-O liquid mixture (progressing at a speed of up to a few mm/s) along the fuel rods. The oxidation kinetics of such a mixture depend on the ability of steam to access the zircaloy and, therefore, on the porosity of the medium. The more the liquid mixture fills the open spaces, the lower the oxidation kinetics are. From this point of view, the zircaloy oxidation phenomenon is globally understood and most computer codes include models for calculating core melt progression in the reactor vessel. The validation of these models is still often very succinct, however. In particular, more analytical test results are needed; these could be used to determine the oxidation rates for U-Zr-O melt mixtures. The hypothesis of zirconia layer spalling in the event of reflooding (detaching zirconia layers from the rods, bringing the metal zircaloy into contact with the water or steam) does not provide a valid explanation for the intensified oxidation process while the rods are being reflooded. To date, no experimental results support the existence of such spalling in the case of the zircaloy 4 or the M5 alloy (there is very little experimental data for the latter), which are used as cladding materials in the PWRs. It has only been found in the case of alloys that are not used in the French PWRs, such as the E110 (Zr-Nb) used in the Russian PWRs (VVER).

### 5.1.1.2.7. Formation of a corium pool and corium flow into the lower head

If the temperature reaches the melting point of $UO_2$, a “molten pool” forms in the reactor core. Due to the formation of the eutectic liquids, the melting temperature may be several hundred degrees below that of the $UO_2$ melting point (3100 K).

As the eutectic molten mass increases, the pool expands axially and radially in the core until it reaches either the baffle or the core support plate (internal structure; see Section 2.3.2). At this moment, the corium flows into the lower head. Considering its low surface/volume ratio, the corium pool formed in this way is very difficult to cool; as a result, it may grow by incorporating the rods located around it, even if it is reflooded. This is what occurred in the Three Mile Island accident (Section 7.1).

It is essential to predict the weight, composition and temperature of the materials reaching the lower head during the course of the accident, as well as the instants when these materials do so, in order to study the subsequent sequence of events of the accident. These phenomena are modelled in most computer codes. Their level of validation and detail are satisfactory, considering the experimental data available. Only partial data is available, however, as they exist either for small-scale, virtually one-directional fuel rod assemblies (around twenty rods, in the case of the Phebus assemblies), or for preformed debris beds, which are also small (RASPLAV, ACRR and Phebus-FPT4; these programmes are described in Section 5.1.1.3.1 and Section 5.1.2.3.1). At present, there are no experimental data allowing detailed characterisation of corium pool formation and flow in the core. More representative data are still to be obtained in order to characterise the evolution of a corium pool through two-dimensional rod assemblies.

The degradation may ultimately result in very different configurations in the core simultaneously, ranging from intact or barely degraded rods to the formation of a corium pool or a bed of debris. These different degraded core conditions are described in greater detail in Section 5.4.1.

### 5.1.1.3. Experimental programmes, modelling and computer codes

#### 5.1.1.3.1. Experimental programmes

This section provides a brief description of the main experimental programmes, ranging from the oldest to those still under way or scheduled in 2015, to study the degradation of the core materials. The programmes performed have provided data for validating the computer codes. An OECD summary report presents all the tests whose results have been used to validate the core melt accident simulation computer codes [13].

**Separate effect tests on the oxidation kinetics of fuel rod materials and the associated chemical interactions:** many tests conducted by different teams including Forschungszentrum Karlsruhe (PzK) of Germany, and Atomic Energy of Canada Limited (AECL) of Canada have determined the oxidation kinetics of zircaloy, $UO_2$ dissolution by the molten zircaloy, $B_4C$ oxidation (FzK and IRSN), zircaloy dissolution by the molten steel, etc.

**Separate effect tests on cladding failure mechanisms:** these tests (for example, the EDGAR tests conducted by CEA) have helped to determine the cladding creep law based on the cladding temperatures and the oxidation conditions.

**LOFT-FP** [19]: this test programme, which was completed in 1985, was conducted by the Idaho National Laboratory (INL) of the United States on an assembly consisting of 121 $UO_2$ fuel rods with in-pile nuclear heating. It consisted of tests on fuel assembly degradation and FP release, and involved temperatures up to 2400 K (locally). Steam cooling was used, followed by water reflooding.

**PBF-SFD** [20]: this test programme, which was completed in 1985, was conducted by INL on an assembly consisting of 32 $UO_2$ fuel rods with in-pile nuclear heating. It also included tests on fuel assembly degradation and FP release, but at temperatures up to 2600–3100 K (locally). Steam cooling was used, followed by water reflooding (for certain tests).

**NRU-FLHT** [14]: this test programme, which was completed in 1987, was conducted by AECL on an assembly consisting of 16 non-irradiated $UO_2$ fuel rods with in-pile nuclear heating. These degradation tests were unusual because they used fuel rods 3.7 m high (full-scale).

**ACRR-MP** [8]: this test programme, which was completed in 1992, was conducted by the Sandia National Laboratory (SNL). It consisted of in-pile tests of small debris bed melting ($UO_2$ + $ZrO_2$) in an inert atmosphere, at temperatures up to 3000–3200 K. The formation and then flowing of a corium pool were observed.

**CORA** [17, 18]: this test programme, which was completed in 1993, was conducted by KIT (formerly FzK) on an assembly consisting of 25 non-irradiated $UO_2$ fuel rods with electric heating. It consisted of tests in which fuel rod temperature reached 2200 K (locally). Each test included a steam pre-oxidation phase, followed by reflooding with water or steam at a high flow rate.

**QUENCH** [21]: this test programme, which was still under way in 2013, was conducted by FzK on an assembly consisting of 25 non-irradiated $ZrO_2$ fuel rods with electric heating. It consisted of degradation tests and involved temperatures up to more than 2000 K (locally). Steam cooling was used, followed by water reflooding. Recent tests have studied the behaviour of cladding materials other than zircaloy-4, such as E110 or M5 (Zr-Nb alloys).

**Phebus FP** [5]: this test programme, which was completed in 2004, was conducted by IRSN on an assembly consisting of 21 irradiated $UO_2$ fuel rods with in-pile nuclear heating. It involved degradation and FP release tests with temperatures up to 2600–3100 K (locally). Steam cooling was used.

**ISTC 1648 (QUENCH)**: this test programme, funded by the International Science and Technology Centre (ISTC), was conducted by NIIAR (Scientific Research Institute of Atomic Reactors) in Russia. It aimed at studying the reflooding of a core that has reached temperatures exceeding 2000 K (locally), and performed three tasks: conducting degradation and reflooding tests using irradiated VVER fuel components, performing reflooding tests using a new VVER assembly of 31 rods, and developing a reflooding model for the SVECHA code by IBRAE (Nuclear Safety Institute of the Russian Academy of Sciences). The results of this programme have not been publicly published, but some reports are available by contacting ISTC.

**PARAMETER**: this test programme, funded by ISTC and launched by LUCH (Scientific Manufacturer Centre, Russia), studied the degradation of fuel assemblies consisting of 19 non-irradiated prototype fuel rods for VVER reactors (the tests were similar to those of the QUENCH project, but used $UO_2$ pellets instead). The experimental apparatus was used to reflood the system from above or below, at temperatures up to 2300 K at the moment it was reflooded. Three tests had been conducted by the end of 2009. After the apparatus had been considerably degraded in the first test, care was taken to ensure that the temperature did not exceed 1870 K at the hottest point during the preliminary oxidation phase in the subsequent tests, in order to preserve the integrity of the fuel assembly holder. A fourth test was conducted in 2010, with a preliminary oxidation phase in air intended to simulate the entry of air into the reactor vessel. This programme's experimental data have not been publicly published either, but some publications discuss the validation of computer codes using some results of these tests.

Few in-pile experimental programmes study the phenomena involved when core melts go so far as to form a debris bed or corium pool in the core and corium flows into the lower head, with the exception of Phebus FP and ACRR. The LOFT and PBF tests attained late-phase fuel degradation but did not involve detailed analysis of rod melt and corium progression.

The Three Mile Island accident remains the only available source of knowledge on the condition of a reactor core following massive melting (Section 7.1). This accident and the condition of the reactor core have been analysed in detail; the results have been published and are publicly available [4, 15, 26, 27]. Figure 7.7 shows the condition of the core following the accident. Of particular interest are the large molten pool within the core, the collapse of a large portion of the rods above the pool (forming a debris bed) and partial corium relocation towards the lower head. Two aspects of the accident scenario are worth noting: the high-pressure sequence, and the corium melt in the lower head after the core was, at least partially, reflooded.

#### 5.1.1.3.2. Core melt modelling and computer codes

This Section provides a brief description of the main dedicated models and computer codes used to simulate the phenomena of reactor core material degradation occurring in a core melt accident (it does not describe the integral computer codes used to process all of the phenomena involved in a reactor when a core melt accident occurs, as these are presented in Chapter 8).

**SCDAP/RELAP (US NRC, United States Nuclear Regulatory Commission)** is a mechanistic[2] computer code developed by INL. It is the result of coupling the RELAP 5 thermal-hydraulic code with the SCDAP core degradation modelling code. Its core modelling is based on parallel, one-dimensional channels and includes several models for simulating various aspects of fuel rod changes during the course of their degradation: heat transfers, residual power, cladding oxidation, fuel dissolution, cladding failures and FP release. This computer code is no longer developed by NRC but a private version known as SCDAPSIM is still developed [3, 6].

2. A "mechanistic" computer code consists of models that, whenever possible, are based on a physical or chemical description of the phenomena involved rather than adopting an empirical approach (based on correlations obtained using experimental results). In reality, mechanistic computer codes always include several empirical models.

**ATHLET-CD (GRS, Gesellschaft für Anlagen - und Reaktorsicherheit, Germany)** is a mechanistic computer code that is the result of coupling the ATHLET thermal-hydraulic code with a core degradation computer module. Much like SCDAP/RELAP, its core modelling is based on parallel, one-dimensional channels and includes several models for simulating various aspects of fuel rod changes during the course of their degradation: heat transfers, residual power, cladding oxidation, fuel dissolution, cladding failures and FP release. GRS is continuing to develop this computer code, notably with the addition of the MEWA module relating to the formation of a corium pool in the core [23, 25].

**ICARE/CATHARE (IRSN)** is a mechanistic computer code that simulates PWR core melt accidents. It is the result of coupling the CATHARE thermal-hydraulic code with the ICARE code that simulates core degradation and is similar to SCDAP/RELAP, except that considerable development has been carried out to simulate the phenomena involved when large-scale melting leads to significant degradation of the core (the formation of a debris bed or corium pool, and corium flows). It also allows axisymmetrical 2D modelling of the core and the reactor vessel. Several models simulate changes in the core fuel rods over time, as well as those occurring in the corium in the core and the lower head; they process heat transfers, residual power, cladding oxidation, fuel dissolution, cladding failures, FP release, 2D corium flow modelling, flow oxidation, fuel rod collapse and corium pool development. Development of this computer code is being continued by IRSN and includes a model of degraded core reflooding and complete modelling of corium behaviour in the lower head [7, 9].

**RATEG/SVECHA (IBRAE, Russia)** is a mechanistic computer code that is the result of coupling the RATEG thermal-hydraulic code with the SVECHA core degradation simulation computer module. This computer code includes highly detailed modelling of certain phenomena, notably cladding oxidation, fuel dissolution, cladding failure and FP release. The computer code is designed to describe in detail fuel assembly degradation (or that of a representative fuel rod). Its major limitation lies in its inability to process the radial propagation of degradation; notably, it does not process corium pool formation in the core, but it is able to simulate a molten pool in the lower plenum. IBRAE is continuing to develop this computer code, including the development of a module relating to corium flow oxidation [24].

### 5.1.1.4. Summary and outlook

The physics relating to the evolution of a PWR core melt accident is now well understood and modelled for the main processes. This notably concerns fuel rod cladding oxidation and failure. The complex phenomena involved in the later phases of the accident can only be modelled with significant uncertainties, however. This is particularly true of fuel rod collapse and corium oxidation. Additional experimental data would be needed in order to refine the modelling, but no related programmes are ongoing or planned in 2015. Given the high cost of the potential tests, which would have to be conducted with real irradiated materials, it seems unlikely that new experimental programmes (national or international) will be initiated in the near future. To reduce the remaining uncertainties, the only alternative is to perform additional analysis on past tests (which were often insufficiently investigated) and develop more detailed models.

## 5.1.2. Corium behaviour in the lower head

### 5.1.2.1. Introduction

In most of the expected cases, the lower head is filled with water when the corium coming from the hottest area of the core flows into it. The results of the fuel assembly melt tests in the Phebus, QUENCH and CORA programmes show that the zircaloy in the assemblies at the end of the tests is only partially oxidised. In the assemblies' hottest areas, 20 to 100% of the zircaloy is oxidised; total oxidation is only seen locally in the areas in which the temperatures and steam concentrations have been high enough to provide intense oxidation over a long period. The corium that flows in the lower head therefore contains a percentage of unoxidised zircaloy that is estimated to be 25 to 80% of the zircaloy (depending on the scenario, following the general trend that a large break sequence leads to a low percentage of oxidised Zr whereas a small break sequence leads to a high percentage of oxidised Zr). This corium is said to be substoechiometric while its composition is not $(U\text{-}Zr)O_2$, which corresponds to its composition after complete oxidation.

The interaction of corium (above 2500 K) with water leads to more or less fine-grained fragmentation of the corium into particles, on the one hand, and intense steam production capable of substantially increasing the pressure in the RCS, on the other hand. When the partially fragmented corium has accumulated in the lower head, it forms what is called a debris bed. This bed is either very compact if there is little cooling (part of the corium is not solidified), or composed of porous solid debris. It is unlikely that a large debris bed can be cooled effectively. In all cases, the corium gradually evaporates the water present in the lower head. If there is no additional water supply and the debris configuration is such that it cannot be cooled effectively, the materials' temperature gradually rises until it reaches the melting point of the steel structures (plates, tubes, etc.) located in the lower head. A substantial quantity of molten steel then gradually enters the corium. As the temperature rises, first the zircaloy and then the oxide debris melt and either form a pool or become part of an existing pool. The formation of this corium pool in the lower head is a critical step in a PWR core melt accident: in this situation, there is a considerable heat flux at the interface between the pool and the reactor vessel that can lead to reactor vessel failure. Reactor vessel failure is described in detail in Section 5.1.3.

### 5.1.2.2. Physical phenomena

When the hot corium pours into the lower head filled with water, steam is produced, leading to a pressure peak or even a steam explosion in the reactor vessel (Section 5.2.3), which creates mechanical stresses likely to damage the RCS. In addition, the reactor vessel is subjected to a heat flux that can locally be very high, resulting in melt erosion of the vessel walls and potentially leading to its failure. Regarding the last point, studies seek to determine the likelihood of in-reactor vessel corium retention or the conditions under which the reactor vessel would fail (timing, location, and characteristics of the corium flowing from the reactor vessel into the containment). It is thus important to be able to predict the changes the corium will undergo, from its relocation towards the lower head to its cooling or flow out of the reactor vessel. The main phenomena governing these changes are briefly described below.

### 5.1.2.2.1. Corium fragmentation and debris formation

When a corium melt comes into contact with the water present in the lower head, the corium becomes fragmented (Figure 5.2). Corium fragmentation is described in detail in Section 5.2.3 on steam explosions. The fragmentation is very complex to model and includes considerable uncertainties [34, 40].

### 5.1.2.2.2. Direct impact of a corium melt upon the reactor vessel

If there is little water in the lower head or there is a substantial mass of relocated corium, the corium melt only partially interacts with the water and part of this very hot melt comes into direct contact with the reactor vessel.

This situation can rapidly lead to reactor vessel failure during its period of contact with the corium melt. Although few experimental studies have been conducted on this phenomenon (a few CORVIS tests with molten materials simulating a corium, notably a molten thermite mixture [a mixture of iron and alumina], arriving on a mock-up of a BWR vessel), this phenomenon is relatively well-known. In such a situation, it would probably form an insulating — and therefore protective — crust between the corium and the reactor vessel, given the very large difference between the temperature of the reactor vessel and the corium solidification temperature. In the TMI-2 accident (Section 7.1), such a crust was probably formed; this would explain why, although a massive corium melt flowed into the lower head, the vessel was not damaged[3]. One of the key parameters is the degree of corium overheating above its melting point; this degree directly influences the thickness and, therefore, the efficiency of the protective crust.

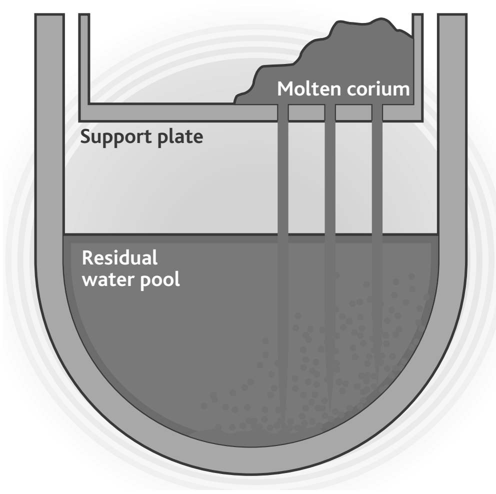



**Figure 5.2.** Schematic diagram of molten corium arriving in the lower head and fragmenting upon contact with water.

3. In the case of TMI-2, the presence of water in the lower head undoubtedly also contributed to the efficient cooling of the reactor vessel during the corium melt.

### 5.1.2.2.3. Steam explosion

The interaction between a corium melt and the water can result in a steam explosion (Section 5.2.3). Although the physics of this phenomenon are generally understood, it is not yet possible to predict with certainty under which conditions this phenomenon can occur. Despite the relatively low probability of an explosion occurring that had been observed in the tests conducted in order to study the interaction between very hot corium and water, such an explosion cannot be entirely ruled out and so the phenomenon is studied because of its possible consequences for the containment. It should be noted that in the TMI-2 accident, corium melt arrival did not lead to a steam explosion despite the presence of water in the lower head. This could indicate that there was no fine-grained fragmentation of the corium on contact with the water. It could also be due to the high pressure in the reactor vessel (approximately 100 bar).

### 5.1.2.2.4. Debris bed dry-out and possible reflooding

Corium melt fragmentation produces corium drops that cool and solidify on contact with the water, forming particles that settle in the lower head and create a "debris bed". This debris bed may be very compact if the cooling of the corium drops resulting from the fragmentation is insufficient to solidify them totally. In this case, the formed debris bed cannot be cooled effectively because the water cannot access some regions of the debris bed due to its low permeability. The debris bed then continues to heat up, gradually drying up the lower head and then melting to form a corium pool that is much more difficult to cool. The final risk is that the corium pool builds up and comes into contact with the reactor vessel and causes it to rupture. The possibility of avoiding the lower head drying and cooling down such a debris bed with the water present in the lower head or with an additional injection of water from the RCS is therefore under study (Section 5.4.1).

One of the parameters currently used to estimate the possibilities of cooling a debris bed is the "critical dry-out heat flux" (CHF), which corresponds to the maximum residual power density of the debris bed multiplied by the height of the bed within which it does not lead to a local dry-out. Below the CHF, the water is present everywhere in the debris bed and the temperature of the debris bed remains low. The CHF depends on the characteristic parameters (debris size, bed geometry and porosity, etc.) of the debris bad. Typical values are approximately 0.2 $\text{MW/m}_2$ for particles with a diameter of 1 mm and 1.2 $\text{MW/m}_2$ for particles with a diameter of 7 mm.

In a core melt accident, water may re-enter the reactor vessel whereas the debris bed is partly or totally dry. In such a situation, reflooding the debris bed may produce a large quantity of steam in a very short time, which can rapidly build up the pressure in the RCS and cause major oxidation of the unoxidised zircaloy in the hot upper parts of the core to resume. Few studies have been conducted on the phenomenology of debris bed reflooding and research programmes are still studying the subject in 2015 (notably the PEARL test programme conducted by IRSN).

### 5.1.2.2.5. Corium pool formation

As mentioned above, the dry-out of a large volume of a debris bed is a key step in the evolution of a core melt accident because it determines when corium pool formation begins or, if only a portion of the corium is fragmented into solid particles, when the propagation of the existing pool begins. Thanks to the results of the ACRR-MP, Phebus FPT4 and RASPLAV AW-200 tests [39, 32, 28], the molten pool formation is now quite well modelled when it occurs under conditions that do not result in significant corium oxidation and the main components of the corium are $UO_2$, Zr and $ZrO_2$. The pool may also contain a large quantity of molten steel. Although the interactions between the liquid steel and a (U-Zr)-O corium have been studied for some time, the effect of these interactions upon the evolution of the corium in the lower head (see later in this document) when the corium is in inert atmosphere was only revealed at the beginning of the 2000s, notably in the OECD's MASCA project [37]. The evolution of a debris bed containing steel under oxidising conditions requires further study. This is because the residual power causes the steam to circulate within the debris, which is likely to oxidise as a result (Figure 5.3). The melting of the debris and the development of a corium pool under oxidising conditions have never been studied experimentally, notably because of the high cost of the tests, which would require the use of real materials. The uncertainties regarding the degree of oxidation of the materials in the formed corium pool are processed in the core melt accident simulation computer codes by means of sensitivity studies.

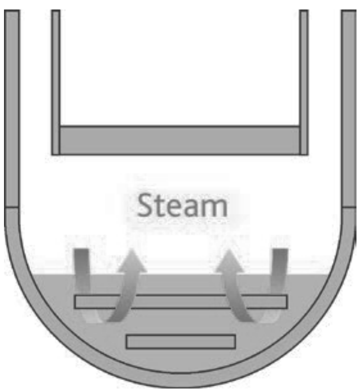



**Figure 5.3.** Schematic diagram of corium configuration after the lower head has dried out: a debris bed (which is more of less porous, depending on the degree of fragmentation of the corium) around steel structures, with steam circulating through natural convection.

### 5.1.2.2.6. Convection movements in the corium pool

The power released by the corium pool can be evacuated both through its lateral edges (thus *via* the vessel wall) and through its upper surface (through convective exchange with any water present, or through radiative transfer). These heat transfers cause natural convection movements of the molten materials in the corium pool (Figure 5.4). One of the key parameters when taking into account this phenomenon is the relationship between the upward heat flux and the lateral flux evacuated through

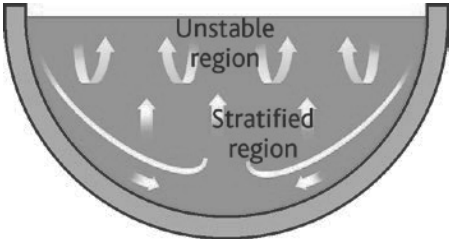



**Figure 5.4.** Schematic diagram of convection movements in a turbulent corium pool with top-down and lateral cooling. These movements cause the corium to flow downwards along the reactor vessel then rise (slowly) in the centre. The top of the pool is the site of considerable agitation in the form of thermoconvective cells (Rayleigh-Besnard instability).

the vessel wall. The movements in the corium pool are mainly turbulent, except in some highly stratified regions at temperatures in which there is almost no convection (the bottom part, for example). This phenomenon is relatively well understood for simple corium pool configurations, and correlations have been established for heat exchanges at the edges of the pool (see Section 5.4.1.1 and the reference documents [33, 41]).

### 5.1.2.2.7. Corium oxidation (in the form of particles or a pool), and hydrogen production

When the corium is being fragmented, it may be oxidised. This oxidation, if it occurs, produces hydrogen, on the one hand, and determines the later evolution of the corium, on the other hand. The ZREX/ZRSS tests (by Sandia National Laboratory, with a Zr + $ZrO_2$ or Zr-stainless steel mixture) and CCM tests (by Argonne National Laboratory, with a $UO_2$ + $ZrO_2$ corium mixture containing 24% steel) have provided partial information on corium oxidation. These tests suggest that, in the absence of a steam explosion, the fragmentation is not fine enough to result in significant debris oxidation. Nevertheless, tests with water at saturation have led to the oxidation of up to 30% of the metallic masses present in the corium. In the event of a steam explosion, oxidation may be complete. Not enough tests have been conducted to allow sufficient quantification of this phenomenon (because of the risks involved).

As to corium pool oxidation, this phenomenon has not been extensively studied and models are inadequate as a result in 2015. The MASCA-2 programme tests (on the evolution of a stratified pool under oxidising conditions, see reference document [37]) have provided some information on this subject, but not enough to allow the oxidation kinetics to be measured; in addition, it is very difficult to extrapolate their small-scale data to the scale of a power reactor lower head.

### 5.1.2.2.8. Metal/oxide stratification in the corium pool

The MASCA MA and STFM tests [37], which were conducted at high temperatures with a corium containing uranium, zirconium and iron in the form of metals and oxides, have revealed the existence of two immiscible liquid phases at equilibrium, one metallic and the other consisting of oxides. Depending on the initial composition of the mixture, the metallic phase, consisting mainly of steel, may contain uranium and zirconium and become denser than the oxide phase. This results in pool stratification with the metallic phase at the bottom of the reactor vessel (Figure 5.5). Phase composition at equilibrium can be predicted using thermochemical databases such as NUCLEA (developed by Thermodata for IRSN and CEA). Pool stratification depending on the evolution of density over time is rarely modelled in the computer codes used to simulate core melt accidents, however. Although stratification of two immiscible liquids is a known phenomenon, the coupled interaction between mass exchange (thermochemistry) and flow dynamics (natural convection and stratification) remains a difficult process to model. In 2015, some computer codes incorporate simplified modelling of the oxide and metallic layers' evolution based on changes in their density.

The challenge lies in being able to predict under what conditions the molten metal layer is lighter than the oxide layer, resulting in the heat flux in the reactor vessel being "concentrated" in the metallic layer (when it is thinner than approximately 50 cm); this phenomenon is named the *"focusing effect"*. In the initial studies on corium retention in the reactor vessel, which adopted a "conventional" approach (for the Westinghouse AP600 reactors, for example), the metal was supposed to merely contain steel and so be lighter than the oxide. The heat flux transferred to vessel wall is then higher in the metallic layer, particularly when it is thin: as a first-order approximation, the heat flux transferred to the vessel wall is inversely proportional to the thickness of the molten metal layer thickness. A thin molten metal layer above a corium pool therefore has the effect of "concentrating" the heat delivered to the wall. This phenomenon, which is rather well understood and modelled [44], is one of the main threats to reactor vessel integrity. It is explained in detail in Section 5.4.1.1.

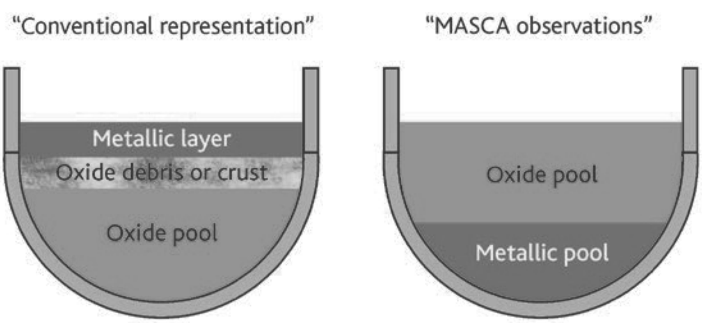



**Figure 5.5.** Layout of the potential metallic, oxide and debris layers resulting from corium fragmentation, as supposed in the "conventional" approach (left) and as observed in the MASCA tests (right).

#### 5.1.2.2.9. Dissolution of reactor vessel steel at temperatures below its melting point

As a result of the formation of eutectic mixtures (Fe-U-Zr), reactor vessel steel may dissolve at temperatures above 1360 K. This can lead to erosion of vessel steel if it is in contact with a corium containing uranium oxide, zirconia and zirconium. The

METCOR tests (studying the interaction between a high-temperature corium containing uranium, zirconium and oxygen, and a steel sample representing a reactor vessel) have made it possible to estimate the erosion kinetics of reactor vessel steel, but the need remains for a more detailed understanding of the process. However, the rate of creep above 1300 K is such that the steel no longer has any mechanical strength at such temperatures (Section 5.1.3). This phenomenon can therefore be considered of secondary importance.

### 5.1.2.3. Experimental programmes, modelling and computer codes

#### 5.1.2.3.1. Experimental programmes

This section provides a brief description of the main experimental programmes dedicated to the behaviour of corium in the lower head, ranging from the oldest to those still under way or planned in 2015.

**DEBRIS** [43]: the purpose of this test programme, conducted by the University of Stuttgart Institute of Research (IKE) in Germany, is to measure pressure losses and cooling (caused by water flow) for two-phase flows in a heated debris bed. The experimental system is one-dimensional and consists of steel balls heated by induction. Initially, a set of measurements taken for an isothermal water-air flow through the bed of balls was used to determine the two-phase pressure losses, as a knowledge of these values is essential in predicting the “critical dry-out heat flux” (CHF). Since 2008, the experimental system has been modified in order to perform debris bed reflooding tests. As the preliminary tests were satisfactory, more quantitative tests have been performed since 2011 to measure debris bed cooling through reflooding.

**SILFIDE** [29]: the purpose of this test programme conducted by EDF and completed in 2000 was to measure the CHF of a debris bed heated within its volume. The experimental system was two-dimensional and therefore differed from the DEBRIS test programme. The debris bed consisted of steel balls heated by induction. Useful results were obtained despite the challenge of establishing a homogeneous power distribution within the balls. In particular, local fluxes were sometimes observed to be higher than the theoretical critical flux (for 3 mm particles, the maximum flux measured in the SILFIDE tests was 1.7 $MW/m^2$ instead of the approximately 1 $MW/m^2$ predicted by the Lipinski correlation). Researchers also observed temporary, localised dry-out prior to reflooding.

**RASPLAV** [28]: this experimental programme, which was completed in 2000, was conducted under the auspices of the OECD by the Kurchatov Institute of Moscow in Russia. IRSN, CEA and EDF also participated in this project. Its purpose was to study the two-dimensional thermal hydraulics of a corium pool composed of “real” materials (the corium was composed of $UO_2$, $ZrO_2$ and Zr). The tests, involving up to 200 kg of corium, produced heat fluxes in accordance with predictions using the correlations developed from tests with simulants. However, it was revealed that the interactions between the materials could result in a non-homogeneous corium composition, notably due to stratification, but this phenomenon is minor compared with the stratification observed in the presence of iron during the MASCA tests (see below).

**MASCA**: this experimental programme, which was completed in 2006, was conducted under the auspices of the OECD by the Kurchatov Institute of Moscow in Russia. IRSN, CEA and EDF also participated in this project. The experimental facilities used for the MASCA programme were found to provide useful results on material interactions and their impact upon heat flux distribution in a corium pool. The MASCA experiments studied how material interactions affected stratification of the corium pool and, consequently, flows and heat exchanges at the edges of the pool. The main tests were used to study the addition of steel, fission products or $B_4C$ to a corium pool composed of $UO_2$, $ZrO_2$ and Zr. At the same time, certain thermophysical properties of metallic alloys composed of uranium, zirconium and iron or oxides were measured, such as their density, viscosity, and solidus and liquidus temperatures. The programme's second phase was aimed at studying the evolution of a stratified corium pool in an oxidising atmosphere.

**SIMECO** [45]: this experimental programme, which was completed in 2009, was conducted by the Royal Institute of Technology (RIT) in Stockholm, Sweden. Its purpose was to study the heat fluxes in a stratified pool in which a thermal power was generated. In tests using simulants (salts or paraffins), three-layer pool configurations were produced, consisting of a heavy "metallic" layer, an "oxide" layer where most of the power was dissipated and a light "metallic" layer. This allowed the heat flux distribution across the corium pool to be measured. In 2015, the results have yet to be interpreted in greater detail, but it already seems that they will help to modify the distribution estimated on the basis of conventional correlations.

**METCOR**: this experimental programme by the International Science and Technology Centre (ISTC), which was completed in 2009, was conducted by the Alexandrov Scientific Research Technological Institute (NITI) in Saint Petersburg, Russia. Its purpose was to study the erosion of a steel sample representing the reactor vessel by a corium ($UO_2 + ZrO_2 +$ Zr). The sample was externally cooled and subjected to a heat flux representative of conditions in a large corium pool, with a temperature gradient of over 1000 K across the sample [30]. The results of this programme seem to show that the erosion does not weaken the reactor vessel, as its mechanical strength mainly depends on the profile of temperatures in the wall under core melt accident conditions.

**LIVE**: this experimental programme, which began in 2004 and still under way in 2015, is conducted by Forschungszentrum Karlsruhe GmbH Technik und Umwelt (FzK) in Germany, with support from the European Commission. Its purpose is to study the behaviour of a corium using simulants in a hemispherical lower head (approximately 1 m in diameter). The chosen simulant is a mixture of $NaNO_3$ and $KNO_3$. The first test studied the steady-state thermal hydraulics of the pool (distribution of thermal fluxes at the wall). The second studied the corium melt and its spread in the lower head, with the formation of a crust through solidification. Other tests performed between 2011 and 2013 have studied the effect of stratification, the melting of debris and the effect of top cooling. Those tests were partially funded by the European network of excellence SARNET-2, with support from the European Commission. These new tests have completed our knowledge of the temperature at the solid-liquid interface and crust stability.

**INVECOR**: this experimental programme supported by the European Commission in the context of the ISTC was conducted between 2006 and 2010 by IAE-NNC-RK (Kazakhstan). Its purpose was to study the interactions between a liquid corium ($UO_2$ + $ZrO_2$ + Zr) and a steel hemispherical lower head approximately 80 cm in diameter by maintaining a constant power density using electrodes inserted into the corium pool. Four tests were carried out. Each test used 60 kg of corium, which was poured into the reactor vessel mock-up then heated, and then cooled using water. The results are quite difficult to interpret because of the presence of the electrodes, which have a considerable influence upon convection in the pool as well as its cooling. The results are primarily qualitative. They reveal that the upper layer of the corium pool is fragmented, encouraging its cooling. It therefore seems that reflooding the lower head, even after a corium melt, are beneficial in retaining the corium in the reactor vessel (in addition to external cooling of the reactor vessel).

### 5.1.2.3.2. Models and computer codes

This section provides a brief description of the main models and dedicated computer codes used to simulate the behaviour of a corium pool and its interactions with the lower head (the integral computer codes used to simulate core melt accidents, which are presented in Chapter 8, are not presented here).

**CFD computer codes**: these computer codes solve Navier-Stokes equations for compressible or incompressible fluids in any geometry (2D or 3D). These include the FLUENT and CFX codes, which were both developed by ANSYS and are used for many industrial applications involving 3D flows. These computer codes generally use numerical resolution methods that are efficient and fast, and their user interfaces are designed for ease of use. Offering many optional models (turbulence, material transfers and chemistry), they are increasingly used to study pools of molten materials. They are intended for rather generic applications, however, and may prove of limited use for modelling a particular phenomenon (such as solid particle formation and stratification, for example).

**MC3D (CEA/IRSN)**: this mechanistic computer code simulates in detail the interactions between a corium and water (fragmentation and steam explosion). It is described in Section 5.2.1 [31].

**CONV 2D/3D (IBRAE)**: this code solves Navier-Stokes equations for incompressible fluids, regardless of geometry (2D or 3D). It can be used to calculate the evolution of a corium pool and its spread outside the reactor vessel, and is similar to a CFD code. It does not have a turbulence model (essential when simulating large pools) or a model for processing the chemical interactions within the corium (no material transfers or chemical kinetics), however [35]. It was used in the preparation of the RASPLAV and INVECOR tests.

**TOLBIAC (CEA)**: this is a dedicated model for simulating corium pools in the lower head. It takes into account the existence of two immiscible liquids that can stratify in either direction; it also accounts for potential crust formation on the upper surface of the pool or along its edges. It can be used to calculate transient changes in axisymmetrical 2D domains [46].

**SURCOUF (CEA/IRSN joint development for the ASTEC integral computer code)**: this module, which was developed for the ASTEC integral computer code (see Chapter 8) is designed to model the evolution of debris in the lower head by integrating the coupled interaction between thermochemistry and thermal hydraulics. The 0D approach accounts for the existence of several layers (light metal, heavy metal, oxide and solid debris) and can be used to calculate their respective positions, based on changes in density. This code has been replaced by the PROCOR code which is significantly improved. PROCOR is also used by EDF, with a coupling to MAAP.

**ICARE/CATHARE (IRSN)**: this mechanistic software calculates core degradation under core melt accident conditions. It offers axisymmetrical 2D modelling of the reactor vessel and includes several models designed to simulate the behaviour of corium in the lower head: corium melt fragmentation, debris bed dry-out, debris melting, metal/oxide stratification, corium oxidation and debris reflooding. The lower plenum mesh is still rather crude, however, and the numerical methods used do not provide as much accuracy as the CFD models [36, 38, 42]. The lack of precision in the mesh is nevertheless acceptable, given the uncertainties regarding the properties of materials or some physical phenomena.

### 5.1.2.4. Summary and outlook

There are still many uncertainties in the description of corium behaviour in the lower head. Firstly, the effects of the material interactions (stratification, oxidation and dissolution) seem very important and are not all properly modelled yet (notably because the experimental results are very recent); this should be improved by analysing the latest results, obtaining more experimental data and developing more advanced models (that, among other things, address the problem that thermodynamic equilibria are not currently processed at the local (mesh) scale). Secondly, the effects of scale are difficult to estimate, and it is sometimes tricky to transpose reduced-scale test results to a full-sized power reactor. Further analysis and modelling is required to make this transposition possible by reducing these uncertainties, considering the fact that it is hardly feasible to perform full-scale tests.

## *5.1.3. Reactor vessel failure*

### 5.1.3.1. Introduction

When a core melt accident occurs in a PWR, the integrity of the reactor vessel may be threatened by three main phenomena. The corium flowing into the lower head may erode the reactor vessel immediately upon direct contact, or the reactor vessel may be damaged by a potential steam explosion immediately the corium comes into contact with the liquid water present; if the reactor vessel withstands this transient phase, its integrity may then be threatened by the effect of a corium pool forming in the lower head.

The reactor vessel is more intensely eroded if the volume of the corium melt is large, or if the water present in the lower head is shallow. In theory, this can result in the reactor vessel very rapidly failing on contact with the melt. Some experiments have shown that a crust forms between the melts and the molten metal, substantially slowing the rate of erosion [47]. If the temperature of the corium in the melts is higher than 2500 K, however, this insulating crust may not form (Section 5.1.2). Other factors probably reduce the degree of erosion, such as the point of contact of the melt rapidly changing over time, resulting in a very short contact time for a given point in the reactor vessel and the presence of water in the lower head.

When a corium melt and water come into contact, this can also very rapidly produce a large quantity of steam, resulting in a very high pressure peak and possibly a steam explosion capable of damaging the reactor vessel (see Section 5.2.3 and references [48, 49]).

Should molten corium form a pool in the lower head, heat exchange between the pool and the reactor vessel may provoke localised, partial melting of the reactor vessel, possibly resulting in reactor vessel rupture. This heat exchange is even greater for high-mass corium pools. Nevertheless, reactor vessel failure does not occur in all cases, as the Three Mile Island-2 accident showed in 1979 (see Section 7.1 and references [50, 51]). When this accident occurred, the reactor vessel remained intact even though a corium pool formed in the lower head. Subsequent analysis concluded that 1) the corium debris was porous, allowing some cooling, and 2) a gap existed between the pool and the inner surface of the reactor vessel. The gap is believed to have allowed water or steam to circulate. It should also be noted that high pressure in the primary coolant system may have a favourable impact on corium cooling when the corium melts (increased critical flux and reactor vessel deformation from creep or plasticity, potentially enlarging the gap).

It should lastly be noted that the reactor vessels of operational PWRs are equipped with a number of guide thimble passageways (also called "penetrations") for insertion of instruments to measure the neutron flux in the reactor core. Reactor vessel rupture may be initiated in the zones around these passageways, due to the presence of singularities and welds. If guide thimble passageways fail in the reactor vessel (by melting, for example), water, steam, fission products and corium may leave the reactor vessel *via* the interior of these guide thimbles.

### 5.1.3.2. Physical phenomena

This document only describes the physical phenomena involved in the case of a corium pool in the lower head resulting in the reactor vessel failing. Three parameters must be determined that are important in the later sequence of accident events outside the reactor vessel: the moment when the reactor vessel fails, the location of the break in the lower head, and the size of the break.

The moment when the reactor vessel fails mainly depends on the pressure of the reactor coolant system (RCS) and reactor vessel temperature (linked with the mass and configuration of the corium pool). RCS pressure is generally uniform throughout the reactor vessel, but it may rapidly increase if water is injected into the reactor vessel.

Reactor vessel temperature is closely linked with the heat flux evacuated through its thick walls.

The location of the break mainly depends on the temperature distribution within the reactor vessel. The area that has been subjected to the greatest heat is the most likely to fail first, excluding singularities and welds; the other sensitive areas are those in which the thickness of the reactor vessel may have been eroded by corium melts, as well as those with singularities due to the presence of the guide thimble passageways and their welds.

Reactor vessel failure may be triggered either by plastic instability or by creep. Plastic instability occurs when the membrane stress acting on the thickness of the reactor vessel exceeds the ultimate tensile strength of the steel, which decreases considerably at higher temperatures. Creep, however, generally occurs at temperatures above 800 K. When the temperature rises throughout the thickness of the reactor vessel, creep may occur even if pressure levels remain low.

Once the reactor vessel begins to crack, the cracking spreads; the final size of the break greatly depends on the method of propagation, and this is directly related to the metallurgical characteristics of the reactor vessels' steels (see later in this document). Differences in chemical composition (even regarding trace elements) can change reactor vessel behaviour at high temperatures; the failure may be either brittle or ductile. Tests conducted on reactor vessel mock-ups [52, 53] have shown that if two materials behave differently at high temperatures (hot shortness vs. ductility), the final break sizes will also differ considerably.

### 5.1.3.3. Experimental programmes, modelling and computer code

In the context of experimental research on lower head behaviour, CEA conducted the RUPTHER programme [60] from 1995 to 1999 in collaboration with EDF and FRAMATOME. Its objective was to determine the tensile and creep properties (between 300 K and 1600 K) of 16MND5 grade steel (the steel used for French PWR reactor vessels) and model the mechanical behaviour of a PWR reactor vessel subjected to accident loads. The specimens used for the validation tests consisted of cylindrical tubes that were subjected to internal pressure and heated to very high temperatures (between 1000 K and 1600 K). The programme revealed certain deficiencies (both in the modelling and in the mechanical characterisation of 16MND5 grade steel). There were also other difficulties, mainly related to the metallurgical complexity of the steel (the effect of elements present in the steel, even at low levels, notably sulphur). The results have also shown that the metallurgical properties of the steel greatly affect its rupture behaviour. Additional programmes were subsequently carried out.

These included two experimental programmes entitled "*Lower Head Failure*" (LHF, 1994-1999) and "*OECD Lower Head Failure*" (OLHF, 1999-2002), which were carried out by the US Sandia National Laboratories (SNL) to study the strength of reactor vessels produced in US steel (SA533B1) subjected to complex thermomechanical loads representative of those resulting from the presence of a corium pool in the lower head [52, 53].

The second of these programmes, which was an extension of the first, was led by OECD. The LHF programme involved eight tests, and the OLHF programme involved four. Although the same type of 1/5th scale mock-up was used for both programmes, the wall thickness was doubled for the OLHF programme in order to study the impact of the temperature gradient across the reactor vessel wall. Several methods were used to heat the mock-up in the LHF tests, namely a superheated azimuthal band (representing a corium pool in the lower head with maximum heat flux at the free surface of the pool), superheating a localised zone (representing a lower head hotspot) and finally, uniform heating throughout the lower head area. The experimental protocol called for increasing the temperature at a constant rate until the mock-up failed. The LHF tests were carried out under constant pressure (seven tests at 100 bar and one test at 50 bar). Two of them studied the behaviour of the guide thimble passageways. In the case of the OLHF tests, only uniform heating was applied (Figure 5.6) and two pressures were applied: 50 and 100 bar. One OLHF programme test studied the influence of a rapid pressure increase from 50 to 100 bar upon reactor vessel failure mode. Another test studied the behaviour of the guide thimble passageways (at a pressure of 50 bar). During tests with guide thimble passageways, weld leaks generally occurred, resulting in the experiments being terminated before the lower head actually failed.

The FOREVER tests [58, 59] were carried out between 1999 and 2002 by RIT (Royal Institute of Technology in Stockholm, Sweden). These tests used 1/10$^{th}$ scale mock-ups of a PWR reactor vessel in 16MND5 steel. The experimental protocol consisted in pouring a mixed oxide melt (30% by weight of CaO; 70% by weight of $B_2O_3$) into the reactor vessel simulating the corium at a temperature of approximately 1500 K. This melt was then maintained at approximately this temperature, and the reactor vessel was then subjected to a pressure of 25 bar until it failed.

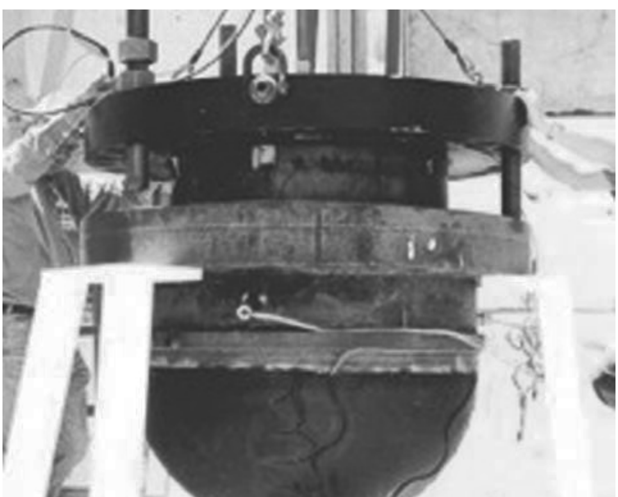


**Figure 5.6.** Lower head mock-up for carrying out the 1/5$^{th}$ scale OLHF tests, and setting up of its internal induction heating system.

In these three series of tests, particular attention was paid to the instant of reactor vessel failure and its mode, as well as the size of the resulting breaks. These tests were used to develop and validate numerical models for the thermomechanical behaviour of a PWR lower head under pressure prior to its failure. The models developed in this way are briefly described below:

- IRSN developed two simplified models, one one-dimensional (1D) and one two-dimensional (2D): the 2D simplified model has been introduced into the ICARE-CATHARE and ASTEC [54] computer codes;
- models with 2D finite elements have been developed by the Association Vinçotte Nucléaire (AVN: the Samcef code), the French Alternative Energies and Atomic Energy Commission (CEA: the Cast3M code), Electricité de France (EDF: the Aster code), the German laboratories Forschungszentrum Dresden Rossendorf (FZD: the Ansys code) and Gesellschaft für Anlagen- und Reaktorsicherheit (GRS: the Adina code), the US Sandia National Laboratories (SNL: the Abaqus code), the Czech Republic's UJV (the Systus code), and the Finnish VTT technical research centre's Pasula code;
- 3D finite-element models have been developed by AVN, CEA and SNL.

Two successive comparison exercises were carried out in order to compare the results of the 1D and 2D models with the experimental results of the OLHF1 test. The first exercise was carried out as part of the OLHF project, and the second was carried out by the European SARNET (Severe Accident Research NETwork of excellence) [55, 56]. These have established that the instant and location of the failure were generally accurately predicted by the models. Figure 5.7 shows that the elongation in lower head steel estimated by the different numerical models for the OLHF1 test is also consistent with the experimental results.

The 3D models also determined an initial failure time and initiating breaking zone that are compatible with the experimental results [56]. Additional work by CEA on the Cast3m code for the OLHF1 test produced a crack propagation simulation and an

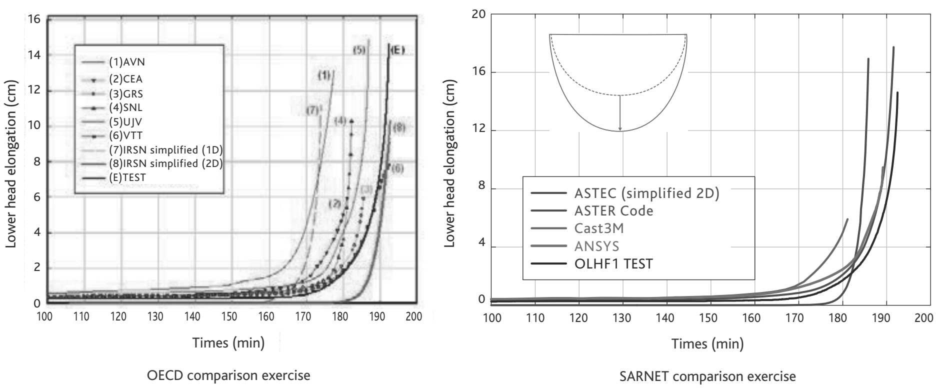



**Figure 5.7.** Comparison between the final elongation in the lower head steel estimated by different numerical models and the experimental results of the OLHF1 test.

estimate of the final break size that were both totally in keeping with the experimental results. The 3D models, on the other hand, did not give satisfactory results for the LHF tests, and this has been attributed to the variability of the steels used in the tests (ductile steels were used for the OLHF tests, whereas steels that were brittle at around 1300 K were used for the LHF tests).

The following conclusions were drawn from the analysis and interpretation of the test results [57]:

- the LHF and OLHF tests revealed variations in the behaviour of reactor vessel steels (brittle or ductile) at around 1300 K, influencing the final break size and difficulty in integrating the results into existing numerical models. This variability seems to be strongly linked to the presence of certain elements in the steels (sulphur, aluminium nitride, etc.);
- the experimental results could not be used to develop a method of estimating the size of the break in function of the mechanical loads applied to the reactor vessel. In order to develop a method applicable to the power reactors, it seems necessary to use a 3D finite-element calculation with a failure criterion that takes into account the variability of the behaviour of the studied steels, notably in the LHF and OLHF tests.

In order to clarify the variability in reactor vessel steel behaviour, IRSN launched a research programme in collaboration with CEA and INSA Lyon in 2003 [61]. This programme focused on the steels used in French reactor vessels and had a twofold objective: to complete the characterisation database for these steels, and to apply the study results to French reactors.

The programme began with an inventory of the metallurgical properties and compositions of the steels used to manufacture French reactor vessels (carried out by AREVA NP) and then turned to the selection of five study materials with sufficiently different metallurgical and mechanical properties to cover the range of steels used.

Samples of the five materials were then heated to a temperature of around 1300 K in order to identify their behaviour (brittle or ductile); these tests confirmed the brittleness of certain steels (ductility trough). Identification of the metallurgical factors responsible for this brittleness under heating also revealed aluminium nitrate precipitates and manganese sulphide precipitates at the grain boundaries and provided an insight into their role. Concurrently, high-temperature characterisation tests (at between 1200 and 1300 K) were carried out on CT (compact tension) specimens to determine the reactor vessel steels' metallurgical and mechanical properties that define the crack propagation kinetics. The results of these tests were used to develop a crack propagation model [62]. Lastly, INSA Lyon conducted tests on the steel tubes at high temperatures to measure the dependence of crack propagation kinetics on the properties of the tested steel. The reference documents present the state of knowledge in this R&D programme [63, 67].

Several theoretical studies and the CORVIS tests [65], which were conducted by the Institut Paul Scherrer (IPS) in Switzerland, have focused on guide thimble and guide thimble passageway behaviour under core melt accident conditions with corium in the lower head. These investigations targeted the corium penetration into the guide thimbles and studied various possibilities of guide thimble passageway failure (see the summary [64]). It has been found that even if corium penetrates quite far into the guide thimbles, the resulting heat flux is usually not sufficient to melt the thimble walls and the RCS pressure and temperature conditions should not cause any plastic instability resulting in their rupture. Tube ejection following the failure of the welds between the lower head and the guide thimble sleeve or the melting of the retaining flange is also unlikely.

It should be noted that two finite-element models for studying guide thimble passageway behaviour have been developed by the Finnish VTT institute as part of the OLHF programme [66]. The results from these models are consistent with the experimental findings.

### 5.1.3.4. Summary and outlook

In order to better appreciate the thermomechanical behaviour of a PWR lower head in the event of a core melt accident and determine the consequences of the potential failure in particular upon the subsequent sequence of events in the accident, the essential parameters consist of the time of failure, the failure mode and the break zone and size.

The numerical models (2D simplified or finite-element models) developed in the context of the RUPTHER, LHF, OLHF and FOREVER programmes have shown their ability to predict the time before lower head failure and the location of the break. The results obtained agree with the experimental data.

Only the 3D finite-element models can be used to provide a more accurate model of the crack and its propagation until a break is created. However, no 3D finite-element model is currently able to correctly assess the size of the break, as this depends on reactor vessel failure mode at high temperature. The failure criterion used in the models must take into account the behavioural variability of the steels used to form the reactor vessels (ductility or hot shortness).

In order to improve the failure criterion and better assess the size of the break in the different cases of core melt accidents, IRSN undertook a collaborative research programme with CEA and INSA Lyon in 2003 on cracking in French reactor vessel steels. Although this programme provided very accurate high-temperature steel cracking kinetic measurements, a crack propagation model is very complex to develop.

The programme has been redirected towards carrying out studies to identify, among the plausible core melt accident scenarios, those for which the propagation of the crack could play an important role in the development of the accident. In the case of accident scenarios with a low pressure in the RCS when the reactor vessel fails (a pressure of less than 20 bar) and with no external cooling of the reactor vessel, these studies show that the reactor vessel failure occurs rather as a result of vessel wall melting. At pressures above 40 bar, on the other hand, cracking of the reactor vessel wall may play an important role in reactor vessel failure. With the aim of completing the results of these studies, other accident scenarios, notably those with external cooling of the reactor vessel, are currently under study.

## *5.1.4. High-pressure core melt*

### 5.1.4.1. Introduction – accident definition and possible consequences

A PWR core melt accident can occur at a high pressure mainly as a result of the following:

- an equipment failure or human error resulting in the RCS valves not being opened;
- a rapid pressure increase in the RCS when it is partly or completely depressurised; such a pressure increase can, for example, be caused when a degraded core is reflooded, due to a very fast interaction between the reflooding water and the core's materials at very high temperatures, or even melt.

These accidents are known as "high-pressure melt" accidents.

At high pressures, the different components of the RCS (hot legs, steam generators, etc.) are simultaneously subjected to the following:

- high temperatures;
- high stresses (mainly due to pressure forces).

The combination of both these factors can result in one of these components failing, i.e. creating a break in it. Such a break is qualified as an "induced break" in the terminology used for PWR core melt accidents.

An induced break can be either of the following:

- a break induced by creep in a hot leg, the steam generator's tubes or even in another RCS component; this mechanical failure occurs under the effect of heating coupled with a high pressure;
- a reactor vessel rupture at high pressures (if no other RCS rupture has occurred previously). In this case, the corium present in the lower head can be ejected into the reactor pit and then into the containment and cause it to heat up directly in a process named "Direct Containment Heating" (DCH), which can result in its failure (see Section 5.2.1).

The creation of an "induced break" reduces the pressure in the RCS, thereby reducing the possibility of DCH occurring. If an induced break in the RCS occurs in the steam generator tubes, however, radioactive substances may be directly discharged into the environment.

It is therefore important to study the behaviour of the RCS in the event of a melt under pressure in order to fully appreciate the associated risks. This chapter solely concerns the induced breaks, as DCH is discussed in Section 5.2.1.

### 5.1.4.2. Physical phenomena

As we have seen in the preceding sections, when a core melt accident occurs in a power reactor, the reduction in the water inventory in the RCS exposes the fuel rods, resulting in the temperature rising and the reactor core's component elements progressively melting. Part of the power released in the core's unflooded zones is then removed from the core through natural convection, i.e. the hot gases (mainly the steam/water vapour that gradually replaces the liquid water as it evaporates) transport a certain quantity of heat from the core into the coldest regions of the RCS. The hot gases are themselves replaced in the core by the cooler gases. As a result, loops form in which the gases flow from the hot areas of the RCS to its cooler ones and the gases that have been cooled in the cooler areas then flow back into the core's hot ones; these are referred to as "convection loops". These movements are "driven" by buoyancy forces according to Archimedes' Principle, i.e. the forces resulting from the difference in density between the hot (and, therefore, lighter) gases and the cold (and, therefore, heavier) gases.

Theoretically, there are two possible modes of gas circulation in the RCS, as shown in Figure 5.8:

- in the first (shown in the left part of Figure 5.8), the gases leaving the core pass through the hot legs of the RCS, the steam generators, the intermediate legs and cold legs before being reinjected into the core's lower part;
- in the second (shown in the right part of Figure 5.8), a water slug remains present in the so-called "intermediate" legs, located, in the case of each RCS loop, between the steam generator outlet and the RCS pump; due to their shape, the intermediate legs (which are called "U" legs) in the loops of the RCS create a siphon (Figure 5.8) in which water can stagnate, forming a slug. The superheated steam leaving the core passes through some of the steam generator tubes (referred to as the "direct tubes") where they cool down, and then return to the reactor vessel through some of the other tubes (referred to as the "indirect tubes") and through the hot leg (which is therefore the seat of a counter-current flow: the hot gases flow from the reactor vessel to the steam generator through the upper part of the hot legs and the cold gases flow from the steam generators to the reactor vessel through the lower part of the same hot legs, as shown in Figure 5.8). This flow pattern has been experimentally demonstrated in a scale mock-up and seems to be the most probable (IRSN's computer models predict high water slug stability; if the slug disappears in a loop, the steam follows the path described in the previous paragraph).

These convective phenomena are not specific to high-pressure melt scenarios, however, a high pressure has the following consequences:

- convective exchanges are much greater at high pressures than at low pressures;
- the pressure present in the RCS generates stresses that are sufficiently great to cause a significant risk of a creep rupture in some of the pipes (the hot leg, SG tube, etc.).

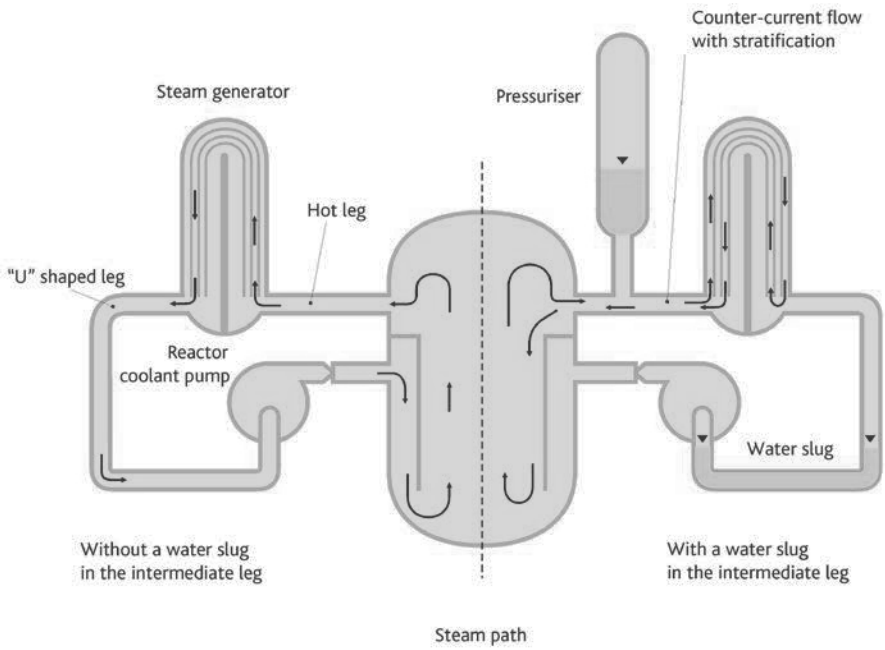



**Figure 5.8.** Modes of steam circulation in the RCS.

In order to determine the location of the break in the RCS, the time sequence of possible RCS failures must be assessed, thereby identifying the earliest. This means that the mechanical and thermal conditions (thermal and mechanical loads) acting on the components of the RCS (SG tubes, RCS piping, etc.) must be known, as well as the behaviour of the corresponding materials at high temperatures.

The mechanical loads are due to the pressure and the thermal expansion of the structures involved (the structures cannot freely expand under the action of heat; they are constrained to do so in a specific way, notably because they are connected to other equipment).

The thermal loads mainly depend on three factors:

- the power released in the core (residual power and power released by the exothermic oxidation reaction of the Zr);
- the transport of heat from the core and into the RCS by means of superheated steam;
- the residual power released by the fission products when they are transported in the RCS (see Section 5.5 for further details of how the fission products are transported in the RCS).

In order to determine the thermal loads, therefore, it is important to be able to model the different convection loops and the release, transport and deposition phenomena of the fission products in the RCS. Other elements must also be modelled in order to assess correctly the thermal loads: whether or not water continues to be injected at the RCS pump seals (a seal failure can result in an RCS rupture), the behaviour of the pressuriser steam bleed SEBIM valves (if a valve jams open, the RCS would be depressurised after a certain number of cycles), and the potential formation of hydrogen "slugs" in the upper part of the SG tubes (hydrogen is mainly produced as a result of oxidation of the zirconium in the cladding by the steam) resulting in the gas flow being blocked.

Studies of high-pressure core melt accidents therefore consist of two parts:

– a thermal-hydraulic part to determine the temperatures (and, also the pressures) in the different parts of the RCS;
– a mechanical part, based on the results of the thermal-hydraulic studies and the properties of the materials involved, to assess when and where the RCS fails.

### 5.1.4.3. Experimental programmes, modelling and computer code

All of the research programmes on core degradation, the release of fission products, corium melt and lower head mechanical strength more or less directly provide data for the high-pressure core meltdown studies. Experimental and modelling programmes specific to this type of situation have also been performed, however.

The first programmes to specifically address high-pressure meltdown were carried out in the United States at the beginning of the 1980s. These notably revealed, in mock-ups, the gas flow patterns. Different existing computer codes were modified in order to model this flow circulation using a simplified geometry; these enabled the thermal loads of the structures to be better assessed. Finite-element mechanical studies were then conducted using these thermal load models. These studies provided more precise models of structural response to the different thermal and mechanical loads. At the beginning of the years 2000, mechanical tests confirmed the validity of this approach and provided data for the modelling of RCS welds.

The improved performance of the computational models can be used to perform Computational Fluid Dynamics (CFD) simulations, which solve 3D fluid mechanics equations, in order to calculate the velocity and temperature fields in the hot legs and the steam generators at a given moment. These methods can partially compensate for the lack of experimental data and help to develop simplified models (for assessing the number of direct tubes and the number of indirect tubes in the steam generators, for example).

#### 5.1.4.3.1. Experimental programmes

**Westinghouse tests:** a test programme conducted by Westinghouse at the beginning of the 1980s, funded by the Electric Power Research Institute (EPRI) in the United States, concerning gas flows and thermal exchanges in the event of a PWR core melt accident. These tests were conducted in a 1/7$^{th}$ scale mock-up reproducing one side of a four-loop Westinghouse PWR (the mock-up reproduced the reactor vessel, two hot legs and two steam generators) and were carried out with sulphur hexafluoride ($SF_6$) in place of superheated steam (this gas behaves like superheated steam under pressure and temperature conditions similar to atmospheric conditions, which greatly simplifies the tests). In particular, these tests revealed circulation flows in the hot legs and SG tubes, the mixing of hot and cooler gases in the SG inlet plenums, and gas stratification in the hot legs. The tests also estimated certain flow data: the mixing ratio in the SG inlet plenums as well as the ratio between the number of "direct" SG tubes (i.e. in which the gases flow from the inlet plenum to the outlet plenum) and the number of "indirect" SG tubes (i.e. in which the gases flow in the opposite direction). These tests are described in several publications, but only partially [68, 69], and were used to qualify computational tools [70, 71]. The ROSA tests, which are mentioned below, aim to provide additional information.

**MECI programme:** conducted by CEA between 2000 and 2004 and financed by IRSN, this programme:

– included a part to determine the mechanical properties of RCS component materials;
– conducted tube burst tests representing the hot legs (half-scale mock-up);
– conducted tube burst tests representing the hot legs (full-scale mock-up).

The material characterisations of the MECI tests added to the existing data on the various grades of steel of the RCS hot legs. They also made it possible to assess uncertainties under creep conditions, determine the properties of the materials used in the burst tests and compare their properties with those available in the literature (including the inventory of RCS material properties compiled by AREVA).

The high-pressure tube burst tests then validated the methods of assessing the failure times for the various structures. They were conducted on the SG tubes and on the tubular test specimens representative of hot leg geometry (half-scale straight tubes) and materials. At constant pressure, the test specimens were subjected to a "temperature ramp" thermal load (heating to provide a constant rate of temperature increase) until they burst.

The tests conducted on the specimens representing the hot legs were mainly intended to determine the behaviour of various grades of materials present in the RCSs. The programme included mock-up tests carried out upon a single material (in other words, entirely consisting of 16MND5 steel, the grade of steel used to construct the French reactor vessels, or 316L steel, the grade of steel used to manufacture the hot leg components) as well as tests conducted upon welded mock-ups representative of the actual welded joints (between the hot legs and the reactor vessel, and between a hot leg's different components, including the joints with the SGs). As a result, two types of welds were studied: "homogeneous" joints (HJ), represented by a welded assembly consisting of two 316L steel tubes (a half-scale study of the links between a hot leg's components), and bimetallic joints (BMJ), represented by the welded joint between two half-mock-ups in 16MND5 and 316L grade steels (to study the links between the hot legs and the reactor vessel).

The test grid is shown in Table 5.1. The first column indicates the component material, the second states the tube's thickness, the third the membrane stress ($\sigma$ in MPa; the stress as defined here is a pressure that "measures" the effect of the forces applied to the structure) and the fourth the temperature heat-up rate (in degrees per second).

In the case of the SG tube tests, two pressure loads were studied: that of a high-pressure secondary coolant system, and that of a depressurised secondary coolant system. These tests were reproduced for various temperature ramp rates, with intact tubes or with tubes containing a notch or recess defect (see Table 5.2). It should be noted, however, that such defects are obtained by machining the parts and so are not completely representative of the defects found in the PWRs.

**Table 5.1.** "Hot leg" mock-up burst test grid.

| **Material** | **Thickness (mm)** | **σ (MPa)** | **ΔT/Δt (°C/s)** |
|---|---|---|---|
| **316L** | 10.5 | 107 | 0.2 |
| **316L** | 10.5 | 107 | 0.05 |
| **316L** | 15 | 75 | 0.2 |
| **316L** | 15 | 75 | 0.05 |
| **16MND5** | 10.5 | 107 | 0.2 |
| **16MND5** | 10.5 | 107 | 0.05 |
| **16MND5** | 15 | 75 | 0.2 |
| **16MND5** | 15 | 75 | 0.05 |
| **BMJ (16MND5L/316L)** | 15 | 75 | 0.2 |
| **BMJ (16MND5L/316L)** | 15 | 75 | 0.2 |
| **BMJ (16MND5L/316L)** | 15 | 75 | 0.05 |
| **BMJ (16MND5L/316L)** | 15 | 75 | 0.05 |
| **HJ (316L/316L)** | 15 | 75 | 0.2 |
| **HJ (316L/316L)** | 15 | 75 | 0.2 |
| **HJ (316L/316L)** | 15 | 75 | 0.05 |
| **HJ (316L/316L)** | 15 | 75 | 0.05 |

**Table 5.2.** SG tube burst test grid.

| **Sample reference** | **Internal pressure (bar)** | | **Rate of temperature increase (°C/s)** | | **Defect geometry** | | |
|---|---|---|---|---|---|---|---|
| | 80 | **150** | **0.05** | **0.1** | **None** | **Notch** | **Recess** |
| 0 | • | | | • | • | | |
| 1 | • | | | • | • | | |
| 2 | | • | | • | • | | |
| 3 | | • | • | | • | | |

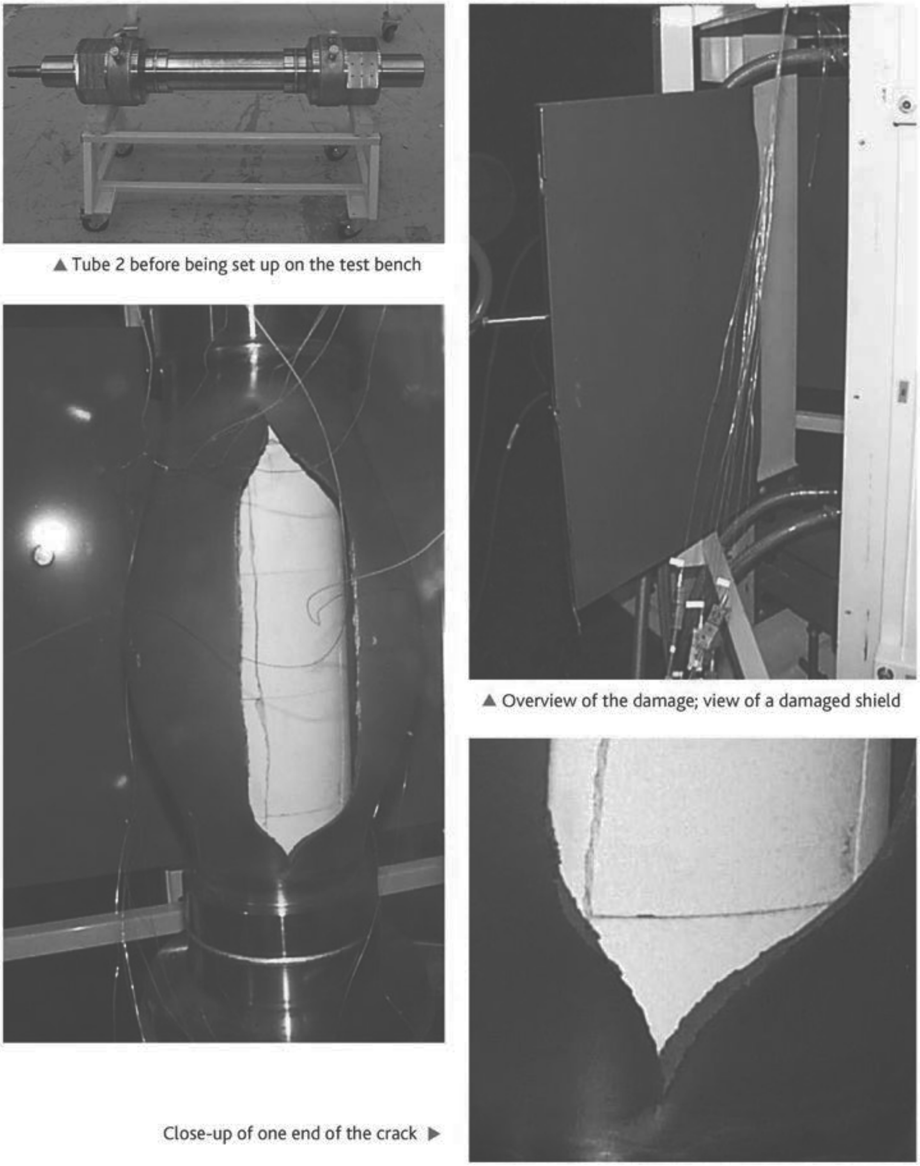
| Sample reference | Internal pressure (bar) | | Rate of temperature increase (°C/s) | | Defect geometry | | |
|---|---|---|---|---|---|---|---|
| | 80 | **150** | **0.05** | **0.1** | **None** | **Notch** | **Recess** |
| 4 | • | | • | | • | | |
| 5 | | • | | • | | • | |
| 6 | | • | | • | | • | |
| 7 | • | | | • | | • | |
| 8 | • | | • | | | • | |
| 9 | | • | • | | | • | |
| 10 | | • | • | | | | • |
| 11 | • | | • | | | | • |
| 12 | • | | • | | | | • |

Figure 5.9 shows the experimental system and the condition of a "hot leg" mock-up after the test.

**ROSA-V programme:** the ROSA experimental programme began in 1970 in Japan and mainly studied the thermal-hydraulic phenomena occurring in PWRs during accident scenarios. The fifth segment of this programme, ROSA-V, was conducted between 2005 and 2009 in the ROSA/LSTF facility of the Japan Atomic Energy Agency (JAEA) and involved many partners including EDF, AREVA, CEA and IRSN in a four-party agreement. The purpose of these tests was to contribute to the development and validation of the thermal-hydraulic models utilised in the computer codes used to compute accident transients that can occur in PWRs by providing "benchmark tests", notably for studying high-pressure core melt accidents. These tests were used to perform comparative exercises between the computer codes used to simulate the thermal hydraulics of the RCS and assess their ability to compute thermal-hydraulics during accident transients. The ROSA/LSTF loop consists of a 1/48$^{th}$ scale mock-up (regarding the volumes; the vertical dimensions are respected) and two loops of a four-loop 1100 MWe PWR. The last tests conducted simulated the natural convection phenomena when superheated steam was present in the loops.

**ARTIST programme**: the ARTIST-1 (*AeRosol Trapping In a Steam-generaTor*) experimental programme, in which IRSN participated, was launched by the Paul Scherrer Institute (PSI, Switzerland) in 2001. It is intended to reproduce the circulation and retention, for the secondary (cold) side of a steam generator, of the fission products (FPs) present in the form of aerosols in the event of a SG tube rupture; its objective is to obtain an experimental database that can be used for safety studies or for the development of models to analyse the retention of FPs, notably in the case of high-pressure melt accidents resulting in an induced break in SG tubes.

The transport and retention of FPs in the RCSs and Secondary Coolant Systems (SCSs) are described in detail in Section 5.5.3.1 of this document.



▲ Tube 2 before being set up on the test bench

▲ Overview of the damage; view of a damaged shield

Close-up of one end of the crack ▶

**Figure 5.9.** Overview of the burst in tube no. 2 in 316L steel with a temperature ramp of 0.05 °C/s and a membrane stress of 107 MPa.

## 5.1.4.3.2. Models

In order to assess the strength of the RCS (resistance to failure), so-called "integral" computer codes are used to simulate a complete accident sequence (these integral computer codes are described in Chapter 8) and, therefore, notably to assess the temperature changes of the different components of the RCS over time (thermal loads).

Given the current possibilities of computation, these complex computer codes generally use highly simplified models of one-dimensional coolant systems to compute the temperature fields. The thermal loads computed with these codes are therefore subject to significant uncertainties.

Complementing the integral computer codes, specialist computer codes are used to perform much more detailed local simulations, notably those of the temperature fields and gas circulation flows, and to assess the thermal loads more precisely. The initial detailed models of convective heat transfers in the RCS in the event of accident transients date back to the 1980s [68]. Subsequent progress in computation has resulted in CFD codes that can be used to conduct thermal-hydraulic studies or to perform finite-element calculations in thermomechanical studies to model convection more precisely without the need for large-scale experimental tests that are difficult to implement.

## ► Modelling of high-pressure core melt in integral computer codes

The integral computer codes presented in Chapter 8 can be used to simulate all of the phenomena that may be involved in a core melt accident, and notably the core degradation and fluid circulation flows in the RCSs and SCSs. They generally use a one-dimensional representation of these systems. To enable the circulation flows and the mixing of hot and cold gases in the SG inlet plenums to be simulated, they are represented in this type of computer code by several volumes, and the gas transfers are performed between the volumes.

The computations performed using the ICARE-CATHARE code can be used to assess the mechanical strength of all of the components of the RCS when a high-pressure core melt accident occurs, depending on the computed thermal loads; some calculations compute the risk of a break occurring in the RCS pump seals. It should be noted, however, that the case of a pressuriser valve jamming in the open position has not been specifically examined (but its modelling would not create any problems). The calculations performed with the code have shown that, if there is a water slug in the intermediate leg of the RCS (see Figure 5.8), it remains in place throughout the period of the accident transient. They have also revealed that the risk of a hydrogen slug forming in the upper part of SG tubes could be avoided.

These computational results must be used with care, however, because of the uncertainties in the thermal load computations. These uncertainties are due to the simplified models used for the coolant systems, on the one hand, and to certain simplified aspects inherent in the ICARE-CATHARE computer code, on the other hand: the code does not model the transport and possible deposition of the FPs released when the core is degraded; furthermore, the core is represented in a highly simplified way, as the code's user must predefine the heat transfers outside the core (two-dimensional [axisymmetrical] modelling of the core can compensate for this simplification, but these models are very costly in terms of computation time).

## ► Modelling of RCS thermal-hydraulics

The approach described in the previous paragraph results in a model of the hot legs or SG inlet plenums consisting of several volume elements (a few dozen at most). By its nature, therefore, it is highly simplified. On the other hand, it can be used to simulate an accident transient lasting several hours.

The CFD approach can be used to model these zones by means of thousands of unit cells and so can numerically simulate the circulation flows of the gases in a RCS loop more realistically than with an "integral" computer code. It requires a long computation time, however, thereby making the calculation of the complete sequence of events in an accident impossible. We must limit ourselves to studying the gas circulation flows at a given moment. The CFD approach has been adopted by the United States Nuclear Regulatory Commission (NRC) in the FLUENT computer code [71] and by IRSN in its CFX and TRIO computer codes (in the latter case, as part of a collaboration with CEA) [72]. The current means of computation restricts the number of unit cells in a model. A steam generator tube bundle (which consists of several thousand tubes) is modelled by a smaller bundle (consisting of approximately ten times fewer tubes), composed of equivalent tubes whose characteristics are determined so that, for example, the total flow cross-section of the tubes of the equivalent bundle is equal to the total flow cross-section of the actual bundle's tubes. The modelling is restricted to the RCS: the exchanges with the SCSs are defined in the form of limit conditions (in other words, only the temperature, which is assumed to be uniform, of the steam in the SCS and a thermal exchange coefficient used to calculate the thermal fluxes between the PCSs and the SCSs are defined). The computations of this type provide a detailed view of the flows in the SG tubes at a given moment and can be used to assess some of their characteristics (mixing ratio in an SG plenum, and the number of "direct" SG tubes and "indirect" SG tubes).

This type of computation provides more precise thermal load results that can then be used to improve the thermal modelling in the integral computer codes as well as to improve the assessment of the mechanical strength of RCS components. Computations performed using the TRIO computer code have, for example, revealed that besides the SG direct and indirect tubes, there were also many tubes with no significant gas circulation flows. They also revealed the possibility of triple stratification occurring in the hot legs, with a "warm" layer between the hot and cold layers.

In addition, they provide 3D profiles of the gas temperatures in the RCS loops. Figure 5.10 shows an example of a thermal profile in a hot loop and a steam generator, calculated using the TRIO-U code. The colours represent the gas temperature ranges (in degrees Kelvin). The superheated steam leaving the reactor vessel passes through the upper part of the hot leg and then cools down when it enters the SG plenum and mixes with the "cooler" steam found there. "Relatively" cold steam flows back towards the reactor vessel through the lower part of the hot leg. A prior computation performed using an integral computer code provides this simulation's "limit conditions" (flow rate and temperature of the hot gases as they enter the hot leg, and SCS temperature).

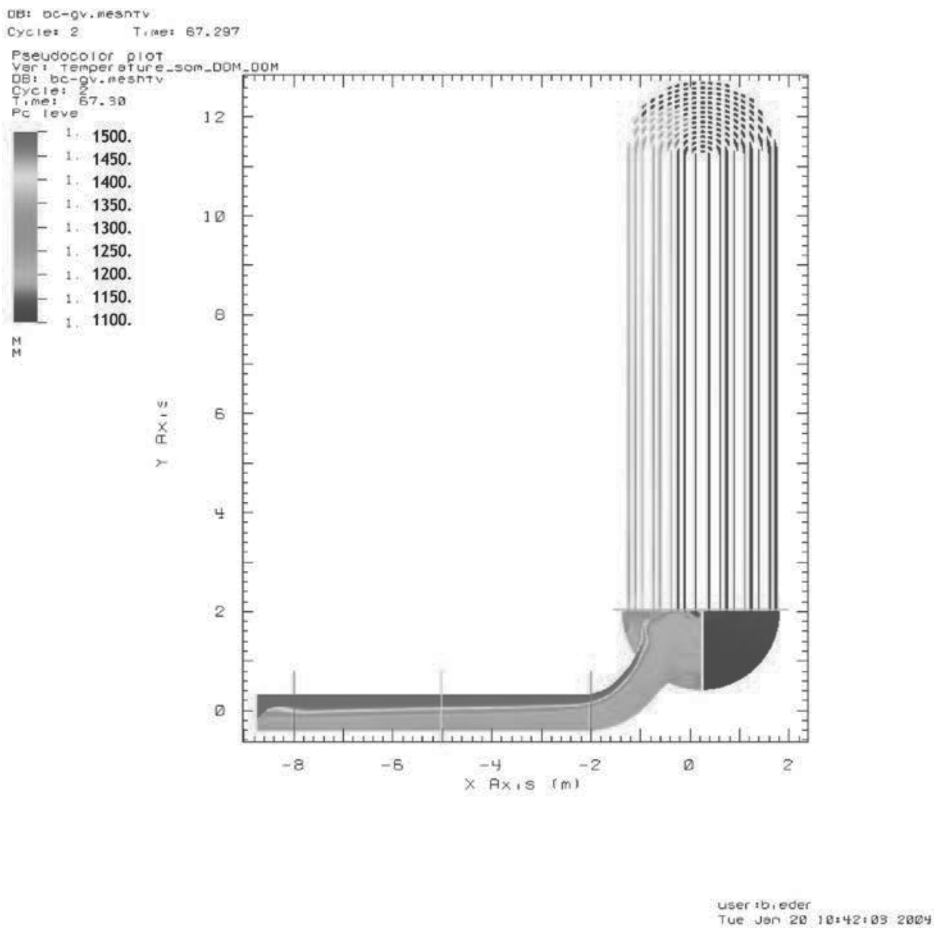



**Figure 5.10.** Example of a thermal field in a hot leg and the associated steam generator, determined using a TRIO-U calculation. The temperature scale is in degrees Kelvin.

The US NRC has conducted similar studies [72]. In particular, it focused on studying the thermal-hydraulic consequences of SG tube leaks existing prior to the accident and showed that such leaks very considerably increased the risk of SG tube failures.

## ► Modelling of RCS component mechanics

In order to simulate the mechanical phenomena, CEA performed finite-element calculations for the assembly comprising a hot leg and the lower head of a steam generator with the CAST3M code at IRSN's request. The mechanical phenomena have been used to study the effects of hot leg expansion upon the mechanical stresses and, consequently, upon the times and places at which it failed. The developed model used takes into account a "realistic" spatial distribution, provided by CFD computations, of the hot and cold layers in a hot leg (in other words, it takes into account the fact that a hot leg is not divided into a cold lower half and a hot upper half and uses the geometrical profile of the separation zone obtained through the CFD computations). To a certain degree, special use of the computational results can be used to take into account the uncertainties regarding material properties as well as those regarding welds.

Figure 5.11 shows a cross-section view of damage to the hot leg at the moment of the failure for a specific thermal load (obtained for a simulation of a total loss of electrical power). One side of the hot leg is welded to the reactor vessel *via* a sleeve (visible at the right side of the figure), and the other side is welded to the lower head of the steam generator *via* an elbow and a conical trunk tube (visible at the left side of the figure). The start of the pressuriser's expansion line that connects the hot leg to the pressuriser can be seen in the figure.

In the computation performed with the CAST3M code, the reactor vessel and the steam generator have been simulated by means of special limit conditions.

The colours represent the level of damage suffered. The damage is a coefficient whose value is between 0 and 1 and is calculated at all points of the unit cell mesh and at every computation step by means of different models specific to the material. A value of 1 represents a failure, whereas a value of 0 represents an intact structure. In this case, the break begins at the beginning of the inner wall of the elbow before the steam generator.

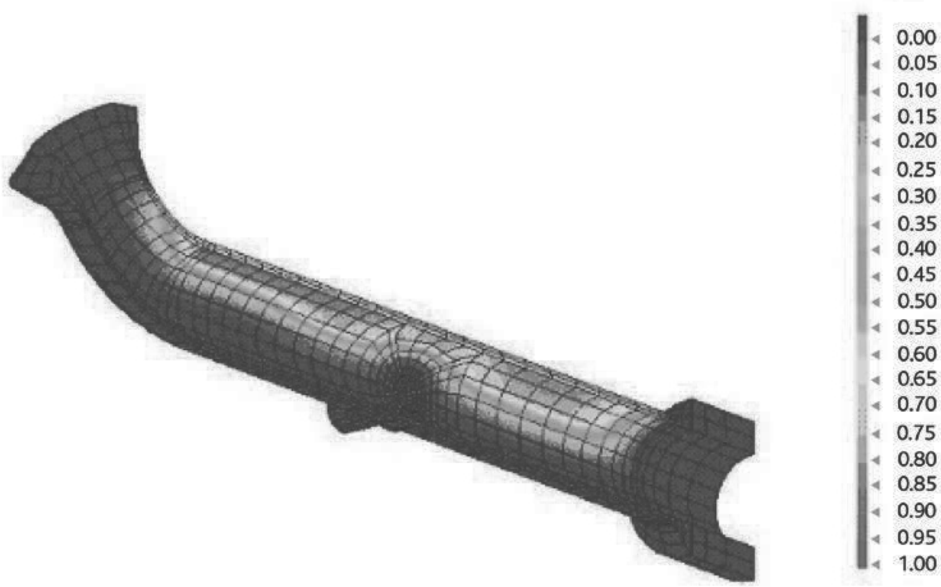



**Figure 5.11.** CAST3M mechanical computation of the strength of the hot leg – damage level at the moment of the "break" (see text for further details).

### 5.1.4.4. Summary and outlook

Research has helped to improve our understanding of high-pressure core melt accidents by increasing our knowledge of the thermal and mechanical loads to which the different components of the RCS are subjected as well as our knowledge of the mechanical behaviour of these components in such situations. Given the complexity of the phenomena involved, notably the RCS gas circulation flows that govern the temperatures of the RCS components, however, it is still difficult to predict with certainty where the first failure in the RCS will occur. The studies performed by IRSN are based on the results of this research; they tend to show that, when a high-pressure core melt accident occurs, the first failure would occur in a SG tube when the SGs are depressurised on the secondary side, or in the hot legs if not.

In the case of the modelling tools, progress could be made in validating the existing tools (notably on the basis of the ROSA test results) or improving the 3D modelling of RCS thermal-hydraulics. This is because only a 3D approach can take into account the complex natural convection phenomena that govern RCS temperature. As things are, it is sufficient to model the mechanics of RCS components, given the uncertainties associated with the thermal load computations.

From the point of view of PWR safety, measures have been taken in France to avoid a high-pressure core melt accident occurring (as in other countries), given the potential consequences of this type of accident, notably in the event of direct containment heating. These provisions include deliberate depressurisation of the RCS if possible before the core melts. This can be achieved by opening the pressuriser steam relief valves. The action of depressurising the RCS is included in the emergency operating procedures and must be performed immediately by the operators as soon as the Severe Accident Operating Guidelines (GIAG) is in use (see Section 4.3.3.4 of the Severe Accident Operating Guidelines).

It should be noted that it has been decided to modify the opening control of the pressuriser steam bleed valves in third ten-yearly outage programme of 900 MWe reactors, in order to make their operation more reliable and thereby make it possible to depressurise the RCS during a core melt accident.

In the case of the EPR, design provisions have been made aiming to “practically eliminate” high-pressure core melt accidents. These are described in Section 4.3.4.2.

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