# Chapter 7
# Lessons Learned from the Three Mile Island and Chernobyl Accidents and from the Phebus FP Research Programme

## *7.1. Lessons learned from the Three Mile Island accident*

### *7.1.1. Introduction*

On 28 March 1979, a core-melt accident occurred on the second unit of the Three Mile Island nuclear power plant (TMI-2, an 800 MWe reactor designed by Babcock and Wilcox) near Harrisburg in Pennsylvania, USA[1] (see Figure 7.1). This accident, which had been considered utterly improbable, caused considerable repercussions around the world. For those in the nuclear industry, it led to a sudden awareness that the risks associated with nuclear power plants needed be reconsidered in depth [1-7].

1. The TMI-2 reactor is similar in its major design principles to the pressurised water reactors (PWRs) operating in France (see Figure 7.1). However, it differs from French PWRs on two main points for facility operation and safety: it only had two core cooling loops, while French PWRs have three or four loops (see Section 2.3.2.2) and the steam generators were countercurrent heat exchangers with straight tubes, whereas French PWRs have U-tubes. In the event of its water supply being interrupted, a straight-tube steam generator dries out in two minutes, whereas a U-tube steam generator takes about 10 minutes to dry out.

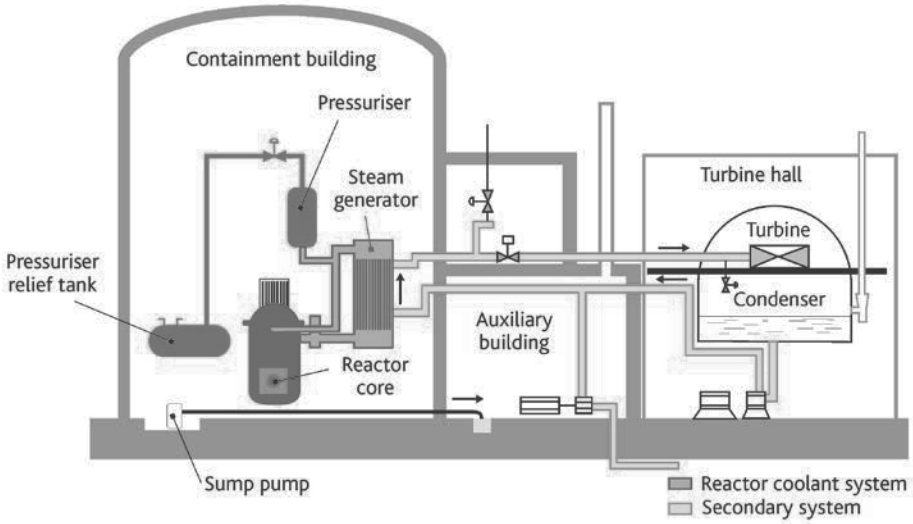



**Figure 7.1.** Diagram of the Three Mile Island power plant

Previously, it had never really been considered that the core of a reactor could melt. The speed with which this melting occurred was also a surprise. After the accident, when observation of the degraded core became possible, it was discovered that nearly half the core had melted and that approximately 20 tonnes of molten materials produced by this degradation had reached the vessel lower head. This raises the question of what would have happened if the operators had not finally closed the breach in the reactor coolant system (RCS) and reflooded the core with water during the accident, or if a larger quantity of molten materials had ultimately accumulated at the bottom of the vessel. The vessel could then have failed, allowing molten materials to flow into the containment building, which would have led to interactions between these materials and the containment concrete, with the eventual risk of a large release of radioactive substances into the environment *via* containment pressurisation or basemat piercing[2].

All these uncertainties and questions raised by the Three Mile Island accident have led to large-scale analysis work in the nuclear industry, a large number of international discussions, and a general review of risks and safety approaches for nuclear reactors [3-7].

A summary of the accident sequence and a physical analysis, in which experts from many countries have participated, is given in the following sections. The analysis is based on interpretation of the data recorded by the power plant's instrumentation during the accident, on knowledge of the final state of core degradation observed after the reactor vessel head was opened (in late 1984, over five years after the accident), on extraction

2. In 2011, the accident on the Fukushima Daiichi nuclear power plant in Japan demonstrated that external hazards (earthquake followed by a tsunami significantly greater than the design-basis tsunami for the facilities) can lead to a core-melt accident. Unfortunately, this accident led to fuel melt and probably to reactor vessel and containment failure on three of the power plant's reactors, along with large releases of radioactive substances into the environment [12].

and examination of the core debris in hot laboratories, and on simulation of aspects of the accident scenario. Collaboration between specialists from various countries has enabled the accident sequence to be largely reconstructed, in particular the thermal-hydraulics in the core and associated systems during the accident, and the stages of core degradation [6, 8 and 9].

The consequences and lessons learned from the accident are presented at the end of Section 7.1. These lessons marked a major turning point in the development of the safety approach for nuclear reactors worldwide.

### *7.1.2. Accident and core-degradation sequence*

The accident's initiating event was a relatively banal operating incident: failure of the main feedwater supply to the steam generators while the reactor was operating at nominal power. This failure was probably caused by an error made during maintenance work on the reactor's auxiliary systems. Due to the low thermal inertia of this steam-generator design, the sudden loss of heat removal *via* the steam generators led to an increase in cold leg temperature and pressure in the reactor coolant system within a few seconds.

As planned for this situation, the RCS relief valve, located at the top of the pressuriser, opened to reduce the pressure in this system by discharging its coolant into the pressuriser relief tank located in the containment building. The incident also very quickly led to reactor trip. Up to this point, control systems functioned as expected.

Two failures then occurred which determined the development of the situation. The first failure was that the pressuriser relief valve did not automatically close when the RCS pressure had fallen sufficiently. Reactor coolant therefore continued to discharge into the relief tank and then into the containment building *via* the tank's overflow once it was full; this corresponds to a loss of coolant accident (LOCA). The second failure was that the emergency feedwater system for the steam generators did not take over from the main feedwater system because its valves, which should have been open, were closed (they had been closed during a regulatory test performed several days previously). The secondary side of the steam generators then dried out in a few minutes, leading to loss of cooling of the RCS by the steam generators.

The first failure had serious consequences because the operators in the control room did not realise that the pressuriser valve had remained open. For over two hours, approximately 60 tonnes per hour of reactor coolant poured into the containment building (for an initial reactor coolant inventory of 200 tonnes). The jammed valve was not quickly diagnosed, as the operators in the control room had no indication of the real position of the valve, only a light that indicated that the close command had been sent. They therefore had no way of knowing if closure had been successfully performed.

The second failure may not have had a major impact on the accident sequence. However, for nearly 25 minutes, operator attention was focussed on re-establishing stable cooling conditions on the secondary side, which probably partly explains why the initial critical phases on the RCS side were not correctly analysed.

The accident sequence can be described in several phases from the initiating event, as suggested in earlier publications [6, 8, and 9].

### ► Phase 1 of the accident: LOCA (estimated duration 100 minutes)

RCS pressure continued to decrease because the pressuriser relief valve remained open. When it reached approximately 110 bar, two minutes after the start of the accident, the high-pressure safety injection system (SIS) started up automatically and cold water was therefore injected into the RCS. Operators then monitored pressure regulation in the RCS by following the pressuriser water-level measurement. In normal operation, with the relief valve tightly closed, the upper part of the pressuriser is filled with a small volume of steam that determines RCS pressure. An instruction tells operators to check that this volume of steam (and therefore the water level in the pressuriser) varies little, which is a sign of RCS pressure stability. However, as the relief valve was open, a two-phase water-steam mixture was escaping from the breach. The apparent water level in the pressuriser, measured using the static pressure difference between the top and bottom of the pressuriser (weight of the water column), therefore seemed to rise quickly, given the large proportion of water in the mixture leaving by the relief valve. Believing the relief valve tightly closed, the operators attributed the rising water level in the pressuriser to the water supplied by the safety injection system and assumed that this rising water level would be accompanied by rising pressure in the RCS. Five minutes after the start of the accident, they took the decision to manually stop the high-pressure safety injection, which would have major consequences. From then on, there was an open breach in the second barrier and emergency cooling was not operating.

From this point, the water that continued to leave *via* the pressuriser relief valve was no longer replaced in the RCS; make-up *via* the chemical and volume control system (CCVS) was not adequate. Approximately 16 minutes after the start of the accident, the volume of reactor coolant lost *via* the breach and the loss of pressure were such that steam started to form in the RCS.

The RCS was now carrying a mixture of steam and water, with an increasing steam fraction over time. Its operation was maintained under these conditions for over an hour, despite a certain number of indications (increase in core neutron flux, reactor coolant pump vibrations, increasing water level in the pressuriser relief tank, and increasing temperature and pressure in the containment building) and alarms that should have alerted the operators to the state of the RCS. Residual heat from the core was being removed both by the steam generators – the operators having succeeded in restoring emergency feedwater supply, which had occupied all their attention – and by the water and steam discharging into the containment building *via* the pressuriser discharge valves (but the operators were unaware of this).

The two reactor coolant pumps were shut down at 73 minutes and 100 minutes from the start of the accident respectively. Given the parameters measured in the containment (in particular temperature and pressure), the operators suspected an RCS leak on the steam generators. They were now counting on core cooling by natural convection.

### ► Phase 2 of the accident: core heating with the RCS and high-pressure SIS shut down (from 100 minutes to 174 minutes)

Shutting down the reactor coolant pumps led to separation of the water and steam phases in the RCS and creation of a volume of steam in the upper part of the reactor vessel. As later estimated, the water level was now around the top of the core.

The reactor was now only cooled by water from the chemical and volume control system. This make-up was not adequate to compensate for the loss of water from the pressuriser relief valve. Loss of water led to a decrease in the water level in the reactor vessel. It was later estimated that the decreasing water level reached the top of the fuel rods in the core 112 minutes after the start of the accident. This point therefore marks the start of uncovery of the fuel rods, which then heated up as they were inadequately cooled.

Between 130 and 140 minutes after the start of the accident, the upper part of the fuel rods was sufficiently hot (a temperature of approximately 800 °C) to cause ballooning and failure of their zirconium alloy cladding, leading to release of gaseous fission products into the containment building *via* the RCS breach (the "high dose rate" alarm in the containment triggered at 134 minutes). At this point, the operators could no longer be in doubt that the situation was serious.

Leakage *via* the pressuriser relief valve was finally diagnosed 142 minutes after the start of the accident. The operators closed an upstream isolation valve, which eliminated the RCS breach and restored the second barrier. However, later assessments show that at this point, half of the height of the fuel rods was uncovered and it was too late to prevent their degradation.

Until 174 minutes from the start of the accident, no means other than the CCVS would be used to cool the core (as the RCS and SIS were shut down). Under the effect of the exothermic oxidation reaction between zirconium alloy and steam (see Sections 4.3.1.2 and 5.1.1.2), the core continued to heat up, which caused increased steam production and increased pressure in the RCS, which was now sealed following closure of the pressuriser relief line. The water level in the core continued to decrease until only one metre of the 3.6 metre height of the fuel rods was covered.

With the gradual drop in water level, heating of the uncovered part of the cladding led firstly to cladding rupture, then to its significant oxidation, and finally to the initial flows of melted metal materials by the formation of eutectic mixtures. Fe-Zr, Ni-Zr, and Ag-Zr eutectics can form at temperatures of several hundred degrees below the melting point of zirconium alloy (see Section 4.3.1.2 and Figure 4.3). The initial liquid formed was most probably the Ni-Zr eutectic between the cladding zirconium alloy and the Inconel of the spacer grids in the central part of the core. Then around 1400 °C, at the same time as exothermic oxidation, melting of the control rod cladding led to a flow of a silver-indium mixture. The stainless steel of the control rod cladding could also have been attacked at around 1300 °C by interaction with the nickel-zirconium eutectic.

When the molten materials reached the water-steam interface, they solidified in contact with water, which led to the formation of a crust around the core axis, called the lower crust in the following text (see Figure 7.2). This crust remained in place until the end of the accident, and samples could be taken and analysed after the reactor vessel head was removed. These analyses showed that it was made up of metal alloys of Zr, Ag, In, Fe and Ni, which coated columns of fuel pellets.

The lower crust's bowl shape can be explained by the changing conditions for core cooling: the central blockage of flow by the forming crust redirected steam towards the perimeter, causing a gradual increase in cooling at the perimeter such that the molten metal alloys resolidified in these areas at levels much higher than the water level in the core.

In the upper parts of the core, continued highly exothermic oxidation of zirconium alloy by the steam led to local attainment of the melting point of zirconium alloy (which ranges from 1800 °C to 1950 °C depending on its oxygen content). Flowing zirconium alloy melted the fuel pellet uranium oxide to form a bimetallic compound (mainly U, Zr, and O). Similarly, zirconium alloy oxidation led to degradation of the fuel rods, leading to solid fuel-pellet fragments falling in the core. This degradation process is assumed to have progressed towards the core perimeter until 174 minutes after the start of the accident.

The lower crust now constituted a crucible that collected metal and oxide compounds. This was probably a mixture of solid debris and molten materials, with a mean temperature of between 2300 °C and 2500 °C at this time.

While the lower crust itself was cooled convectively by steam and also by radiative heat transfer to the water surface, the materials collected in the crucible were poorly cooled and gradually heated up until the centre melted. Figure 7.3 shows a schematic diagram of the state of the core at 174 minutes:

- in the lower part of the core, the fuel rods are intact over a height of approximately one metre;
- a leaktight bowl-shaped crust, made up of resolidified materials, has formed above these intact fuel rods;
- the crucible thus formed contains a mixture of solid debris, along with a pool of molten materials in its central part;
- in the upper part of the core, the fuel rod cladding is highly oxidised, but the majority of fuel rods are still in place.

### ► Phase 3 of the accident: partial reflooding of the core - formation of a debris bed (between 174 and 180 minutes)

At 174 minutes from the start of the accident, the operators restarted the reactor coolant pump on one of the cooling loops to try and restore reactor coolant circulation. This brought 28 $m^3$ of water into the reactor vessel in 6 minutes. This was the largest supply of coolant since the reactor coolant pumps had been stopped 100 minutes after the start of the accident.

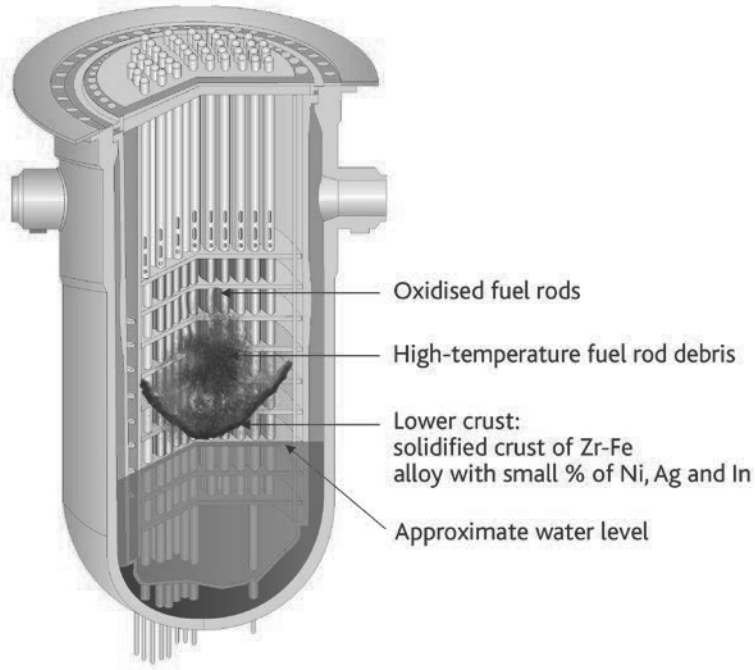



**Figure 7.2.** Assumed state of the core after lower crust formation.

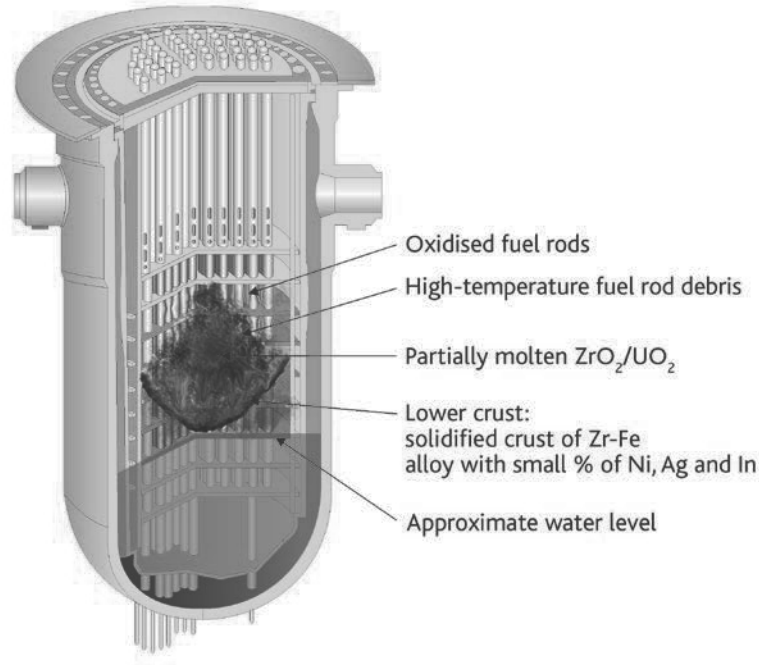



**Figure 7.3.** Assumed state of the core 174 minutes after the start of the accident.

It caused a rapid increase in RCS pressure, associated with the vaporisation of water in contact with the overheated core components, rapid oxidation of the remaining unoxidised zirconium alloy in the upper half of the core, and probably deterioration of heat exchange in the steam generators due to the hydrogen produced by zirconium alloy oxidation.

This water supply to the core probably stopped development of the corium pool above the crust. However, the thermomechanical stresses caused by partial quenching of the damaged oxidised fuel rods in the upper part of the core led to fragmentation of the oxidised cladding and fuel pellets, which then formed a debris bed in the upper part of the core (see Figure 7.4). Later observations and analyses have shown that the bed was made up of several tonnes of compact debris.

Six minutes after restarting, the reactor coolant pump was stopped by the operators because RCS pressure was rising sharply.

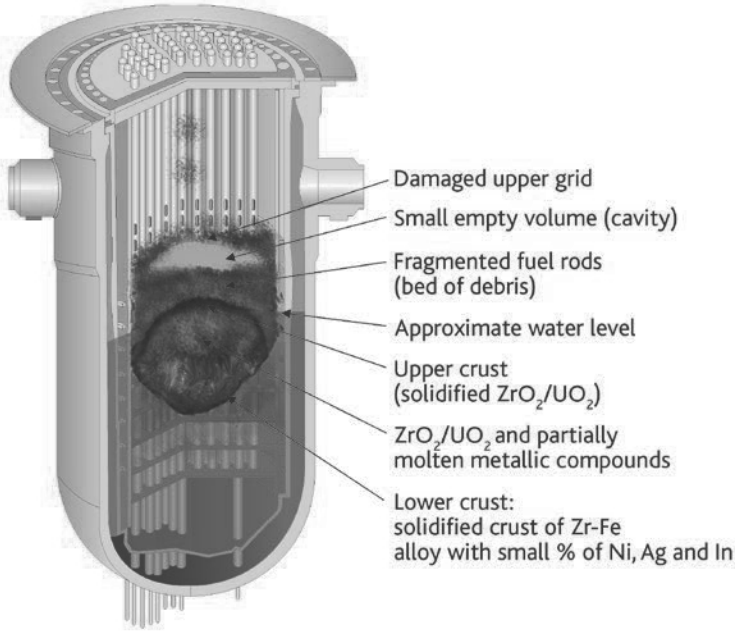



**Figure 7.4.** Assumed state of the core during Phase 3 of the accident.

## ► Phase 4 of the accident: heating of the debris bed – development of the corium pool (between 180 and 200 minutes)

The sharp rise in RCS pressure associated with supply of water to the core led operators to reopen the isolation valve of the pressuriser relief line. This opening caused radioactivity alarms to trigger, including some outside the reactor building. At this point, the first two barriers were "breached" and isolation of the third and final containment barrier (the containment building) had not yet been performed.

In the absence of containment isolation, the triggering of alarms outside the reactor building was due to automatic transfer of contaminated liquid from the containment sumps into storage tanks located in the non-leaktight auxiliary building. These tanks themselves overflowed and contaminated liquid poured into the auxiliary building, leading to releases outside the facility.

An emergency situation was declared 200 minutes after the start of the accident and this led to isolation of the containment building, interrupting transfer of radioactivity to the auxiliary building.

According to the later reconstruction of the accident, the quantity of water in the reactor vessel decreased during Phase 4, due to water boiling off under the effect of the residual heat. At 200 minutes from the start of the accident, the water level in the core was only about 2 m.

Between 180 and 200 minutes, the cooling water flowrate was low. The centre of the debris bed was not cooled due to its mass, its low permeability, the residual heat released inside it and the presence of the lower crust that hindered coolant flow. The debris bed therefore heated up around the corium pool and the pool grew towards the top of the core through gradual melting of debris.

### ► Phase 5 of the accident: total reflooding of the core - continued development of the corium pool (between 200 and 224 minutes)

At 200 minutes after the start of the accident, the operators started up the high-pressure safety injection system and ran it for 17 minutes. Later analysis of the data gathered on RCS temperatures and pressures shows that the reactor vessel was full of water 7 minutes after this system restarted.

It is estimated that during vessel filling, water managed to penetrate the upper debris bed, cool it and rewet it. However, the corium pool continued to heat up. The restarting of the high-pressure safety injection system, between 200 and 217 minutes after the start of the accident, occurred when the corium pool was already too large to be effectively cooled. It is estimated that at 224 minutes, almost all the compacted debris in the crucible formed by the lower crust had melted (see Figure 7.5).

### ► Phase 6 of the accident: movement of core materials towards the reactor vessel's lower plenum (between 224 and 226 minutes)

At 224 minutes from the start of the accident, while the operators were occupied with cooling the core which they did not suspect to be badly damaged, a certain number of measurements suggested that fuel movements were occurring in the core. It was only much later, after examination of the vessel's lower plenum, that the events that occurred at this time could be reconstructed. The crust finally failed on one side and 20 tonnes of molten materials flowed to the bottom of the core, destroying the internal structures located in the core perimeter as they passed (see Figure 7.6).

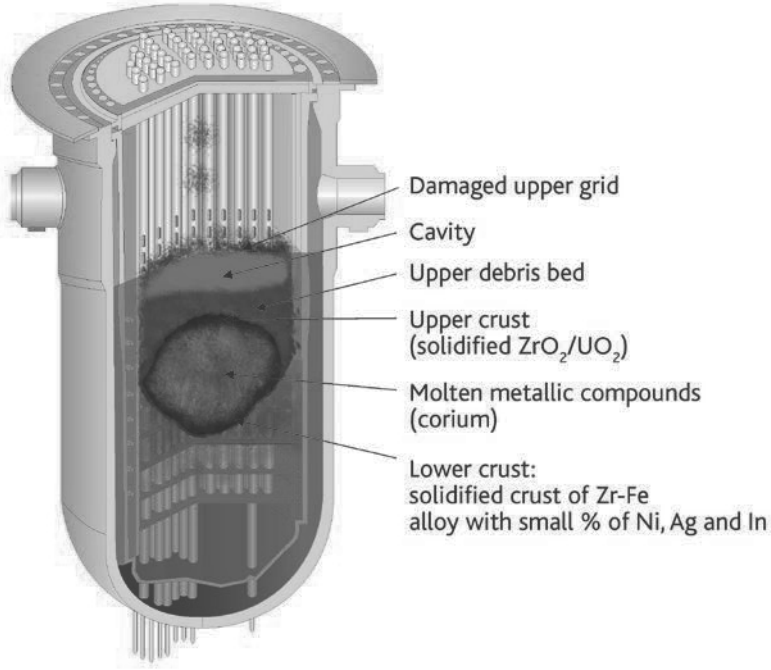



**Figure 7.5.** Assumed state of the core at the end of Phase 5 of the accident.

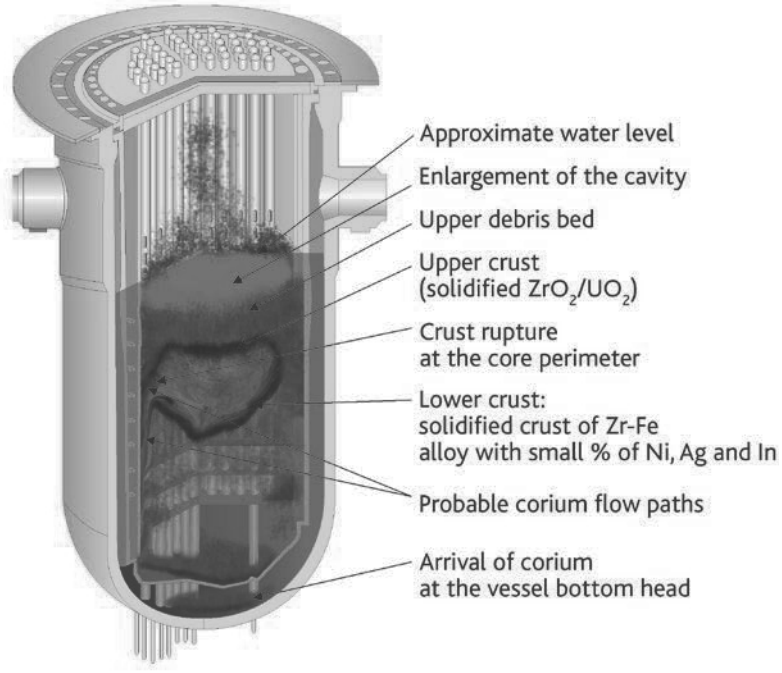



**Figure 7.6.** Assumed state of the core after the flow of molten materials (Phase 6 of the accident).

The mechanism of crust rupture is not known with certainty. Some writers have emphasised compression of the corium pool under the weight of the upper debris bed, while others emphasise the mechanical loads on the crust due to the partial depressurisation of the RCS which occurred between 220 and 224 minutes, after high-pressure safety injection was stopped.

The flow of molten materials to the vessel lower head while the vessel was practically full of water could theoretically have led to a violent vapour explosion (see Section 5.2.3). It is important to note that nothing in the recorded data, or in the state of the core observed after the accident, suggests that such a sudden mechanical phenomenon occurred during the flow of 20 tonnes of molten materials into the lower plenum. One possible explanation is that the relatively long duration of the flow of molten materials (of the order of a minute) did not promote their mixture with the coolant.

The water present in the reactor vessel was finally able to solidify and cool the molten materials after several hours. The vessel lower head did not fail despite the flow of molten materials. One possible explanation, which has already been mentioned in Section 5.1.3.1, is the existence of a gap between the molten corium and the vessel wall, which would have allowed circulation of water or steam and reduced heat exchange between the molten materials and the vessel lower head.

### ► End of the accident: restoration of stable cooling (to 16 hours after the start of the accident)

During Phases 3 and 5 of the accident, operators had attempted to restart RCS cooling. These attempts were hindered by the large quantity of non-condensable hydrogen in the RCS, produced by oxidation of the zirconium alloy cladding and other core materials. Nevertheless, these actions did cool the degraded core as the hydrogen was vented by opening the pressuriser relief line. In doing this, hydrogen and radioactive products entered the containment building.

Hydrogen thus accumulated in the containment building. Hydrogen combustion occurred 9 hours 30 minutes after the start of the accident. It has been shown that, at this time, the molar concentration of hydrogen in the containment was slightly below 8%, along with a small quantity of water vapour (around 3.5%). This combustion led to a pressure peak of 2 bar in the containment (which was designed to withstand 5 bar). The containment suffered no damage, but when entered several months later, fire and pressure damage was observed on some parts of the internal structures.

At 11 hours 8 minutes after the start of the accident, the isolation valve on the pressuriser relief line was definitively closed, bringing to an end transfer of contamination into the containment building.

At 13 hours 23 minutes after the start of the accident, the safety injection system was restarted to fill the RCS.

At 15 hours after the accident, the quantity of water in the RCS was sufficient for reactor coolant pumping to recommence. The reactor coolant pumps were restarted from 15 hours 49 minutes after the accident. Normal, stable cooling was thus obtained approximately 16 hours after the start of the accident. One day after the start of the accident, the reactor coolant pumps were again stopped, as the natural convection flowrate between the reactor vessel and the steam generators had become adequate to remove the residual heat from the core.

The final state of the core in shown in Figure 7.7.

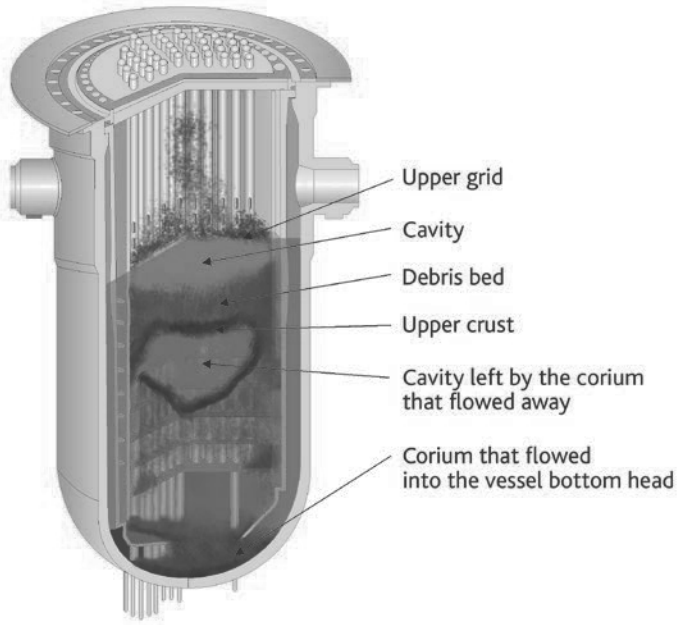



**Figure 7.7.** Final state of the core.

## *7.1.3. Environmental and public health consequences of the accident*

As stated above, there was considerable internal damage to the power plant: nearly half the fuel melted, nearly half the gaseous and volatile fission products (krypton, xenon, iodine and caesium) passed into the reactor coolant, whose radioactivity concentration reached $2.96 \times 10^{16}$ Bq/m$^3$. Over 2000 m$^3$ of this radioactive water poured into the containment *via* the RCS breach.

Despite partial melting of the reactor core and the significant release of radioactive products into the containment building, the immediate radiological consequences on the environment were negligible [1 and 2]. The containment building performed its function. The small releases into the environment that did occur prior to containment isolation

were caused by contaminated liquid from the containment sumps being pumped into a non-leaktight auxiliary building.

In the design of the Three Mile Island facility, start-up of the safety injection system did not automatically provoke isolation of the containment building, i.e. closure of isolation valves on all pipes entering or leaving the reactor building that are not indispensable for reactor core safety. For several hours, the sump pumps therefore transferred water that was increasingly loaded with radioactive products into an auxiliary building. Due to the lack of leak tightness on certain systems, hot contaminated water escaped into this building and evaporated, releasing the iodine and xenon that it contained. These gases and vapours were taken up by the building's general ventilation system and discharged outside *via* iodine filters whose efficiency was found to be inadequate (it was later observed that the filters had not been subject to suitable inspections to check their efficiency). It was only when these transfers of radioactive products triggered alarms that the order to isolate the containment building was given manually.

On the basis of radioactivity measurements performed on the accident site, it has been estimated that only 0.01% of the fission product inventory was transferred from the core to the environment, even though the auxiliary building in question was not designed to constitute a leaktight containment building. With regard to iodine, whose $^{131}$I isotope produces the most significant short-term environmental consequences (see Section 5.5), it as been determined that the total release of this radionuclide into the environment did not exceed $10^{-5}$ % of the inventory present in the reactor core. In the 16 hours following the accident, 37 x $10^{10}$ Bq of $^{131}$I were released, and in the thirty days that followed approximately 259 x $10^{10}$ Bq were released. Releases of other radioactive products were estimated at approximately 18.5 x $10^{9}$ Bq of $^{137}$Cs and 3.7 x $10^{9}$ Bq of $^{90}$Sr.

Many studies have shown that the accident had no health consequences for the general public and had no significant impact on the environment [1 and 2]. However, the Three Mile Island accident traumatised the public. For a whole week, the authorities in charge of public safety were unsure of the degree of severity of the accident and whether partial or total evacuation of nearby inhabitants was required. In particular, they were afraid that the hydrogen bubble formed in the upper part of the reactor vessel could explode (falsely, because this explosion is not possible without oxygen), leading to catastrophic failure of the containment and large releases of radioactive products into the environment.

The contradictory information drip fed by the authorities during the accident did nothing to reassure the public, and over 200,000 people fled the region during the crisis.

### *7.1.4. Lessons learned from the accident with regard to the physics of core melt accidents*

The Three Mile Island accident led to the development of numerous R&D programmes with regard to the physics of core melt accidents, with the aim of better understanding the phenomena which occur during this type of accident (see Chapter 5).

Numerous experimental results have since been obtained internationally with regard to the phenomena associated with a core melt accident on a water-cooled reactor. Knowledge and understanding of the complex phenomena involved in such an accident have greatly increased, and the ability to predict changes in reactor state under severe accident conditions using simulation software (see Chapter 8) has greatly improved.

The multiple analyses and studies of the Three Mile Island accident mean that it is possible to assess the ability of software to simulate core melt accidents, from the initiator event to possible releases outside the containment building. Since the accident, numerous comparisons of simulation software have been performed at the international level to understand the remaining uncertainties in simulating the accident [8-10].

The first two phases of the accident, corresponding to LOCA and the beginning of core heating (see Section 7.1.2), are now correctly modelled by existing software with regard to changes in RCS thermal-hydraulic parameters and the core degradation sequence (hydrogen production, formation of the lower crust, and melting and flow of materials into the crucible formed by the lower crust).

On the other hand, the core reflooding phase (Phase 3) and the later core degradation phases (beyond Phase 3) are not yet correctly modelled. The main weakness of current simulation software concerns reflooding of the degraded core. The phenomena which require more precise modelling to simulate reflooding are the circulation of fluids (water and steam) in the degraded core, heat transfers and zirconium alloy oxidation. These phenomena affect hydrogen production and the flow of molten materials (see Section 5.4.1 for further details). As part of the European Severe Accident Research NETwork (SARNET), IRSN is leading a research programme, PEARL, on the reflooding of a degraded core, with a view to reducing the associated uncertainties by 2015.

## *7.1.5. Lessons learned from the accident for the safety of French nuclear power plants*

### 7.1.5.1. Introduction

The Three Mile Island accident caused considerable shock and many lessons have been learned in the area of nuclear safety, particularly in France.

While core melt accidents on water-cooled reactors had already been the subject of in-depth scientific analysis in the USA from the 1970s (see the WASH 1400 report [11]), it was not until the Three Mile Island accident that the designers and operators of nuclear facilities became aware that core melt accidents were really possible. However, it should be noted that, since publication of the WASH 1400 report, French and other safety bodies have sought to draw practical conclusions from this report in terms of improving the safety of nuclear facilities and drawing up emergency plans for accident scenarios.

While the Three Mile Island accident did not call into question the overall design of nuclear facilities[3], it clearly showed that accidents more severe than those considered up to that point in the design of nuclear facilities (i.e. LOCA resulting from a double-ended guillotine break on the RCS) were possible and that they could result from a series of technical failures and human errors.

The Three Mile Island accident raised a series of questions, such as:

- during an accident, how can inappropriate operator actions, which could aggravate the consequences and lead to core melt, be avoided?
- how best to use the containment building, the final barrier against the dissemination of radioactive substances?
- among real incidents, how can those that could be precursors to core melt accidents be identified, and necessary preventive measures be taken in time?
- how to prepare to confront a core melt accident (a question that applies to both operators of nuclear facilities and public authorities)?

### 7.1.5.2. Analysis of the causes of the accident

The error committed by the operators in picturing the events – they did not understand the origin of the difficulties encountered and persisted in an incorrect picture of the sequence of events – highlights the importance of human factors in the safety of nuclear facilities. The operators did follow the applicable instructions, but on the basis of incorrect or incomplete information:

- with regard to the position of the pressuriser relief valve, the operators saw the indication "valve closed", but this information was incorrect because it was associated with the command to close the valve and not its actual position; this was a crucial aspect of the accident;
- although the operators focussed their attention on the water level in the pressuriser, following the applicable instructions, they had neither the training nor the procedures to deal with a breach located in the upper part of the pressuriser;
- faced with the rapid rise of the pressuriser water-level indicator, and believing the relief valve to be closed, the operators manually stopped the safety injection. The mental picture that the operators had of the situation was false and they lacked direct data on the state of the reactor core.

3. Application of the defence-in-depth concept requires implementation of provisions with regard to a certain number of accidents, which has led to the idea of strong containment in particular. This containment provided significant protection for the general public and staff on the Three Mile Island power plant.

The combination of the following failures and technical inadequacies played a significant role in the accident sequence:

- inadequate indicators in the control room (for example, the position of the pressuriser relief valve, and the water level in the pressuriser relief tank, which would have indicated that the tank was filling up during the accident);
- the lack of prioritisation on alarm signals in the control room; several alarms triggered at the same time, which contributed to disorientating the operators and meant that they could not correctly analyse the situation;
- incorrect positioning of the valves on the emergency feedwater system for the steam generators;
- by design, the start-up of the safety injection system did not automatically lead to isolation of the containment building;
- non-leaktight systems and poor iodine trap efficiency in the auxiliary building, into which contaminated liquids from the containment sumps were pumped.

### 7.1.5.3. Safety lessons

The partial core melt accident on the Three Mile Island reactor confirmed that combinations of failures could lead to a severe accident.

Independently of research performed on core melt accidents, safety reviews have covered three major subjects: human factors in facility operation, experience feedback from operation of nuclear power plants, and management of emergency situations.

#### ► Human factors in facility operation

Prior to the Three Mile Island accident, safety analyses mainly assessed the reliability of safety-related reactor components. The Three Mile Island accident highlighted the fact that people are also an essential aspect of safety, which was already known but little considered.

While operator actions are usually positive from a safety perspective, in certain cases, human actions can contribute to the initiation or development of incidents. It is now important to study operating and working conditions in detail to identify, in particular, the safety problems that could result from organisational difficulties, or from inadequate or unsuitable resources and data.

The explicit acknowledgement of human factors in safety has led to improvements in two technical areas that aim to improve the organisational structure, and specify the allocation of responsibilities and what is expected of each person involved:

- Improving the conditions for operation

  Conditions for operation have been improved by better selection, initial training and ongoing training of operators, and now involve the systematic use of simulators in training. In this regard, the standardisation of the French nuclear fleet

means that simulators are available that are directly representative of the various types of facility. Training covers normal operation, incidents and accidents.

During the accident, the procedures available in the Three Mile Island plant were clearly inadequate. In most countries, particularly in France, procedures and instructions have been reviewed and rewritten. The revision included both the form and content of the documents.

A new approach to accident operation of facilities was thereby implemented (see Section 2.5.2) in order to:

- ensure the "human redundancy" of operators, in particular during an accident, by a safety engineer whose role is to provide independent verification of the relevance of the operating strategy implemented, by monitoring a certain number of safety parameters (using a "safety parameter display console");
- cover the simultaneous occurrence of several apparently independent events as well as possible. In France, an approach has been developed and implemented to give operators the means to bring the facility back into a safe state, independently of the path that led to the situation in question; this is called the "state-oriented approach" (see Section 2.5.2).

In the state-oriented approach, the procedures to be followed are no longer based on the operator's understanding of the sequence of events to which the reactor has been subject (the event-oriented approach) but rather on its actual state at a given moment (characterised by physical data: core sub-criticality, residual heat in the core, water inventory in the RCS, water inventory in the steam generators, leak tightness of the containment building, etc.).

The event-oriented approach is not able to cover all possible combinations of equipment failures and human errors, which may be simultaneous or separated in time. Furthermore, it makes diagnostics difficult in the event that the facility changes in ways that were not foreseen.

In the state-oriented approach, each abnormal state is associated with actions to be performed to bring the facility back to a satisfactory condition. The operating team may perform the corresponding actions without necessarily having understood the sequence of prior events. A key element in the state-oriented procedures has been the addition of a "water level in the reactor vessel" indicator, which means that the operator can know if the core is correctly covered by water (which was not the case during the Three Mile Island accident).

Furthermore, since 1981, beyond design-basis procedures that aim to prevent core melt (H1 to H4 Procedures, described in Section 2.5.2) and ultimate procedures that aim to prevent core melt and mitigate the radiological consequences (U Procedures, described in Section 2.5.2) have been adopted as a principle in France. In the event of core melt, these latter procedures aim to limit releases of radioactive products outside the containment building and therefore into the environment.

All procedures have been tested on a simulator.

– Control room improvements

The observations made following the Three Mile Island accident, with regard to inadequacies in terms of indicators and prioritisation of alarms in the control room, led to modifications in unit control rooms, including those currently in operation. Better presentation of information was sought, by replacing the majority of "command-sent" indicators by valve-position indicators. Certain measurement ranges have been widened. New indicators have been added to supply more complete information on the state of the core (which was lacking during the Three Mile Island accident), such as the indicator of boiling margin (the difference between the actual temperature of the reactor coolant and the boiling point at the RCS pressure) and measurement of the water level in the reactor vessel. In addition, alarms have been prioritised and key data duplicated on a safety parameter display console. This console guides the safety engineer during implementation of accident procedures.

## ► The importance of precursor events

Another important lesson from the Three Mile Island accident concerns taking into account experience feedback from the operation of nuclear power plants.

A precursor incident very similar to the Three Mile Island accident (pressuriser relief valve jammed open) took place in 1977 on an American reactor of the same type (on the Davis–Besse nuclear power plant), but with no damage to the reactor. The operators committed the same analysis error as at Three Mile Island (shutting down the cooling). The lessons from this incident had not been converted into instructions for operators when the Three Mile Island accident occurred. This example illustrates that the systematic study of significant incidents, and any changes to operator procedures and instructions that could be recommended following such studies to prevent such incidents recurring, could in fact prevent more serious accidents.

Since the Three Mile Island accident and the analyses that followed, detection of precursor events that could lead to an accident has become a major concern for operators and nuclear safety bodies. The organisation of operational monitoring and experience feedback has therefore been developed with this new objective.

Following the Three Mile Island accident, a systematic analysis of the possible causes of a breach in the second barrier (the RCS) has been performed, in particular leading to development of more reliable valves for opening and closing the pressuriser relief line.

## ► Managing emergency situations

The Three Mile Island accident demonstrated that the operators, power station managers and authorities responsible for public safety were not sufficiently prepared to manage a core melt accident. Power station managers, and local and federal authorities, did not know how things could develop and whether it was necessary to evacuate. For over a week, the authorities believed in the possibility of a hydrogen explosion which

could damage the reactor vessel and the containment building, causing a large release of radioactive products into the environment. This possibility should have been quickly discounted because the low oxygen concentrations in the core meant that such an explosion was impossible. In uncertainty, people left their homes in a large area around the plant, although the authorities never ordered an evacuation.

From then on, it has been seen as essential that the necessary resources be developed to manage such situations in a less improvised manner in the event that a new situation of this type should occur:

- improved confidence in the behaviour of the containment building, even under conditions very different than those foreseen in design;
- tools available to predict possible changes in the situation, the corresponding releases and their transfer into the environment under accident conditions.

The Three Mile Island accident was partly associated with a false understanding of the situation by operators. It is very difficult for a given team to call into question its initial interpretation of events. It therefore became clear that implementation of crisis teams, distinct from the operating teams and able to take a step back from the situation, could provide alternative insights. Similarly, it was seen necessary to clarify the roles of the various people, and manage the distribution of information in an accident situation. Emergency plans were developed on these bases. The necessity of regular training (crisis drills) was also brought to light.

Since the early 1980s, specific emergency plans have been implemented for nuclear facilities in France. On-site emergency plans have been developed by operators of nuclear facilities with the goal of managing any possible accidents as well as possible, mitigating the consequences, providing help for anyone injured on site and informing the authorities and the press. The authorities have drawn up off-site emergency plans that meet the general objective of public safety in the event of an accident occurring on the facilities.

## *7.1.6. Conclusion*

The Three Mile Island accident has provided many lessons: the importance of defence-in-depth and human factors, along with thorough operating procedures and alarm prioritisation, and the essential role of the containment building, the final barrier between radioactive substances and the environment. All reactors worldwide have benefited from the lessons learned from the Three Mile Island accident. Taking these lessons into account has meant that the calculated probability of core melt on second-generation PWRs has been reduced by a factor of 10. Furthermore, in France, the drawing up of ultimate procedures means that the consequences of such an accident can be mitigated.

For safety experts, the Three Mile Island accident remains a major source of lessons that help understanding of the complex phenomena that occur during a core melt accident. The repercussions of this accident still influence certain research programmes and efforts are continued, in particular to better understand and correctly model the development of such an accident.

Third-generation reactors and the EPR in particular, take lessons from the Three Mile Island accident into account from the beginning. Core melt accidents have therefore been considered in the EPR design. In particular, a core catcher located at the bottom of the containment can collect and cool molten core materials in the event of failure of the vessel lower head (see Section 5.4.3).

## 7.2. Lessons learned from the Chernobyl disaster

### 7.2.1. Introduction

On April 26 1986, seven years after the Three Mile Island accident (see Section 7.1), the reactor exploded on the fourth unit of the Chernobyl nuclear power plant in Ukraine, which was then a Soviet Socialist Republic. This accident is the worst that has ever occurred on a civilian nuclear facility. Prior to the Fukushima accident in 2011, it was the only example of an accident on a power reactor with core destruction and uncontained radioactive releases[4]. This disaster caused considerable consternation worldwide, in particular in what were then Soviet Socialist Republics and in Western Europe.

There were 31 fatalities among the staff present on-site during the reactor explosion and among those who intervened on the site during the days immediately following (notably fire-fighters). Furthermore, several hundred people received high doses of radiation, in particular among the "liquidators[5]". This accident also drew public attention onto the issues of large-scale radioactive contamination. Fallout from the radioactive cloud, which followed highly varied trajectories during the ten days that followed the destruction of the reactor core, affected a large part of what was then the Soviet Union and practically all of Europe.

Since this accident, the authorities in many countries have more actively considered the short-term and long-term management of a post-accident situation. Following the disaster, resources were deployed on an impressive scale in an attempt to mitigate the effects on people and the environment. This involved fire-fighting in a highly radioactive environment, evacuation of a vast number of people, treatment of those who received the highest doses of radiation, protection against the dissemination of radioactivity, decontamination of large areas, monitoring programmes for the food chain, and medical monitoring of the affected population. Greater attention has since been

4. The Fukushima Daiichi accident led to fuel melt and probably to reactor-vessel and containment failure on three of the power plant's reactors, along with large releases of radioactive substances into the environment. In contrast to the reactivity accident at Chernobyl, the cores of the damaged reactors on the Fukushima Daiichi plant did not explode. Initial estimates of the releases performed following this accident show that they were smaller than the releases from the Chernobyl accident (same level for noble gases, approximately 10 times lower for $^{131}$I and three times lower for $^{137}$Cs) [12].
5. "Liquidators" is the name given to the civilian and military workers who were sent onto the accident site in the days that followed the explosion, and until the early 1990s, in particular to construct a rudimentary protective barrier around the damaged reactor, with a view to preventing new releases of radioactive substances (a concrete "sarcophagus" was thereby built around the damaged reactor in the six months following its explosion), and to clear the most contaminated soils over a radius of 30 km around the plant. It is estimated that approximately 600,000 people were involved.

paid to the long-term social and economic disruption caused by an accident affecting a nuclear facility.

Finally, with regard to the important topic of public information and communication, the difficulties encountered during the Chernobyl disaster have led to discussions on the need for better "transparency". The Three Mile Island accident had already highlighted the need for progress in this area.

## *7.2.2. Accident sequence, releases and consequences*

### 7.2.2.1. The RBMK reactor

In 1986, the Chernobyl nuclear power plant had four operating RBMK reactors (in Russian, *Reactor Bolshoy Moshchnosty Kanalny*, which can be translated "high-power pressure-tube reactor"). Two other RBMK reactors were under construction on the site.

The design of an RBMK reactor (see Figure 7.8) is very different from that of the pressurised water reactors (PWR) that EDF operates for power generation in France. The Chernobyl-4 1000 MWe (3600 MWth) RBMK reactor was a thermal neutron reactor, moderated by a graphite stack (whereas water fulfils the role of moderator in a PWR[6]). Each RBMK on the Chernobyl site contained approximately 190 tonnes of uranium oxide (slightly enriched in uranium-235, to around 2%) with zirconium-niobium alloy cladding. The core was cooled by ordinary "boiling" water, circulating from bottom to top in zirconium-niobium pressure tubes (sometimes called "channels"). The neutron moderator was graphite. The water circulating in the pressure tubes was a neutron absorber.

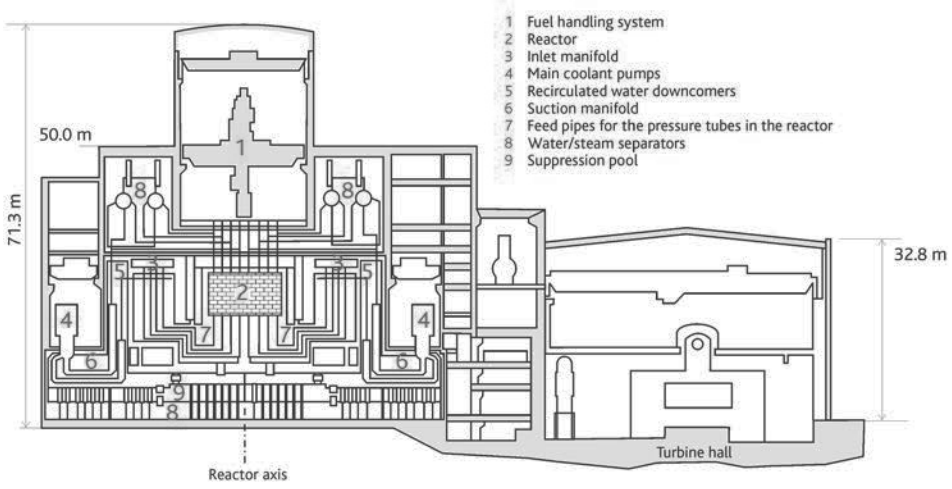



**Figure 7.8.** Schematic diagram of the Chernobyl-4 reactor.

6. The moderator slows down neutrons, so that they produce fission reactions in the fuel more efficiently (see Section 2.1 for further details).

The reactor core was installed in a cylindrical vessel (14 m in diameter, 7 m high). This vessel was filled with a graphite stack, traversed by 1681 pressure tubes containing fuel assemblies and control rods. Cooling water was circulated in these pressure tubes. The water was heated in contact with the fuel and turned into steam, which was sent directly to the turbines *via* two independent cooling loops with no secondary system. The part containing the reactor core was closed off by a 2000-tonne upper concrete slab.

The power level in the core and its distribution were controlled by 211 absorber rod assemblies, which occupied pressure tubes distributed across the whole reactor core just as the fuel assemblies occupied the other pressure tubes. Motorised mechanisms were used to extract or insert these assemblies. The maximum speed of assembly movement was 0.4 m/s, i.e. very slow, taking 18 to 20 seconds for complete insertion (in French PWRs, control rod drop time is around 2 to 3 seconds). These absorber rod assemblies were made of boron carbide with graphite at the bottom, which had the following disadvantage: when the assemblies were in the upper position, their insertion initially replaced water, which is a neutron absorber, with graphite, which is less of an absorber, thereby increasing core reactivity rather than reducing it.

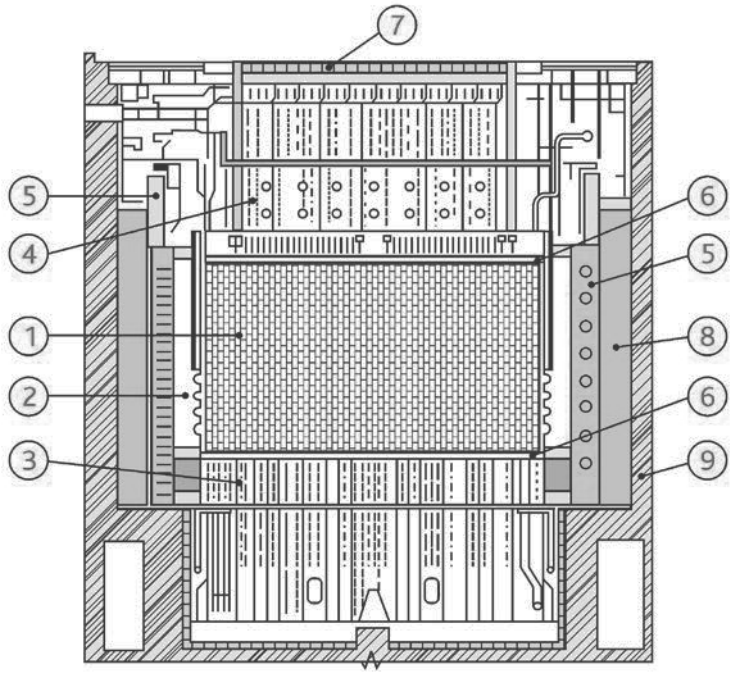
According to the designers, the advantages of the RBMK reactor were the lack of a pressure vessel, as each pressure tube constituted a small reactor coolant system, the lack of steam generators, the ability to continuously renew the fuel (which provided fuel-cycle flexibility), the ability to adjust the flowrate of each channel, and the ability to manage them individually both thermally and with regard to fuel cladding integrity.

The disadvantages of such a reactor were the complexity of the coolant distribution system and, more importantly, the difficultly and complexity of managing power level and distribution, which was the main cause of the accident. RBMK reactors were therefore characterised by:

- possible instability due to radial and azimuthal power fluctuations, more marked for larger cores, due to core poisoning by the xenon ($^{135}Xe$)[7] produced following a power step;
- the possibility of exceeding prompt criticality, leading to runaway reactor power due to a positive void coefficient (see Section 2.1): as water is a neutron absorber, a reduction in water flowrate or increase in steam production in the core leads to a reduction in neutron absorption and an increase in the number of fissions.

During a power increase, the variation in core reactivity is the result of the combination of positive effects (such as the void coefficient) and negative effects (such as

7. In a thermal reactor, one of the products of fission reactions is iodine-135 which decays to xenon-135 in a few hours. This is then transmuted by absorbing neutrons from the fission reactions. Under normal operation, power is relatively stable, and formation and transmutation of xenon-135 are in equilibrium. In the event of a sudden reduction in power, and therefore in the number of fissions, there are insufficient neutrons to transmute the xenon-135 and so it accumulates, as it continues to be produced from the iodine-135 produced by fissions prior to the power reduction. If an attempt is now made to increase reactor power by extracting control rods, the power increase does not occur because xenon-135 is a neutron absorber. It takes around 10 hours to return to a normal neutron balance.




1 Graphite blocks
2 Steel cylinder
3 Lower tube-penetration structure (slab)
4 Upper tube-penetration structure (slab)
5 Annular water tank
6 Steel blocks at the top and bottom of the columns
7 Loading slab
8 Sand
9 Concrete wall

**Figure 7.9.** Schematic diagram of the Chernobyl-4 reactor core

dilation of structures that support the core, and the Doppler Effect[8] associated with the higher fuel temperature). In RBMK reactors, negative effects dominate at nominal power due to the temperatures in the core. On the other hand, at low thermal power (below 700 MWth), the effect of the positive void coefficient dominates, and the power coefficient can become positive (during a voluntary increase in the neutron flux produced, for example, by reducing the penetration of the absorber rods in the core, accentuated by the water vaporisation that accompanies this variation). The increase in reactivity is normally compensated by the control rods. However, if this is not the case, it can lead to exceeding prompt criticality and runaway reactor power (see Section 2.1). Also note

8. The Doppler effect corresponds to a broadening of $^{238}U$ neutron absorption resonances in the neutron energy spectrum. This broadening increases with fuel temperature and leads to a reduction in fission reactions and core reactivity as it heats up (see Section 2.1).

the disadvantage mentioned above, regarding the transitory increase in reactivity when control rods are inserted into the core.

Another problem with the RBMK design was the lack of a large containment building that could withstand significant overpressure around the reactor core. There were just several containment compartments designed to provide confinement of radioactive substances in the event of an accident involving failure of a single pressure tube (the case of an accident involving failure of multiple pressure tubes had not been taken into account in the containment design).

### 7.2.2.2. Accident sequence

The Chernobyl disaster is very well documented; in particular, readers may refer to the reference documents for further details [4, 13-17].

The accident occurred during a test to check the possibility of powering the main reactor coolant pumps from one of the turbo-generators for a few seconds while it was slowing down under its inertia in the event of loss of offsite power, thereby providing additional time for emergency takeover by the diesel generators. This test was performed neither under the planned conditions nor in compliance with reactor operating procedures. In particular, several safety systems were disabled. Furthermore, as described below, although the test was planned at medium power of around 700 MWth, it was performed at lower power and with a delay compared to the planned schedule.

The sequence of events that led to the disaster can be summarised as follows:

- reactor power reduction was started on April 25 at 1:00 am; power was gradually reduced from an initial 3200 MWth to approximately 1600 MWth by around 1:00 pm;
- at the request of the Grid Control Centre in Kiev, the reactor was maintained at half power for around 10 hours to supply the grid. This unplanned period at half power led to reactor poisoning by xenon. The control rods were therefore gradually removed from the core to maintain the power level;
- power reduction was resumed around 11:00 pm;
- at 0:28 am on April 26, the power level was down to 850 MWth; the operators then switched over to the medium-power control system. This switchover, which was poorly controlled, led to an excessive power drop to 30 MWth and further increased core poisoning by xenon. The operators sought to perform the test at all costs and withdrew a large number of control rods from the core;
- the operators started up the two recirculation pumps at 1:03 am and 1:07 am respectively; the increase in fluid flowrate in the core led to a reduction in steam formation and a consequent reduction in reactivity. The operators decided to withdraw more control rods;
- at 1:15 am, the operators disabled the reactor trip signals to that they could perform the test;

- between 1:15 am and 1:22 am, cold water was injected into the core, which further decreased the reactivity. The automatic control rods reached their high position at 1:19 am. The operators then decided to further withdraw manual control rods, dangerously reducing the shutdown margin in the core;
- at 1:22 am, the computer indicated that the shutdown margin was equivalent to only 6 to 8 control rods, whereas the instructions prescribed immediate reactor shutdown if the shutdown margin was less than the equivalent of 15 bars. Despite all this, the operators decided to perform the test;
- four seconds after 1:23 am, the test was started; the slowing of the generator leading to slowing of the reactor coolant pumps, caused a decrease in water flowrate and increased vaporisation in the core, which led to a reactivity insertion and an increase in power, which further accelerated vaporisation. The situation became divergent;
- forty seconds after 1:23 am, the head operator hit manual reactor trip; however, given the design of the control rods (with graphite tips), their entry into the core caused a reactivity insertion, which was probably the final trigger of the reactivity accident;
- forty-four seconds after 1:23 am, the reactivity insertion caused a sudden power surge followed by an explosion. According to some witnesses, a second explosion occurred two seconds later.

An attempt to reconstruct the behaviour of the reactor core during the accident has been made on the basis of later examinations of samples of fuel-bearing materials taken from the damaged facility and the environment. The power excursion probably caused fuel fragmentation, in particular in the lower part of the core. Water in the core would have interacted with very hot dispersed fuel particles, leading to massive vaporisation, an increase in pressure and probably a steam explosion (see Section 5.2.3). This explosion would have led to failure of the pressure tubes, followed by lifting of the reactor's upper slab. The energy released by the explosion has been estimated as equivalent to 30 to 40 tonnes of TNT. The explosion destroyed the reactor building and led to direct releases of radioactive substances into the environment.

Later examinations suggest that:

- high temperatures, of at least 2600 °C, were attained in the reactor core;
- molten core materials flowed towards the lower parts of the reactor and formed several pockets of accumulated "lavas", resulting from interactions between the molten corium and the structural materials (steel, concrete etc.) encountered; this took place over a period of approximately six days following the explosion;
- a large pocket of molten materials, resulting from drainage of local lava pockets, then formed within a crucible-shaped crust located above the concrete structures under the core; this stable, thermally-insulating crust held out for four days following the previous phase;

– this crust gave way about ten days after the explosion, forming three flows. The lava flows then cooled and solidified, resulting in a major reduction in the emission of radioactive substances.

Due to the lack of confinement, a large fraction of the radioactive products contained in the fuel were released into the atmosphere over several days (see Section 7.2.2.3). Large releases were produced over a ten day period from April 26 to May 5, 1986, despite the efforts of the plant operator and the authorities to manage the accident (between April 27 and May 2, approximately 1800 helicopter runs were made to cover the reactor by pouring about 5000 tonnes of materials such as sand, boron, clay, lead and dolomite).

Within the facility, incandescent debris were projected by the explosion and caused various fires, in particular on the roof of the turbine hall. It took fire-fighters about three hours to extinguish them and fire-fighters were exposed to high doses of radiation during their interventions. At about 5 am, a graphite fire was declared. Numerous fire-fighters were exposed to additional radiation attempting to extinguish this fire. The core's enormous mass of graphite burned for ten days after the accident and, after the initial explosion, was probably the main cause of the dispersal of radioactive substances to high altitudes. Radioactive releases continued for about twenty days, but were much smaller after the tenth day when the graphite fire was finally extinguished. A photograph of Unit 4 of the power plant taken just after the accident is given below in Figure 7.10; it shows the immediate damage to the facility.

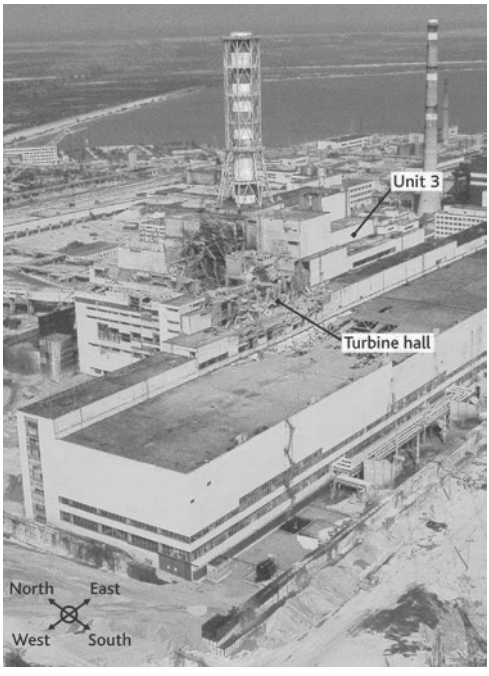



**Figure 7.10.** Photograph of the power plant after the accident (Source: DR).

### 7.2.2.3. Releases and consequences

A third of the releases of radioactive substances occurred at the time of the explosion, which put the reactor core into direct contact with the environment. This large initial release can be largely attributed to mechanical fragmentation of the fuel during the explosion. It mainly contained volatile compounds (noble gases, and iodine and caesium isotopes). The other two-thirds of releases were produced over the following ten days, when molten fuel remained in the reactor and the graphite was on fire. These releases were at their most intense between the seventh and tenth day when core temperatures were at their highest. The sharp reduction in releases observed after ten days may possibly be attributed both to rapid cooling of the fuel, as the molten materials and core debris crossed the lower biological shield and reacted with other colder materials in the reactor, and to extinction of the graphite fire.

Estimates of total releases produced using all available sources [16] (environmental measurements and analyses of what remained in the facility) show that, as percentages of the initial core inventory, 100% of noble gas radioisotopes were released, 50 to 60% of the radioiodine, 20 to 40% of the radioactive caesium, and around 3 to 6% of other radioisotopes (including actinides and fuel fragments that fell near the reactor). This corresponds to release of approximately $2 \times 10^{18}$ Bq of $^{131}I$ (an isotope with a half-life of about 8 days that plays an essential role in short-term effects) and approximately $10^{17}$ Bq of $^{137}Cs$ (an isotope with a half-life of about 30 years that plays an essential role in medium to long-term effects).

Due to the explosion and the intense heat produced by the fuel itself and by combustion of hydrogen (produced by zirconium oxidation) and graphite, the radioactive substances released rose to a relatively high altitude (between 1000 and 1500 m), which contributed to reducing local contamination but led to contamination over long distances [17]. The winds at this altitude carried radioactive substances firstly towards Scandinavia (contamination was first detected in Sweden, prior to any official announcement regarding the accident), then towards Central and Western Europe. Given the duration of the releases and the changes in weather conditions, all of Europe (excluding Spain and Portugal) was ultimately affected by part of the releases. As there were numerous, often very localised, precipitation events during contamination transfers, significant fallout occurred locally, leading to non-uniform contamination of soil and vegetation. Most typical are the "leopard skin" contamination patterns observed in the south of Belarus in particular. Similar phenomena, but at a lower level of contamination, have been observed in Italy, Switzerland and France (in the Vosges, Corsica and the Mercantour Massif). Reference Document [17] gives a detailed description of the dispersion of radioactive pollutants released during the accident at the European and French scales, and estimate mappings of $^{131}I$ and $^{137}Cs$ deposits in France (for 2007).

From a health perspective, 28 people (notably fire-fighters) died quickly due to the very high radiation doses they received (three other people had been killed during the explosion). Many health problems have been observed among the 600,000 "liquidators" (soldiers and civilians who built the rudimentary concrete sarcophagus to contain the damaged reactor and who cleared the most contaminated soils over a radius of 30 km

around the plant). However, the lack of systematic medical monitoring of the liquidators means that the health consequences of the accident on these people cannot be established with certainty. Discussion of these highly complex and controversial aspects is outside the scope of this work.

In the most contaminated areas of Belarus, Ukraine and Russia, an undeniable health effect of the radiation is the very marked increase in thyroid cancer among children under 15 at the time of the accident. The incidence rate is between 10 and 100 times the "natural" rate [17]. Thyroid equivalent doses are mainly due to iodine-131. These doses were received in the three months following the accident. Consumption of contaminated produce and, to a lesser extent, inhalation of radioactive substances during the radioactive releases, were the cause of these thyroid doses.

The inhabitants of the most exposed zone (116,000 people), located in a radius of 30 km around the damaged reactor, were evacuated in the days following the accident. This zone is called the "exclusion zone". Deposits of caesium-137 (the main contributor to long-term doses) generally exceeded 500,000 Bq/$m^2$ in this zone, and reached several million Bq/$m^2$ in the most affected areas. Distribution of the deposits is not uniform around the site; it depends on the various wind directions during the releases.

At greater distances (of the order of several hundred kilometres) caesium-137 deposits are non-continuous; there are patches of contamination wherever the releases encountered precipitation. Three countries were affected over vast expanses by caesium-137 deposits exceeding 37,000 Bq/$m^2$ (the threshold adopted after the accident to define contaminated areas): approximately 41,840 $km^2$ in Ukraine, 46,450 $km^2$ in Belarus and 56,930 $km^2$ in Russia. In these areas, deposits are variable and can reach several hundred thousand Bq/$m^2$, and in places even exceed a million Bq/$m^2$.

In addition to the people evacuated in the days that followed the accident, 250,000 people were evacuated from the most contaminated areas between 1986 and 1995. This large number of people evacuated is explained by the decision of the Soviet authorities to evacuate people from areas where $^{137}Cs$ soil contamination exceeded the threshold of 555 kBq/$m^2$ [18]. Relocation proved to be a deeply traumatic experience for those affected.

The consequences of the accident in France are incomparable with those observed in the most affected countries around the Chernobyl site. However, France was affected by arrivals of contaminated air from the beginning of May *via* the south-east and north-east, which mainly affected the eastern half of the country and had little effect in the western part. Forest soils were the most contaminated, especially in mountain areas, and contamination patches of several thousand Bq/$m^2$ were measured, mainly in Alpes-Maritimes, Corsica and the Vosges. However, recent studies confirm that the doses received by the French public and the associated health risks may be considered as low [17]. In particular, the mean thyroid doses received by children in France are approximately 100 times lower than those received by children in Belarus, where a significant number of thyroid cancers were observed. No epidemiological study has brought to light an excess of thyroid cancers at the dose levels received in France. Nevertheless, the possibility of such an excess cannot be excluded, in particular among children.

### 7.2.3. Lessons learned in France regarding safety

The Chernobyl disaster was very different from the Three Mile Island accident. Chernobyl was a reactivity accident, mainly associated with the specific characteristics of RBMK reactors (in particular, that their operation could lead to a positive void coefficient, which is not the case for PWRs, see Section 2.1). Nevertheless, in terms of safety, the Chernobyl disaster led to the following actions for French nuclear power plants:

- research and emphasis on possibilities for reactivity accidents other than those covered in existing safety reports;
- research and emphasis on physical and organisational possibilities for operation with safety systems disabled;
- enhancement of the safety culture among nuclear facility licensees, in particular with the aim of preventing the disabling of safety systems and deviations from operating technical specifications;
- research and emphasis on situations where there has been non-compliance with the operating technical specifications;
- improvement of systems for measuring radioactive releases into the environment;
- development of dialogue to specify measures that the public authorities could implement to mitigate the consequences of radioactive releases in the medium and long term after the emergency phase;
- movement towards greater information transparency: in particular by the production and use of the International Nuclear and Radiological Event Scale (INES) for accident and incident severity, and by increased involvement of civil society (*via* local information commissions and their national association) in the institutional framework responsible for nuclear safety, which includes public authorities (French Nuclear Safety Authority [ASN]), institutional consultancy (IRSN) and licensees.

In particular, the Chernobyl disaster, a criticality accident not covered by the designers and unknown to plant operators in what was then the Soviet Union, led to a review of the way in which reactivity accidents are considered and covered in France. The possibility of sequences not covered in existing safety reports was explored, bringing to light potentially hazardous scenarios, especially during a reactor outage. These scenarios include accidental injection of "clear" water (i.e. water with a low boron concentration, boron being a neutron absorber) into the core. Additional provisions have since been taken to eliminate, as far as possible, this type of "non-uniform dilution" accident.

Although very different from the Three Mile Island accident, the Chernobyl disaster led to an acceleration of research work regarding reactor core melt accidents, including research and studies on the possibilities of early containment failure. The three phenomena that could lead to sudden loss of containment in the short term – steam explosion, direct containment heating and hydrogen explosion –, have been, and continue to be, the subject of major research programmes (see Section 5.2).

The Chernobyl disaster also raised awareness of the possible long-term effects of a nuclear accident. For the EPR, this led to specification of general safety objectives regarding core melt accidents (see Section 4.3.4.1 for further details).

- core melt accidents that could lead to large early releases must be "practically eliminated"; this includes high-pressure (> 20 bar) core melt accidents;
- low-pressure core melt accidents must be managed such that the associated maximum conceivable releases would only require very limited public protection measures in terms of scope and duration.

### 7.2.4. Lessons learned in France regarding "nuclear crisis" management

The Chernobyl accident made the issue of assessing and managing large-scale contamination by massive ejection of radionuclides from a nuclear facility into an urgent practical question. In France, lessons learned from analysis of the consequences of the accident, and changes to techniques for assessing environmental contamination, have identified areas for improvement in the management of a nuclear crisis. The lessons concern two key questions: "How can the consequences of accidental environmental contamination on people and the environment be assessed with adequate precision and speed?" and "What provisions can best mitigate the impact of such contamination on public health?".

After the Chernobyl disaster, the French authorities decided to enhance their means to assess the radiological consequences on people and the environment by:

- expanding and improving systems for the early detection and characterisation of airborne contamination;
- developing software to model radionuclide transfers into the environment (fallout and contamination of the food chain);
- developing methods to assess and monitor dosimetry consequences for the public (external irradiation from the ambient environment and internal contamination *via* inhalation and ingestion) and implementing protective actions to mitigate such internal exposure.

These developments are described in detail in Reference [17].

Ten years after the accident, in the context of provisions that aim to enhance public protection, the French government decided to distribute stable iodine tablets to people living near a nuclear facility that could release radioactive iodine, so that they would be immediately available to be taken, if needed, on instruction from the prefecture. This would prevent thyroid cancer (taking stable iodine protects the thyroid by saturating the gland – thus preventing it from absorbing radioactive iodine). It was also found necessary to improve the effectiveness of the on-site emergency plans implemented by licensees and the off-site emergency plans implemented by the authorities. More frequent drills, including drills involving the public, have been performed to validate and improve the corresponding provisions.

### 7.2.5. Conclusion

The Chernobyl disaster led the affected countries to review the safety of RBMK reactors, and more generally, the safety of all nuclear power plants in Eastern Europe. International cooperation programmes for technical and financial assistance have been developed to this end. Under international pressure, the Chernobyl power plant was definitively shut down at the end of the year 2000. Other reactors of the same type were also shut down when former Soviet countries joined the EU.

Lessons have also been learned from the Chernobyl disaster for French and other Western European reactors. New research has been launched with regard to reactivity accidents. Measures have been taken to enhance the safety culture among licensees (in particular, with the aim of preventing the disabling of safety systems and deviations from operating technical specifications). However, this disaster has specifically demonstrated that the contamination resulting from a severe accident can be very widespread, cover a whole continent, and have large-scale, long-term socio-economic consequences. In France, lessons learned from the disaster mainly concern the development of resources to manage a nuclear crisis in both the short-term (emergency phase) and the medium to long term (post-accident phase). Environmental monitoring networks (for air, soils and animal and vegetal produce) have also been considerably enhanced. Research has been developed regarding possible measures for mitigating the consequences of a nuclear crisis (such as radiological protection and public health monitoring, radiological monitoring and rehabilitation for affected areas, redeployment of industrial and agricultural activities from those areas, economic support for the affected sectors, and relationships with other countries affected by the accident).

The consequences and health effects of the disaster are still only partially known. In particular, large-scale epidemiological research is being undertaken for the "liquidators" and the populations affected by the evacuations.

## 7.3. The Phebus FP programme

### 7.3.1. Background

Since the Three Mile Island (TMI-2) accident on March 28, 1979, which led to approximately half the reactor core melting, albeit with minimal releases of fission products into the environment, a series of experimental safety-research programmes has been performed by various entities worldwide. Numerous computer models have also been developed to simulate a core melt accident sequence, assess the consequences, and determine the effectiveness of various provisions that could be implemented to mitigate the effects. Launched by IPSN (the forerunner of IRSN) in 1988, the Phebus FP (FP for fission products) programme of experiments was one of the main research programmes focussed on core melt accidents for water-cooled reactors. This programme was launched in partnership with the European Commission and EDF, and performed in close collaboration with CEA, which operates the Phebus reactor. Collaboration quickly expanded to include the USA, Canada, Japan, South Korea and Switzerland. The

collaborative nature of the programme enabled regular international discussions with regard to understanding and interpreting results, and the ability of simulation software to reproduce these results. This was a key factor in the programme's success.

A number of the results obtained under this programme were unexpected. Such results, which are important for safety analyses, regard fuel rod degradation and cladding oxidation, the effect of control rod materials on fuel degradation and fission-product chemistry, and the behaviour of iodine in the reactor coolant system (RCS) and the containment. Analysis of the results as a whole, and their use in studies on the radioactive releases into the environment that could result from a core melt accident, has revealed a certain number of lessons [21, 24, 37 and 58]. Software to simulate the various physical phenomena involved during such an accident has been significantly improved *via* the development of new models. Specific small-scale tests have been performed to understand the unexpected phenomena observed and to validate new models. A list of the main remaining uncertainties was drawn up at the end of the Phebus FP programme. The European EURSAFE Project [50], part of the Fifth Framework Programme for Research and Technological Development (FP5), whose objective was to develop a realistic assessment of possible releases into the environment for better management of the associated risks, has specified research priorities with a view to reducing these uncertainties. Part of this research is the subject of the International Source Term Programme (ISTP) [27, 30], jointly launched by IRSN, CEA and EDF in 2005, which involves a series of analytical tests, in particular regarding iodine chemistry, fuel degradation in the presence of boron carbide (a neutron absorber), oxidation of cladding in air, and the release kinetics of fission products from the fuel. This programme is due to be completed in 2014.

### *7.3.2. Description of the Phebus FP test setup and test matrix*

Five integral in-pile tests were performed during the Phebus FP programme. By providing experimental conditions representative of a PWR under core-melt accident conditions [25 and 57], the setup can be used to study fuel degradation including formation of a molten pool, hydrogen production, the release and transport of fission products in the RCS, aerosol physics, and iodine chemistry in the RCS and containment.

The various physical phenomena studied occur in 1) the reactor core, which is represented by 20 fuel rods, analogous to those in a PWR, along with a 1-metre-long absorber rod, 2) the RCS, whose steam generator is represented by an inverted U-tube, 3) the containment, represented by a 10 $m^3$ vessel with a water-filled part for the sump, a gas-filled part and painted surfaces. These three areas are reproduced at 1:5000 scale compared to a 900 MWe PWR (see Figure 7.11).

The text matrix for the series of tests performed is described in Table 7.1. The experimental conditions were identical for the first two tests (FPT0 and FPT1), except with regard to the fuel: new fuel was used during test FPT0, while fuel with burnup of 23 GWd/tU was used for test FPT1 (burnup equivalent to two fuel cycles in a PWR). The main characteristics of these two tests were a relatively high steam flowrate at the test device inlet to maintain a fairly oxidising atmosphere during the entire phase of

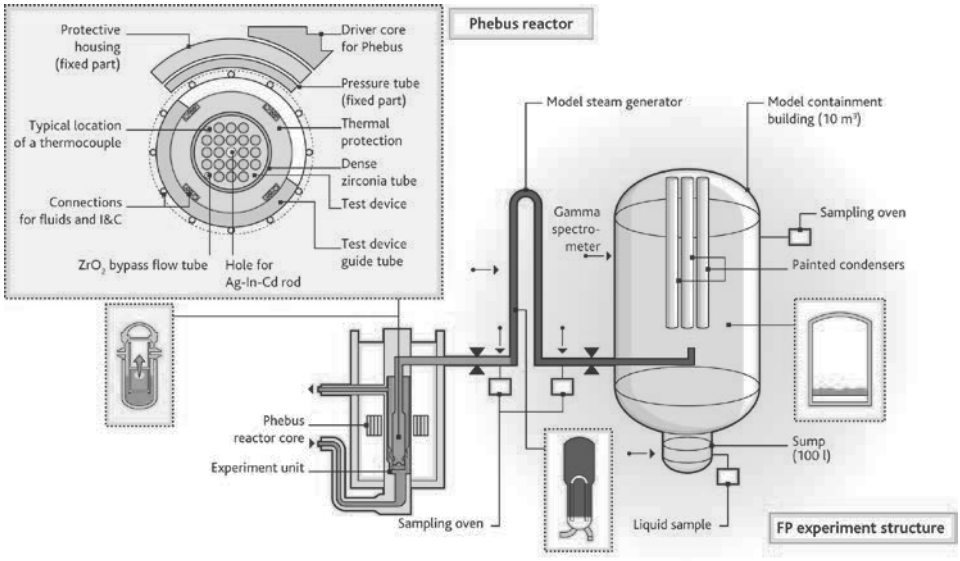



**Figure 7.11.** Schematic diagram of the Phebus FP test setup. Top-left box: test device (assembly of 20 fuel rods).

zirconium alloy cladding oxidation by steam, the use of a silver-indium-cadmium control rod (neutron absorber), an RCS adjusted to 700 °C on the hot leg and 150 °C on the cold leg, painted surfaces in the containment building to study iodine interactions with paints (painted condensers), and an acidic aqueous solution in the containment sump. During the FPT2 test, the injected steam flowrate was much lower, leading to a less oxidising (hydrogen-rich) atmosphere during fuel rod cladding oxidation, and with the aqueous solution in the containment maintained alkaline at a temperature above that of the gas phase to cause evaporation from the sump, unlike in the previous tests. The experimental conditions for the FPT3 test were identical to those of the FPT2 test, except with regard to the type of control rod, where a boron carbide rod was used, and the pH of the sump (which was acidic rather than alkaline). Finally, the main objective of the FPT4 test was to study the release kinetics of low-volatile fission products and actinides from a debris bed made up of fragments of fuel pellets and oxidised cladding, typical of what was found after the TMI-2 accident.

Each test was performed in two phases. During the first phase, which lasted several hours, the fuel temperature was gradually increased up to its melting point and fuel assembly failure. This led to the release of fission products and structural materials into the containment vessel *via* the RCS. Then, over a four-day period, the behaviour of these fission products and structural materials in the containment was studied, whether they were in the form of aerosol particles (agglomeration, transport and deposition phenomena, etc.) or gases, in particular with regard to iodine. The radioactivity of the released fission products produced a significant dose rate in the containment and the effect of radiation on fission-product chemistry was also studied.

**Table 7.1.** Objectives and experimental conditions for the Phebus FP tests

| Test | Main objective | Experimental conditions | | | |
|---|---|---|---|---|---|
| | | Fuel assembly(1) | Reactor coolant system | Containment vessel | Date |
| FPT0 | Formation and development of a corium pool and release of fission products into a steam-rich mixture | New fuel<br>1 Ag-In-Cd rod<br>"re-irradiation" for 9 days | No steam condensation in the steam generator | Painted surfaces<br>Sump at pH 5 | December 2, 1993 |
| FPT1 | As for FPT0 but with irradiated fuel | Fuel with burnup of 23 GWd/tU<br>1 Ag-In-Cd rod<br>"re-irradiation" for 9 days | As for FPT0 | As for FPT0 | July 26, 1996 |
| FPT2 | As for FPT1 but with release of fission products into a hydrogen-rich mixture | Fuel with burnup of 32 GWd/tU<br>"re-irradiation" for 9 days | As for FPT1 but with boric acid injection | As for FPT1 but with sump at pH 9 with evaporation | October 12, 2000 |
| FPT3 | As for FPT2 | As for FPT1 but with $B_4C$ rod<br>Fuel with burnup of 24 GWd/tU<br>"re-irradiation" for 9 days | As for FPT0 | As for FPT2 but with sump at pH 5 with evaporation, hydrogen recombiner coupons | November 18, 2004 |
| FPT4 | Release of low-volatile fission products and actinides from a bed of $UO_2$ + $ZrO_2$ debris | Fuel with burnup of 38 GWd/tU<br>no "re-irradiation" | Addition of integral filters upstream of the test device<br>Chemical analyses of the samples taken | | July 22, 1999 |

(1) The fuel was "re-irradiated" in the Phebus reactor in order to reproduce the inventory of short half-life fission products, in particular $^{131}I$, which had decayed from the fuel since its unloading from the reactor where it had been irradiated.

## *7.3.3. Main lessons regarding fuel rod degradation*

### 7.3.3.1. Oxidation of fuel rod cladding

In the event of a core melt accident on a water-cooled reactor, fuel degradation begins by oxidation of the cladding with rapid temperature rise (see Section 5.1.1.2). During the first Phebus FP test, FPT0, the observed temperature rise was faster than predicted by prior simulations, with significant hydrogen production, beyond the design of the measurement devices (see Figure 7.12), and with fuel-rod temperatures rising to very high levels, above 2400 °C. The initial simulations performed after the

test, using the same models as for the pre-test simulations but with the real boundary conditions from the tests, did not reproduce the observed data. Due to the sudden increase in cladding temperature, there was competition between its oxidation, which progressed from the external surface forming an oxide layer, and the dissolution of $UO_2$ fuel in contact with the remaining molten metal cladding. These phenomena were interrupted when the outer zirconium oxide layer ($ZrO_2$) became too weak to retain the liquid metal within (this is called "cladding dislocation"). The computer models used to simulate these phenomena include correlations that specify a cladding dislocation criterion based on its temperature and the thickness of the $ZrO_2$ layer. These correlations have been modified to provide correct simulation of cladding oxidation and the associated hydrogen production (see Figure 7.12) for the FPT0, FPT1 and FPT2 tests [54], and the FPT3 test [56].

Furthermore, hydrogen production kinetics were strongly influenced by the steam flowrate injected into the assembly. At low flowrates, the steam was almost entirely consumed by cladding oxidation in the lower part of the fuel rods, and the fluid was steam depleted in the upper part, leading to a significant period of "steam starvation" (i.e. the phase in which steam has been almost entirely consumed by oxidation) during the main cladding oxidation phases for the FPT2 and FPT3 tests. Hydrogen production calculated using these correlations is seen to be a slight overestimate [56], which is probably associated with the coupling between oxidation phenomena, fuel dissolution and cladding failure.

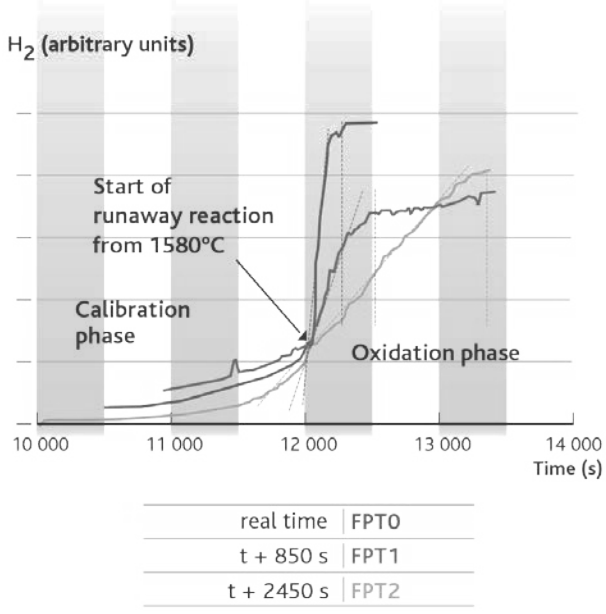



**Figure 7.12.** Hydrogen production kinetics measured during the FPT0, FPT1 and FPT2 tests (solid lines) and calculated using correlations revised following the Phebus FP tests (dotted lines). In order to aid comparison, the curves for the FPT1 and FPT2 tests have been drawn with a time shift compared to the real start time of the test.

### 7.3.3.2. Fuel degradation

The Phebus FP experiments produced greater fuel degradation than had been previously obtained in integral experiments of the same kind. In particular, it was observed that fuel melt, and structural failure of the rod assembly into a pool of molten materials, could occur at a temperature of 2350 °C (± 200 °C), which is much lower than the melting point of pure $UO_2$ (2830 °C) [54]. The very severe degradation observed during the Phebus FP tests seems to be associated with significant interactions between the fuel and the structural materials (mainly steel structures and zirconium alloy cladding), probably intensified by fuel swelling, due to the presence of large quantities of gaseous and volatile fission products. Fuel oxidation by steam leads to a change in its stoichiometry (an increase in the quantity of oxygen with respect to uranium during the tests), reducing its melting point, which seems to have played an especially significant role [20]. Although detailed modelling of these phenomena still needs to be improved and developed, current simulation software correctly reproduces the final state of assembly degradation (see Figure 7.13) as long as a suitable reduction is applied to the fuel rod relocation temperature[9] [55].

The sensitivity of degradation models to significant parameters (cladding dislocation criteria and fuel relocation temperature) has been assessed for simulations of the TMI-2

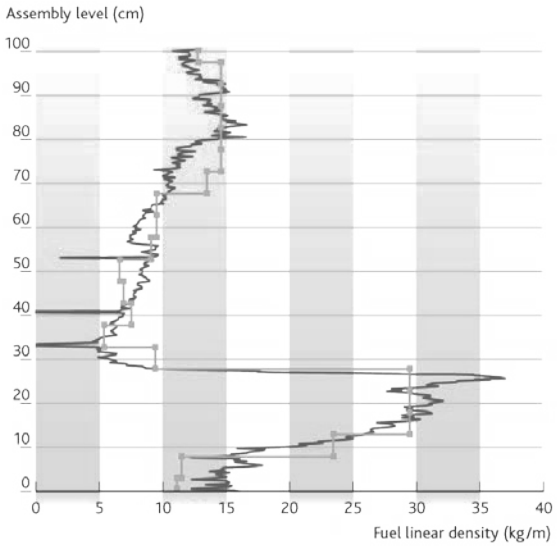



**Figure 7.13.** Examples of axial fuel distributions as measured (in purple) and as calculated using the ICARE2 code (in orange) [55] at the end of the FPT2 test, which highlights an accumulation of materials on the lower part of the degraded assembly (area where a corium pool formed) and a lack of materials above (area where a cavity formed).

9. "Fuel relocation temperature" is the name given to the temperature at which the simulation software assumes fuel assembly failure and large-scale downward movement of the fuel, whether due to melt phenomena or solid-phase flows.

accident using the American MELCOR code [43]. During this study, it was shown that the calculated behaviour of the core is strongly dependent on the degradation models used and the values of their parameters. A fuel relocation temperature of 2230 °C for structural failure, deduced from the results of the Phebus FP tests, has now been adopted for the MELCOR code and gives quite good results. The same conclusions and experience feedback have been drawn from studies performed using the ICARE/CATHARE and ASTEC code developed by IRSN (described in Sections 5.1.1.3.2 and 8.3).

Degradation of the debris bed in the FPT4 device, including transition from a debris bed to a pool of molten materials, has been reproduced (see Figure 7.14), assuming fuel swelling [31] that causes a reduction in porosity inside the debris bed and the consequent redirection of steam towards the bed perimeter (and a reduction in convective heat exchange).

Following the FPT4 test, analysis of the results of destructive tests on the fuel show that it appears to have been oxidised during the transient, solid-state interactions led to partial early melting (at 2530 °C, which is low in the absence of unoxidised metals), and part of the corium was formed by separate melting of the two components ($UO_2$ and $ZrO_2$). While no direct measurement of post-test fuel porosity is available, these observations are consistent with the assumption of fuel swelling. Finally, analysis of the composition of the molten phases shows that temperatures exceeding 2700 °C were attained.

The FPT3 test [53] was performed under conditions very similar to those of the FPT2 test, except for the presence of a boron carbide ($B_4C$) control rod instead of a silver-indium-cadmium alloy (Ag-In-Cd) absorber. Although the power transient was stopped at a lower level than in the FPT2 test, with less extensive assembly degradation, earlier degradation

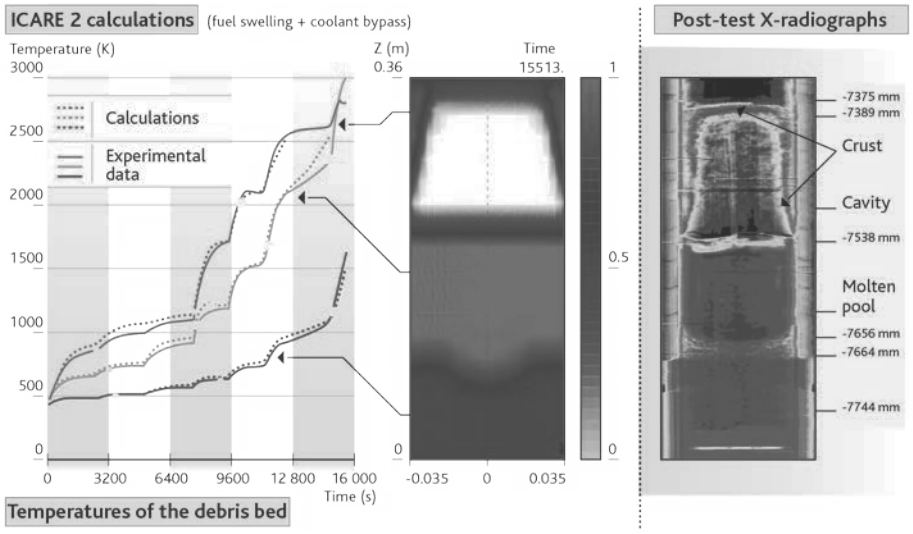



**Figure 7.14.** Changes in (calculated and measured) temperatures in the FPT4-test debris bed and the final state of fuel degradation indicated by the ICARE2 code (calculated specific gravity) and by post-test X-radiograph (measured specific gravity).

occurred, in particular during the cladding oxidation phase. Furthermore, simulation software is currently unable to correctly reproduce all the results of the FPT3 test, in particular the duration of the hydrogen-rich (steam-starved) phase (see Figure 7.15). Some hypotheses have been formulated regarding the possible role of $B_4C$ to explain early cladding degradation [28 and 56] and have been explored using separate-effects experiments (the BECARRE programme, completed in 2010 as part of the ISTP [27]). These experiments demonstrated that a molten $B_4C$-steel mixture could be projected onto the fuel rods next to the boron carbide control rod and accelerate cladding degradation.

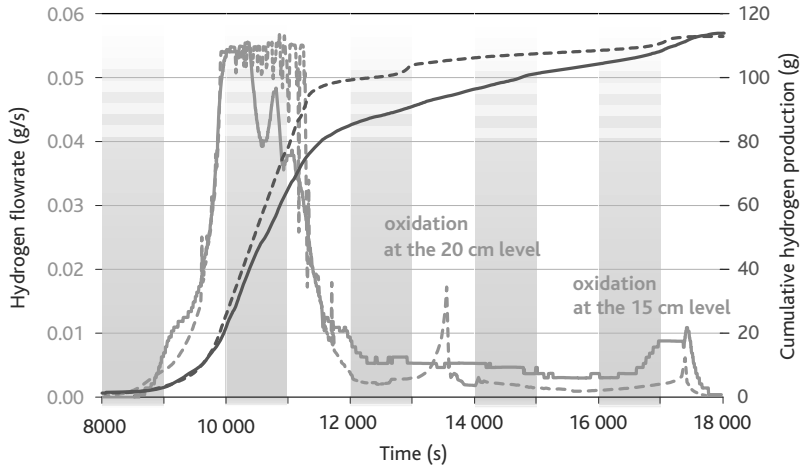



**Figure 7.15.** Hydrogen release flowrate and cumulative hydrogen production during the FPT3 test. Comparison between (ICARE2) calculation (- - -) and experiment (——).

## *7.3.4. Releases from the core*

### 7.3.4.1. Fission product releases

Generally, very similar percentages of fission product inventories were released from the fuel during the three tests FPT0, FPT1 and FPT2. However, due to the lower flowrate of steam injected into the test device during test FPT2, a non-negligible fraction of the volatile fission products (Mo, Cs, I and Te) released was deposited in the upper part of the fuel assembly and in the non-thermally-regulated area located just above it. This was also observed during test FPT3.

Fission product releases depend on the characteristics of the $UO_2$ fuel (temperatures, degree of oxidation, and burnup) and its interactions with the other components that make up the core as degradation of the materials progresses. These dependencies were clearly revealed during the Phebus FP tests. Volatile fission product releases are generally well predicted by all simulation software (see the results for caesium in Figure 7.16), although some software, in particular codes based on a CORSOR-type approach[10], overestimates

10. CORSOR-type models use "releases as a function of temperature" correlations derived from small-scale experiments.

fission product release kinetics at the beginning of the power transient, specifically during the runaway cladding oxidation reaction [26 and 29]. Semi-empirical software adequately simulates volatile fission product releases both for analytical tests and for Phebus-type integral tests despite the fact that it does not simulate all phenomena in detail, but rather uses simplified models to take into account the most influential phenomena, such as the significant increase in diffusion inside the uranium oxide matrix during its oxidation. This is true of the ASTEC code for example.

It should be noted that in the specific case of the FPT0 test, characterised by fuel with very low burnup, the early release of volatile fission products can only be explained by partial dissolution of the fuel from the cladding oxidation phase [35].

Semi-empirical simulation software has been less successful in predicting the release kinetics of semi- and low-volatile fission products, which are strongly dependent on chemical interactions (see the results for barium in Figure 7.16). However, better understanding of the phenomena that govern these releases has been obtained *via* use of mechanistic models [35], such as those included in the Modelling Fission Product Release (MFPR) code (see Section 5.5.2.3 for more details). This knowledge has been gradually integrated into the software by using simplified models. Barium releases during the Phebus FP tests performed using a fuel-rod assembly are much lower than those measured during analytical tests. The difference has been attributed to interactions between the fuel and the cladding materials (such as the zirconium and iron in the control rod cladding), leading to a marked reduction in barium volatility [34]. This hypothesis is confirmed by the results of the FPT4 test (using a debris bed made up of fuel and cladding fragments with no control rod material), during which barium releases were much higher during the initial test phases, while the materials were still solid, than during the higher-temperature phases when melting started along with interactions between the fuel debris and fragments of oxidised cladding [22].

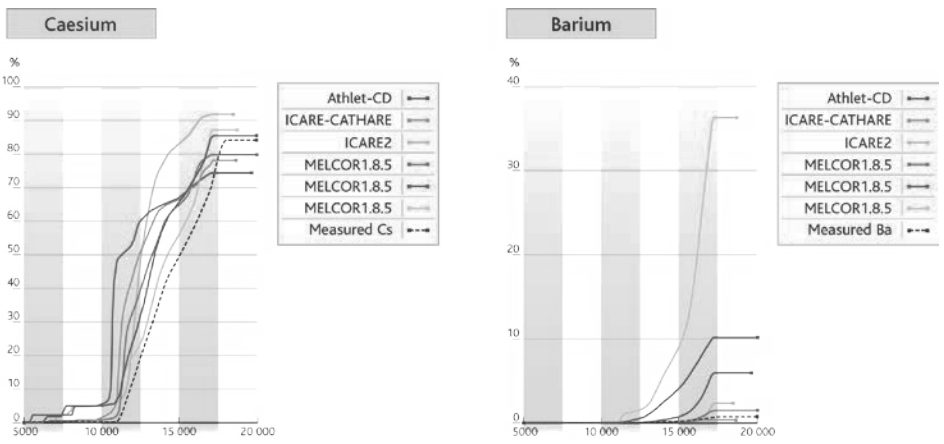



**Figure 7.16.** Caesium (left) and barium (right) release kinetics during the FPT1 test, as simulated by various computer models (results from International Standard Problem 46 (ISP-46) [26]). Comparison with experimental measurements (black dotted line).

The results of the Phebus FP tests therefore clearly demonstrate a link between fission product release kinetics and fuel degradation phenomena.

### 7.3.4.2. Aerosol emissions from structural materials

Aerosol emissions from structural materials and control rods are significant for two reasons:

- the mass of structural materials in the core is much greater than the mass of fission products. These materials make up a large share of the mass of aerosols circulating in the RCS and released into the containment. This share represented the majority during the first two tests (over 50% of the total mass of aerosols), but was lower during the FPT2 and FPT3 tests (approximately 35% of the total mass of aerosols in the containment vessel). The total mass of aerosols present in the containment over time and the deposition kinetics of these aerosols therefore depend on the emission of structural materials;
- certain elements involved interact with the fission products (e.g., silver reacts with iodine), which can modify the behaviour and volatility of the fission products.

Emission of materials from silver-indium-cadmium control rods was poorly simulated by most of the software [26], in particular with regard to silver (see Figure 7.17). The major phenomena governing the emission of these elements are now relatively well understood and models have been developed and integrated into some simulation software, such as ICARE, ASTEC and MELCOR. However, the impact of control rod degradation on the emission of some of its constituent elements still needs to be better taken into account. Emission of the tin originally contained in the zirconium alloy cladding (see Figure 7.17) has also been reconsidered. On the basis of experimental results from the Phebus FP programme, a new model was produced that predicts gradual release of this element as the cladding oxidises during a core melt accident on a water-cooled reactor.

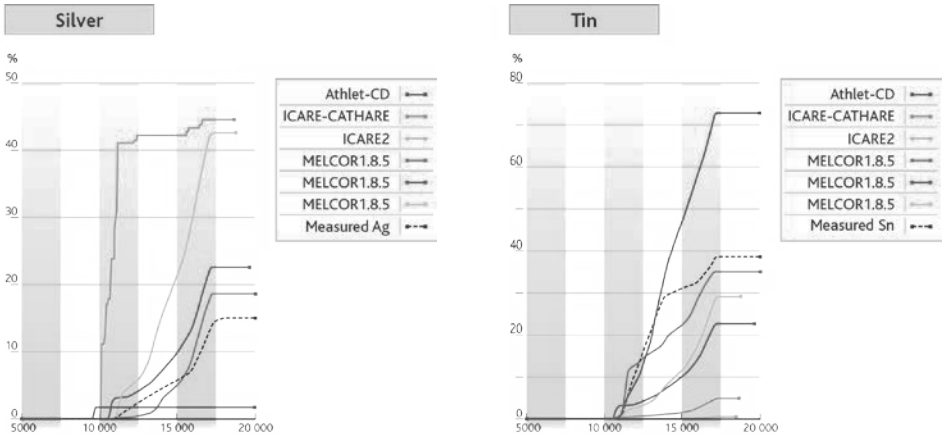



**Figure 7.17.** Silver (left) and tin (right) release kinetics during the FPT1 test, as simulated by various computer models (results from ISP-46 [26]). Comparison with experimental measurements (black dotted line).

Finally, models for uranium emission, one of the major constituents of the aerosols emitted and released into the containment, have also been reviewed on the basis of the results of the Phebus FPT4 test. A wide range of predictions for uranium emission from various computer models was clearly observed during simulations prior to the test, demonstrating the need to improve models. According to simulations performed after the test using the thermal conditions from the experiment, uranium emission from the lower part of the debris bed should have reached 60 g (for a total uranium mass of approximately 4.5 kg in the debris bed) [52]. However, a large unmeasured fraction of this uranium was deposited in the upper part of the bed, such that the calculated fraction of uranium released from the fuel is compatible with the value from the experiment, estimated at approximately 11 g following chemical analysis of the integral filters located downstream of the test device [22].

### *7.3.5. Transport of fission products and aerosols in the RCS*

During the Phebus FP tests, two main retention zones for aerosols and fission products were identified in the part of the test setup that represented the RCS. These zones, where wall and fluid temperatures dropped rapidly, were the vertical part of the hot leg immediately above the fuel assembly (cooling from around 1750 °C to 700 °C) and the steam generator rising leg (cooling from 700 °C to 150 °C).

During the first two Phebus FP tests, FPT0 and FPT1, most elements, with the exception of iodine, cadmium and some of the caesium, were transported as aerosols into the RCS hot leg, where the temperature was maintained at 700 °C. During these tests, iodine and cadmium were deposited in large quantities at the steam generator inlet *via* condensation of their vapours. In contrast, due to its lower volatility, caesium was deposited in equal quantities in each of the two zones with steep thermal gradients mentioned above. Finally, other elements, such as molybdenum and silver, which are even less volatile, were mainly deposited in the RCS vertical line located above the fuel assembly.

During the FPT2 test, several notable differences were observed with respect to the previous tests. In particular, besides the three elements cited above (I, Cs and Cd), indium and tellurium were also partially transported in vapour form at 700 °C. For each of these elements, the fraction transported in vapour form was relatively constant throughout the test and represented over half its total mass. An identical deposition rate in the steam generator hot leg was measured for these elements and for molybdenum.

Partial, temporary revaporisation phenomena for deposits on the RCS hot leg were clearly demonstrated during the FPT1 and FPT2 tests. Such phenomena were mainly observed for caesium after its releases from the fuel had ceased, and are explained by a reduction in the partial pressure of various caesium species in the fluid.

Tellurium seemed to have a very particular behaviour during the FPT2 test, with a large fraction deposited in the RCS upstream of the steam generator, possibly by chemisorption. In the long term, these deposits could lead to releases of iodine (daughter isotopes of tellurium).

Finally, various chemical forms of iodine, transported into the RCS hot leg in vapour form during the FPT2 test, have been brought to light *via* analysis of the condensates deposited on the walls of the sampling lines, in an area where the temperature dropped from 700 °C to 150 °C. Several chemical species were found: caesium iodide was only detected after the main cladding oxidation phase (mass-spectrometry detection of Cs and I in equal proportions) and other unidentified more volatile species, whose condensation temperatures were approximately 200 °C and within a range of 330 °C to 430 °C respectively (see Figure 7.18). The specific case of the presence of gaseous iodine in the RCS is covered in Section 7.3.7.

Analysis of aerosol and fission-product transport in the RCS during the FPT0 and FPT1 tests, performed using the ASTEC code [46], shows that calculations generally adequately predict the behaviour of vapours and aerosols, along with the overall quantities deposited in the RCS. However, fission-product retention is underestimated in the vertical line above the fuel assembly and overestimated in the steam generator. Underestimation of deposits in the vertical line may be explained by the fact that the fluid flow

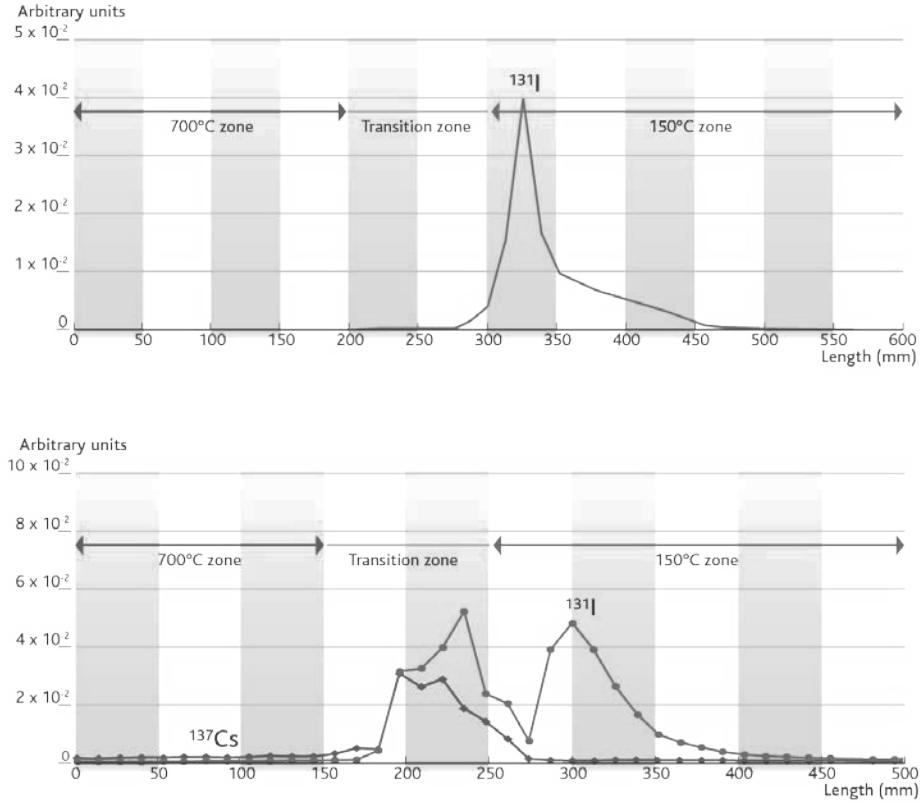



**Figure 7.18.** Caesium (in pink) and iodine (in green) vapour condensation profiles on the lines with a temperature gradient from 700 °C to 150 °C for the initial cladding oxidation phase (top) and for the fuel heating phase (bottom) during the FPT2 test.

in this area is not hydraulically or thermally established, given the steep temperature gradients and sudden changes in geometry (transition from flow in a fuel assembly to flow in a cylindrical pipe). Non-established flow conditions promote mass transfer to the walls. Furthermore, most simulation software overestimates retention in the steam generator rising line by a factor of 2. No significant improvement has been obtained even though a number of hypotheses have been put forward to explain this discrepancy [46] and two-dimensional simulations have been performed to follow particle paths [44].

With regard to fission-product speciation, it should be noted that, during the Phebus FP tests, caesium is mainly transported in condensed form from the RCS hot leg onwards. This clearly shows that the hydroxide (CsOH) is not the predominant chemical form, whereas this was generally assumed to be the case prior to the Phebus FP programme. According to thermodynamic equilibrium calculations, formation of caesium molybdate ($Cs_2MoO_4$) in the RCS would be promoted under the experimental conditions of the first two Phebus FP tests. This species is predicted by the ASTEC/SOPHAEROS simulation software (see Section 5.5.6) and has been introduced into other code packages such as MELCOR [40]. With regard to the FPT2 test, analysis of data from the tubes with thermal gradients and the sampling lines (see Figure 7.18) highlights the difficulties of simulating iodine species, in particular with the prediction of the presence of caesium iodide in all test phases, which is not always in agreement with experimental results [42].

### 7.3.6. *Thermal-hydraulics and aerosol behaviour on the containment vessel*

Thermal-hydraulics in the Phebus vessel that represents the containment building were mainly governed by the steam injection and condensation flowrates. Simplified simulations, performed using a coarse mesh to represent this containment (one or more compartments), reproduce the results of thermal-hydraulics measurements (temperatures, pressures, humidities, etc.) and the distribution of aerosols at the end of the test [26] fairly successfully. In line with the results of experimental measurements, they predict that the majority of the aerosols would settle on the elliptical lower head of the containment vessel, the remainder being deposited on the surfaces where water vapour condensed. In the simulations, for the first two tests, a small fraction would deposit on the side walls of the containment vessel (walls which were heated to prevent condensation of water vapour). The kinetics and distribution of aerosol deposits measured during the FPT2 test differ somewhat from the results of the previous tests, in particular with slower deposition kinetics (for all phenomena), a smaller fraction deposited on the surfaces where water vapour condenses, and a larger fraction deposited on the vessel side walls. These differences can be explained by the lower steam injection flowrate and consequent lesser condensation in the vessel during this test, and by smaller, less dense particles on average (less structural materials): the lower condensation rate reduced deposits on condensing surfaces and the smaller aerosol particles deposited onto the side walls more easily by Brownian diffusion.

The mass of aerosols retained on the elliptical lower head of the containment vessel and the condensing surfaces at the end of the Phebus FP tests is generally well simulated

by "point" code using standard models for aerosol deposition by sedimentation and diffusiophoresis [26 and 47], although for some models, the respective fractions of aerosols deposited *via* these two mechanisms were not accurate. A new model for particle deposition by diffusion onto the vessel side walls, based on a description of turbulence damping in the boundary layer near the walls, correctly predicted the mass of aerosols deposited on these surfaces [49]. Depending on the test, the calculated mass varied from 2 to 4% of the quantity of aerosols in the vessel. It was higher for the FPT2 test, in agreement with the experimental results. In summary, throughout the Phebus FP tests, "point" code adequately predicted the thermal-hydraulics in the containment vessel [48] and the aerosol physics.

### 7.3.7. Iodine chemistry in the containment vessel

One of the most unexpected results of the Phebus FP tests – a result which is safety related – was experimental evidence of the existence of a small volatile iodine fraction at low temperature in the containment vessel at a very early stage (as soon as fission products started to be released from the fuel) during the FPT0, FPT1 and FPT2 tests [45]. The gaseous iodine measured in the containment vessel during this phase was interpreted as coming from the RCS, contrary to the thermo-chemical models which predicted that all the iodine would be in condensed form (CsI) at its outlet. Kinetic limitations on the gas-phase chemical reactions involving iodine (incomplete reactions) are the most plausible explanation because none of the simulations that assume chemical equilibrium of the gaseous mix in the RCS correctly reproduce the experimental results [23 and 41]. These limitations could be produced by the specific thermal conditions in the RCS during the Phebus FP tests, with steep thermal gradients at the test device outlet and steam generator inlet. These limitations are accentuated for lower concentrations of fission products, which would explain the higher fraction of volatile iodine measured during the FPT0 test performed with very low burnup fuel and levels of fission products 50 times lower than in the FPT1 and FPT2 tests using higher burnup fuel.

The fraction of volatile iodine was found to be even greater during the FPT3 test. The reasons for this are still to be elucidated, but could be associated with the fact that the FPT3 test was performed using a boron carbide control rod instead of the Ag-In-Cd alloy rod used during previous tests. The "CHemistry of Iodine in the Primary circuit" (CHIP) experiment programme is underway as part of the ISTP [30] in order to better understand and quantify the phenomena involved.

The key role of the silver present in the control rod has been clearly shown by the Phebus FP tests, in particular the first two, which were characterised by large fractions of silver released. During the first two Phebus FP tests, iodine was detected and measured in the sump of the vessel representing the containment building, mainly in an insoluble form that was identified as silver iodide. The kinetics of the reaction between iodine and silver can be quite rapid under certain conditions, which leads to a suppression of the expected volatilisation of the iodine in the sump, following either radiolytic oxidation of the $I^-$ ions dissolved in the water, or the formation of organic iodides from the submerged painted surfaces (see Section 5.5.6 for further details). A set of analytical and semi-integral test programmes have quantified the kinetics

of the various silver-iodine reactions, meaning that the phenomena involved can be modelled [19 and 38].

The importance of the non-submerged painted surfaces of the containment vessel in the formation of organic iodides was clearly shown by the first two Phebus FP tests. In fact, this formation dominates that of organic iodides from submerged painted surfaces when the soluble iodine fraction in the sump is low (i.e. when releases of silver fractions significantly dominate releases of iodine fractions) [32]. For this reason, at least during the first two tests in the Phebus FP programme, organic iodides were the most common volatile iodine species measured in the containment atmosphere after approximately one day.

Generally, the Phebus FP tests demonstrated that the long-term concentration of volatile iodine in the vessel that represents the containment building (beyond 24 hours) mainly depends on the physicochemical phenomena affecting the gas phase, and therefore the concentration of volatile iodine arriving from the RCS or formed in the containment vessel. After one day, the concentration of iodine in the containment atmosphere remained constant, which shows that equilibrium was attained both for the reactions that create and destroy volatile iodine and for adsorption and desorption processes (see Figure 7.19). Furthermore, gas-phase radiolytic reactions, such as reactions involving air radiolysis products that break up molecular iodine and organic iodine, are key to long-term iodine speciation and therefore for assessing releases into the environment in the

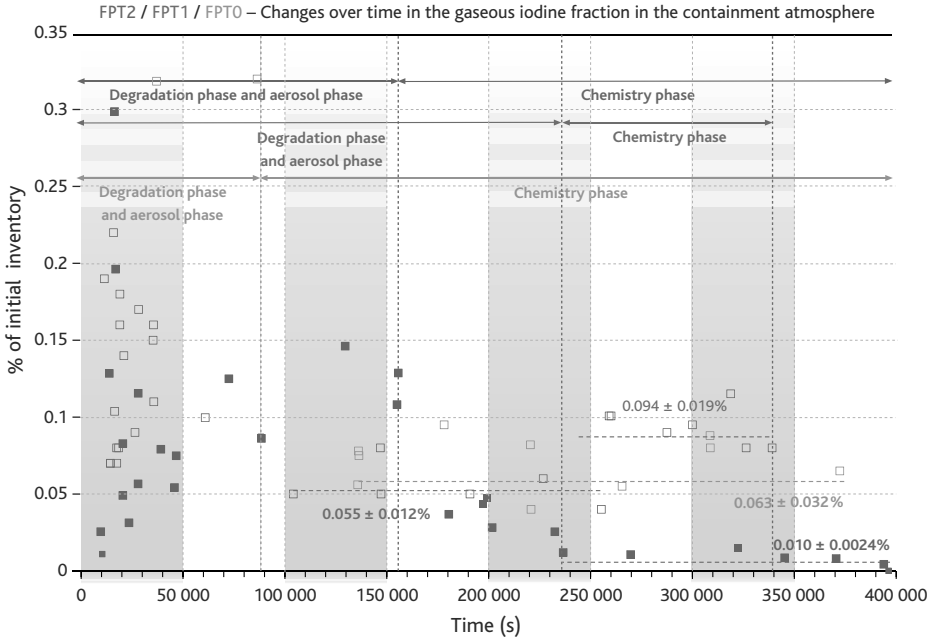



**Figure 7.19.** Changes over time in the concentration of volatile iodine in the vessel representing the containment building during FPT0, FPT1 and FPT2 tests (percentage of the total inventory in the fuel). Equilibrium values are given for the aerosol phase and chemistry phase for FPT1 and for the chemical phase for FPT0 and FPT2.

event containment failure on a PWR (venting *via* filters or basemat piercing). Radiolytic oxidation of volatile iodine species by ozone, nitrogen oxides and various radicals formed by air radiolysis, leads to the formation of iodine oxides in condensed form [33 and 39]. The concentration of volatile iodine in the containment after one to two days therefore depends on what happens to the products of radiolytic iodine oxidation, in particular their affinities for the surfaces inside the containment (paints, steel, etc.).

The analyses of all results concerning the behaviour of iodine during the FPT0 and FPT1 tests, performed in the context of the international Interpretation Circle[11] for iodine chemistry in the containment, are summarised in [41]. These analyses have contributed to improving understanding of iodine behaviour, in particular bringing to light:

- the probable occurrence of kinetic limitations during gas-phase chemical reactions in the RCS, which could account for the formation of volatile iodine at low temperature;
- the key role of silver from the control rods in liquid-phase iodine chemistry in the containment, the irreversible formation of insoluble silver iodide preventing the revolatilisation of molecular iodine by radiolytic oxidation of iodide ions and the formation of organic iodine in the liquid phase;
- the importance of the unsubmerged painted surfaces of the containment in the production of organic iodides when iodine is not very soluble in the liquid phase (e.g., when significant quantities of silver are released into the containment), leading to the production of mainly organic forms of volatile iodine;
- the equilibrium attained in the containment between the creation and destruction of volatile iodine, which leads to pseudo steady-state concentrations of iodine in the gas phase;
- the importance of gas-phase reactions between iodine and air radiolysis products in long-term iodine speciation (iodine oxides and nitroxides) and consequently in the assessment of possible releases into the environment in the event of core melt.

Lessons learned from the analysis of results concerning the behaviour of iodine during Phebus FP tests have been used to improve and develop models, which have then been integrated into most simulation software that covers iodine chemistry in the containment buildings of power reactors, such as:

- liquid-phase reactions between iodine and silver (taking into account the Ag metal/$I_2$(g), Ag metal/$I^-$ and Ag oxide/$I^-$ reactions, and oxidation of silver by water radiolysis products);
- radiolytic oxidation of molecular iodine and organic iodine *via* air radiolysis products and the formation of iodine oxides;
- formation of various organic iodides *via* the adsorption of molecular iodine onto unsubmerged painted surfaces.

11. The international partners of the Phebus FP programme share their interpretations of these test results. This is performed in the context of Interpretation Circles, one of which is devoted to iodine chemistry.

### 7.3.8. Use of Phebus FP test results in safety analyses

Knowledge acquired thanks to the Phebus FP programme has been put to use in accident simulation software such as ASTEC, MELCOR and MAAP. These computer codes, described in Chapter 8, are used to perform safety analysis; in some cases, the results of the Phebus FP programme can be used directly. One example is the use made by IRSN and the US Nuclear Regulatory Commission (NRC) with regard to the assessment of possible releases into the environment following a severe accident. In particular, at IRSN, this involved quantification of delayed and filtered releases and level-2 probabilistic safety analyses (PSA-2) [27]. The most recent results have been integrated into these analyses.

At US-NRC, a critical analysis, supported by the results of the Phebus FP tests, has been performed regarding the recommendations and assumptions gathered in Report NUREG-1465 [59], which covers releases in accident situations, on the basis of the opinions of an Expert Group [36], some members of which were from IRSN.

The results of the Phebus FP programme were also used to specify the research priorities for core melt accidents under the European EURSAFE project [50]. The lessons which continue to be learned from the Phebus FP tests are used by Severe Accident Research NETwork of Excellence (SARNET) [51]; in particular, research priorities are periodically reviewed [60]. The ISTP [27], which aims to reduce the uncertainties brought to light during the Phebus FP programmes, is part of this collaborative effort.

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