5.4. Retention and cooling of corium inside and outside the reactor vessel

5.4.1. In-vessel corium retention

5.4.1.1. Physical phenomena and associated safety issues

In-vessel corium retention assumes that reactor vessel integrity is preserved during an accident that causes reactor core melt.

Corium may be retained in the vessel either as a result of core reflooding leading to interruption of its melting, or flooding of the reactor pit with the aim of removing heat from the corium (debris or pool) while it is in the lower plenum of the vessel and thereby preventing vessel failure.

Research into in-vessel corium retention became one possible research option after the Three Mile Island 2 (TMI-2) accident in 1979 (see Section 7.1). During this accident, part of the core (approximately 20 tonnes of corium) was found at the bottom of the reactor vessel and the vessel did not fail, see [1]. The resistance of the reactor vessel to the thermal stresses caused by the residual heat released by this corium was attributed [2] to the fact that the molten corium flow at the bottom of the vessel was underwater (reactor flooded and pressurised (~ 100 bar)), but this has never been completely confirmed.

However, core reflooding may not be beneficial in all conditions. The following phenomena can occur during reflooding:

– massive steam generation, with hydrogen production and an increase in reactor coolant system pressure;
– steam explosion through corium-water interaction;
– continuation of core melt, despite water inflow;
– faster release of fission products.

Theoretically, reflooding could take place in all possible core configurations (fuel rods intact, rods slightly damaged but with ballooned cladding, rods melted leading to one or more flows of molten material, debris bed, corium pool, etc.). It is therefore necessary to determine the effects of reflooding on the subsequent development of the accident, based on the core configuration at the time when water is injected into the core.

After the TMI-2 accident, studies into the possibility of in-vessel corium retention following reactor pit flooding (cooling from outside the vessel) led to reactor designs 15 years later that incorporated this possibility (examples include the Westinghouse AP600 and AP1000 in the USA [3], the KAERI APR1400 in South Korea and the European ESBWR. However, it remains difficult to demonstrate the effectiveness of external cooling in retaining corium inside the vessel.

5.4.1.2. In-vessel corium retention through RCS flooding

5.4.1.2.1. Physical phenomena and state of current knowledge

► Conditions with fuel rods intact or only slightly damaged

If the core is reflooded when the fuel rods are intact or only slightly damaged (rod temperatures between approximately 1200 °C and 1800 °C), high levels of hydrogen can be generated, as demonstrated by the QUENCH test results by Forschungszentrum Karlsruhe (FzK) in Germany (programme described in Section 5.1.1.3.1). The speed of the cladding oxidation reaction caused by the steam depends on the cladding temperature and the steam flowrate through the core, which is itself related to the progress of the quench front. Current software is able to satisfactorily estimate the progress of the quench front for a geometry with fuel rods intact. For these conditions, the model has been validated by various experimental results [25] (including the PERICLES tests by the CEA and the RBHT tests at the University of Pennsylvania in the United States). The thermal-hydraulic models are still sufficiently accurate when the fuel rod cladding begins to undergo deformation or the first molten material flows appear, because these deformations are not significant enough to lead to major disruption of the flow patterns. The main uncertainties are due to imprecision regarding rod geometry at the time of reflooding (in particular, the heat exchange surface) and the related laws of heat exchange. For these conditions in which core geometry has not been significantly altered, if the flowrate is high enough, core damage is likely to be halted, provided that reflooding does not cause the mechanical destruction and collapse of a large proportion of the fuel rods due to thermal shock. Any debris bed obtained in this way could no longer be properly cooled. The conditions under which the fuel rods might collapse and the size of the resulting debris are unknown, but interesting data has been deduced from test results from an OECD programme performed in the HALDEN reactor at the Norway Institute for Energy Technology (accumulation of irradiated fuel pellet debris in a rod that swelled during a LOCA). The ISTC-1648 test programme, funded by the International Science and Technology Centre (ISTC) and performed by NIIAR in Russia (Research Institute of Atomic Reactors), which aimed to study the reflooding of a length of irradiated fuel rod (see Section 5.1.1.3.1) also provided some data. The evidence gathered can be summarised briefly as follows, based on three fuel rod temperature ranges:

- below 1200 °C, it is unlikely that the fuel rods will fragment and only small amounts of hydrogen will be released due to cladding oxidation; the core can therefore be cooled if the water flowrate is high enough;
- between 1200 °C and 1600 °C, the fuel rods may become fragmented and collapse, forming a debris bed if the cladding is embrittled by significant oxidation; if the rods do not collapse, hydrogen production remains low and the core can without doubt still be cooled if the steam flowrate is high enough;
- above 1600 °C, oxidisation of the zirconium alloy cladding leads to a runaway oxidation reaction, resulting in high hydrogen production and major rod damage, possibly with flows of liquefied materials. The core can no longer be cooled, at least locally in the places with molten material flows had occurred.

► Conditions with debris bed formation

If the fuel rods collapse inside the core, the fuel fragments form a porous medium known as a debris bed. If a debris bed forms, the pressure loss increases significantly and makes it a lot more difficult to access the collapsed areas. If the flooding water cannot reach some parts of the debris bit, these parts can only be cooled if the steam flow produced downstream at the quench front is sufficient, otherwise they heat up to melting temperature, creating a molten pool of core material. A debris bed can also form at the bottom of the reactor vessel when the corium flows through the water. The maximum heat that can be removed from a debris bed by water, before it dries out and melting occurs, is called the "critical heat flux". It is expressed per m² of the upper surface of the debris bed. The phenomena that occur when a debris bed is reflooded are satisfactorily understood, because a number of experiments have been performed and various models developed since the 1980s. However, the only data developed are point models or 1D models, validated with 1D experimental results. There is still uncertainty about the extrapolation of reflooding calculation results to multi-dimensional and heterogeneous geometries. In particular, some experimental calculations and observations, which unfortunately are incomplete, suggest that the heat removed from a debris bed for a multi-dimensional configuration may be higher than the heat removed from a debris bed in a one-dimensional configuration (possibly as much as twice as high), and that even after the debris bed has dried out, the steam flowing through the bed may keep some of the debris below melting point. However, much uncertainty remains, because in the TMI-2 accident (discussed in detail in Section 7.1), the heterogeneity of the debris formed (including the presence of liquid "pockets") and the presence of small debris (< 1 mm) may be the reason why corium melt could not be prevented after reflooding. Reflooding of a heavily damaged core or a debris bed remains poorly modelled by calculation software.

The main multi-dimensional thermal-hydraulic models that exist for a porous medium are included in the ICARE/CATHARE (IRSN), WABE (IKE/GRS) and MC3D (CEA/IRSN) software packages. The multi-dimensional effects on heat removal through reflooding in particular remain to be confirmed with experimental data from sufficiently large experimental set-ups that can provide reliable local temperature measurements and steam generation during reflooding. This is the aim of the PEARL programme of experiments, initiated by IRSN in 2010, in partnership with EDF, with the participation of the European network SARNET-2.

► Conditions with a liquid corium pool

What happened at the TMI-2 accident? When the corium flowed to the bottom of the reactor vessel, the vessel was full of water. Around ten tonnes of corium in oxide form (approx.1 m^3) had flowed in compact form to the bottom of the reactor vessel and around ten further tonnes of debris were above the corium in compact form. Analysis of reactor vessel samples showed that the temperature of the internal surface in contact with the compact corium mass at the bottom of the vessel reached approximately 1100 °C and the external surface reached approximately 800 °C. The internal pressure in the reactor vessel was around 100 bar at the time. The vessel then cooled very slowly. Assuming perfect contact between the compact corium and the reactor vessel, all thermal calculations have shown that the reactor vessel temperature should have continued to rise, eventually causing vessel failure. The explanation put forward as to how the reactor vessel withstood these conditions assumes that a gap formed between the corium and the vessel. According to this assumption, the gap would have formed due to two phenomena:

- traces of water in porosities within the steel boiling and preventing contact between the corium and the steel;
- a process of differential expansion between the solidifying corium and the reactor vessel, which was heating up.

It is thought that the ingress and flow of water in this gap cooled the reactor vessel sufficiently and sustainably enough to prevent failure.

Some experiments have been performed to try to confirm this hypothesis of a gap between the corium and the reactor vessel in the TMI-2 accident. Small-scale tests have been performed on corium flows at the bottom of reactor vessels containing water, to reproduce the TMI-2 corium flow conditions and analyze the results. Such tests have been performed by FAI (FAUSKE & Associates, Illinois, USA) [11], JAERI (Japan) and KAERI (South Korea) [12]. All these tests were carried out with an alumina thermite mixture to simulate the corium. Other tests to determine the maximum power (or critical heat flux) that could be removed be water flowing in a gap between the corium and the reactor vessel have been carried out by IBRAE (Russia) [13], Siemens (Germany) [14] and KAERI [15].

These tests have not been highly conclusive and the main finding is that the gap is likely to have been formed solely by the traces of water boiling. The hypothesis of differential expansion appears less plausible. However, as suggested by a CEA study [4], the possibility of heat removal through water boiling in a gap is also very slim. It remains difficult to explain with certainty why the reactor vessel withstood failure in the TMI-2 accident conditions.

In more general terms, for a power reactor core meltdown accident, this CEA study shows that the heat (or critical heat flux) that can be removed by boiling water in a gap is very approximately proportional to the square root of the pressure. For instance, for a 3 mm gap and pressure of 1 bar, the critical flux is of the order of 0.02 MW/m^2, which should be compared with the 0.5 MW/m^2 that needs to be removed if half the mass of the core were at the bottom of the reactor vessel in the form of a corium pool. The conclusion is that too little is known about the real conditions at the bottom of the reactor vessel (Is water present permanently or not? What is the critical flux value? Is there a gap in the event of meltdown? etc.) for the cooling mechanism through gap formation to be considered plausible for most foreseeable core meltdown accident conditions on a PWR at low pressure.

Without more pertinent experimental results, with the reactor coolant system depressurised and with no reactor pit reflooding, it would appear difficult to demonstrate that reflooding the reactor coolant system would prevent reactor vessel failure once a large pool of molten material has formed within the core.

5.4.1.2.2. Experimental programmes

The main programmes of experiments studying core reflooding for pressurised water reactors are LOFT-FP, PBF-SFD, CORA, QUENCH, ISTC 1648 (QUENCH) and PARAMETER. These programmes are briefly described in Section 5.1.1.3.1.

5.4.1.2.3. Review and future outlook

Two reviews [22, 23], dating from 2005 and 2006, summarise current knowledge of the various risks associated with PWR core reflooding. These documents identify the main uncertainties and the R&D programmes that will be required.

The themohydraulics and fuel behaviour during a core meltdown accident require finer modelling in order to better understand accident development in the core of a power reactor. This implies more precise and detailed modelling of transient conditions, in particular the two "key" transitions from a damaged core to a molten pool and then from a molten pool in the core to a molten pool at the bottom of the reactor vessel. The models used in the software for fuel rod deterioration in the core are based on a multi-dimensional description of the material transfers in order to better calculate the transient changes to materials in the reactor vessel, but there are no experimental results from relatively large-scale tests to deal with the scale effects and validate these multi-dimensional models.

The three priorities for further study are as follows:

1. the geometric evolution of a heavily damaged core or a debris bed during reflooding (can a damaged core be cooled or not?); tests will be required to more clearly understand the progress of the quench front in a damaged core, according to its geometry, particularly for conditions involving a debris bed and the specific geometric features of debris from irradiated fuel rods. The size distribution of the debris will be a significant result that could be obtained through additional out-of-pile tests with real rods. This is the aim of the reflooding tests with lengths of fuel rods under the ISTC 1648 (QUENCH) programme;
2. the evolution of a dry debris bed and its transformation into a molten pool (if it cannot be cooled); it would be good to study dissolving and oxidation, two phenomena that have an impact on stratification in the pool;
3. the arrival of corium at the bottom of the reactor vessel, in particular when it is full of water; it would be good to study corium fragmentation, oxidation and cooling when it gets into the water and when it spreads at the bottom of the vessel. These issues were partially examined in the programmes that looked at steam explosion (see Section 5.2.3).

5.4.1.3. In-vessel retention with reactor pit flooding

5.4.1.3.1. General approach: orders of magnitude

Two main parameters affect the integrity or otherwise of the reactor vessel under core melt accident conditions with molten corium flowing to the bottom of the vessel:

- the mechanical strength of the reactor vessel at all points, particularly in areas subjected to the highest thermal load;
- the mechanical strength of the reactor vessel to withstand a steam explosion caused by an in-vessel corium water interaction.

Order of magnitude calculations have shown that, following core degradation, the materials in the core of a PWR 900 would take up a volume of a similar order of magnitude to the hemispherical vessel bottom, if they formed a very compact mass at the bottom of the reactor vessel with no voids (e.g. a corium pool). Assuming that the residual power of these materials is 20 MW and that they emit a uniform heat flux, the heat flux calculated at the edge of the pool is of the order of 0.8 MW/m^2. This heat flux is extremely high and can only be removed if there is efficient convection at the free surface of the corium pool and the interfaces between the corium pool and the reactor vessel. Even in this case, part of the vessel wall would melt and its residual solid thickness would only be a few centimetres. With a simple calculation, it can also be shown that if this heat flux is not efficiently removed (e.g. if there is no steam flow above the corium pool), the reactor vessel will be perforated after only a few minutes. Ensuring vessel integrity therefore requires a way of removing the heat flux from the corium pool at all points in the reactor vessel. This condition is essential, but it is not the only criterion. The weakened vessel also needs to continue to withstand the pressure. Given that the residual thickness of the steel is reduced, the reactor vessel cannot resist high pressure in the reactor coolant system, requiring the RCS to be depressurised. The mechanical strength of the reactor vessel is therefore assessed at final pressure, after depressurisation, taking into account the thermomechanical loads caused by the corium pool. It also needs to be evaluated for a pressure peak in the reactor coolant system. A pressure peak could, for example, result from a steam explosion following inflow of water from the RCS onto the corium pool at the bottom of the reactor vessel [10].

5.4.1.3.2. Mechanical strength of the vessel depending on corium pool configuration

To assess the mechanical strength of the reactor vessel when in contact with a corium pool in core melt accident conditions, vessel behaviour is studied under the worst-case limit conditions, which are no core reflooding, flow of all corium mass to the vessel bottom and stationary thermal-hydraulic conditions in the molten pool. These are the conditions in which the highest heat flux is received by the reactor vessel.

The heat flux distribution on the vessel wall depends on the configuration of the corium at the vessel bottom (whether or not it is stratified). The core materials can be distributed according to their respective densities in order to define the various possible corium configurations for a given core inventory (masses of oxides, zirconium and steel in the core). There are however other parameters that affect the corium pool configuration:

- the degree of zirconium oxidation (which can range between 25% and 80% depending on the accident scenarios considered, see Section 5.1.2.1);
- the mass of molten steel (between a few tonnes and several tens of tonnes);
- the possibility of solid layers (debris and solid crusts), in particular at the corium pool interfaces.

One of the most critical configurations for the reactor vessel is when low-density molten metals (mainly containing steel) float on top of a pool of high-density corium "oxides" (approximately 8000 kg/m^3) (Figure 5.46). These are the bounding conditions in terms of thermal loading on the reactor vessel that have been most extensively studied and for which the limit conditions and heat transfer to the pool have been determined. This configuration was also used to support the first external vessel cooling studies, in particular the AP600 concept. This configuration will be referred to as the "reference configuration" hereinafter.

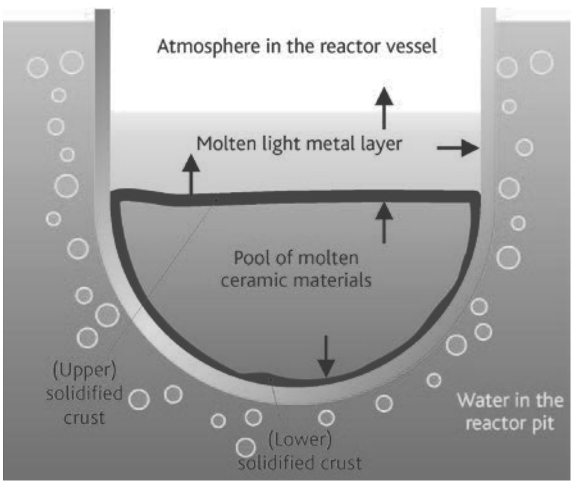

Figure 5.46. Configuration of the corium pool at the bottom of the reactor vessel with external cooling.

5.4.1.3.3. Study of the corium pool stratified configuration

► Heat flux distribution and cooling for the stratified configuration

For any given corium configuration, the heat flux distribution depends on the limit conditions between the melting mass and the solid wall (either the crust or the steel of the vessel) and the coefficients of heat transfer by natural convection. The temperatures at the edge of a corium pool have been determined in various studies, which are summarised in [5]. The main difficulty in determining the temperatures is related to the fact that the melting materials are a mixture of oxides and metals. These mixtures melt over a relatively wide range of temperatures that depends on the composition of the mixture. Such a mixture may also contain a soft zone, between the molten pool and the solid crust by the vessel wall, which could affect heat transfer. Reference [5] has shown that in steady-state thermal-hydraulic conditions (i.e. when the heat fluxes have been established), no such soft zone can exist, because the pool composition becomes homogeneous and will solidify in the same way as a pure body (with a flat interface between the solid and liquid). In addition, when there is sufficient external cooling, the solid crust has a constant thickness (i.e. the speed of progress of the solidification front is zero). In this case, the temperature at the liquid-solid interface tends towards the liquidus temperature corresponding to the liquid mixture. There is a clear separation between the solid and liquid. Experimental confirmation of this conclusion has been provided by various tests (PHYTHER by the CEA (described in [5], RASPLAV (Kurchatov Institute, Russia) [16], and SIMECO (Royal Institute of Technology, Sweden). The solidification transient was studied by IRSN in 2005 [17].

The assumption of thermochemical equilibrium in determining the interface temperatures also applies to the metallic layer of a stratified corium pool. If the liquidus temperature corresponding to the composition of the metallic layer (chiefly formed of steel and zirconium) is lower than the melting point of steel, the steel may be dissolved by the molten metal. The interface temperature with the solid steel of the reactor vessel establishes itself at this liquidus temperature. To put things simply, depending on the composition of the liquid metal layer, the temperature of the inner surface of the reactor vessel wall may be substantially lower than the melting point of steel. Temperatures at the liquid-solid interface are calculated with thermodynamic software (such as GEMINI) on the basis of the composition of the liquid layer in question. The corollary of this choice is that the pool is completely liquid and the heat transfer laws identified from tests with simulation materials (pure bodies like water) can be transposed to the real materials.

Heat transfer correlations have been deduced from tests with simulation materials (BALI, COPO, ACOPO, RASPLAV-Salt, etc.) for various geometrical configurations [9]. Efforts have also been made to validate CFD software for natural convection. The results are encouraging, but further improvements to the turbulence model are still required in order to improve the precision of the results. The use of such software on the scale of a power reactor vessel gives results with a wide uncertainty interval. Given the current state of knowledge, it is preferable to use a simpler approach based on correlations from the tests.

► Order of magnitude of the heat fluxes and focusing effect

To give an order of magnitude, for the reference configuration shown in Figure 5.46, the residual heat is distributed as follows, assuming that the entire mass of oxides from the core is at the bottom of the reactor vessel:

- half the residual heat released from the pool of oxides is transferred to the bottom of the vessel;
- the other half is transferred from the pool of oxides to the upper layer of liquid metals.

If there is no water inside the reactor vessel, the metal layer transfers most of the heat received from the pool of oxides and the internal heat it releases to the steel vessel wall, which is in contact with the liquid metal layer. The metal layer can generate a heat flux "focusing effect" on the wall surface that is in contact with the liquid metal. At the point of contact with the metal layer, the heat flux is very approximately inversely proportional to the thickness of the metal layer. For a thickness of more than 50 cm (corresponding to approximately 50 tonnes of steel), the heat flux is below 1.5 MW/m^2. Reactor vessel integrity is only assured if the heat flux transferred to it can be removed by two-phase natural convection from the cooling water outside the vessel. This naturally raises the question of the critical flux on the external vessel wall (upper limit higher than the heat flux that can be removed by external reactor vessel flooding).

► Critical flux for natural external water circulation

The critical heat flux associated with external cooling of the reactor vessel, in particular in the area around the metallic layer, will therefore by the limiting factor for heat removal from the vessel. Significant efforts have been made around the world to determine this critical flux and to increase it. Various tests have been performed (with 2D or 3D geometries and different wall heating modes). The most interesting of these include tests by ULPU (University of California, Santa Barbara) [19], the SULTAN tests (CEA) [18] and tests by KAIST (Korea Advanced Institute of Science and Technology, South Korea).

The first phenomenon that determines the critical flux value if the reactor pit is reflooded is the water circulation by natural convection within the reactor pit. Simply reflooding the reactor pit is not enough to cool the reactor vessel. The water circulation needs to be organised such as to "maximise" liquid flowrate along the vessel walls. This implies the existence of a "rising hot leg" (the reactor vessel) and a cold leg. The geometry of the vessel (radius and spherical or elliptical shape of the vessel bottom) and the presence of insulating materials around the vessel may affect water circulation and pressure loss. For a geometry that maximises water circulation and in the absence of elements to hinder flow, maximum critical heat flux is obtained when the water flow-rate is high enough to limit boiling close to the wall in the heating zone (no mass boiling in this area). However, above the heating zone, boiling should be higher to create a strong enough "chimney effect", whereby the steam generated drives an increased liquid flow. If water flowrate is not high enough, there is mass boiling around the heating zone and the critical flux is reduced because the heat is removed less effectively. Having said that, the water flowrate cannot exceed the flow created by the chimney effect related to mass boiling above the heating zone. This maximum flowrate corresponds to a maximum critical heat flux of the order of 1.5 MW/m^2.

Analysis of the test results mentioned above show that the spread of estimated critical flux values is often fairly high. Results from the ULPU tests give values close to 2 MW/m^2 (but with a wide spread of experimental results), whereas results from the SULTAN and KAIST tests show critical flux values on a vertical wall that range from 1.2 to 1.5 MW/m^2.

Various effects have been studied in an attempt to identify provisions that could increase the critical heat flux, in particular effects linked to the condition of the reactor vessel's outer surface. According to some authors, [6], a spray-on porous metal coating on the outer surface of the reactor vessel could significantly increase the critical flux (by a factor of up to 2). However, this conclusion is not universally shared and experimental verification is still required.

► Limitation linked to vessel mechanical strength

For a heat flux of 1.5 MW/m², the vessel thickness supporting the mechanical load (i.e. the place where the temperature is below 600 °C) is 1 centimetre. This thickness can withstand pressures up to a few tens of bar. An increase in critical flux would lead to an inversely proportional decrease in the thickness supporting the mechanical load, and a consequent reduction in the yield pressure of the reactor vessel. These considerations strongly temper the potential benefits of any work to demonstrate critical flux values of above 2 or 3 MW/m².

► Limitation linked to the minimum mass of molten steel

One of the key parameters that determines the thickness of the metallic layer above the oxide pool, and hence the highest thermal loads on the reactor vessel, is the mass of molten steel in the corium produced by core melt. For a critical heat flux of 1.3 to 1.5 MW/m², the minimum thickness of molten steel required to prevent the thermal focusing effect is of the order of 50 to 60 cm for a 1000 MWe PWR. Given the characteristics of these reactors, this thickness corresponds to a mass of molten steel of the order of 50 to 60 tonnes. According to studies by Westinghouse for the AP600 and AP1000, this quantity of steel would be found at the bottom of the reactor vessel after the lower in-vessel structures and part of the vessel walls have melted. Findings from the OECD MASCA programme [20] suggest that complex physical phenomena could reduce the mass of metal and lead to a focusing effect. These phenomena are as follows:

- part of the liquid metals (e.g. the lower in-vessel structures) getting trapped in solid oxide debris;
- part of the molten metal flowing to the bottom of the reactor vessel due to physico-chemical effects related to the presence of non-oxidised zirconium (details below);
- for a fixed quantity of metal, this would lead to reduced thickness of the metallic layer on the top of the corium pool (see Figure 5.47).

The physico-chemical effects are linked to the presence of non-oxidised zirconium in a metallic phase. This zirconium can react with the uranium dioxide in the oxide phase and lead to the formation of a uranium metal phase. This phase can mix with the liquid steel and lead to the formation of a liquid metal layer that is denser that the pool of oxides that would now be at the bottom of the vessel. By using thermodynamic software (such as GEMINI2 from Thermodata), the composition of complex metal-oxide mixtures can be calculated in equilibrium at various temperatures. If the density of the phases resulting from these calculations is determined, the maximum mass of metal that might end up below a pool of oxides can be estimated and, by subtraction, for any given quantity of steel, the mass of metal present in the upper layer. This method was used by CEA and IRSN to calculate the mass of metal necessary to prevent the heat flux transferred from the metal layer to the reactor vessel exceeding the critical flux [7]. Calculations were performed for different reactor types: the French 900 MWe and 1300 MWe PWRs, the AP600 and AP1000 reactors developed by Westinghouse, and the Korean APR1400 reactor. The results showed that one key parameter is the fraction of non-oxidised zirconium present in the molten pool. The higher this fraction, the greater the mass of metallic uranium produced and the greater the mass of metal at the bottom of the reactor vessel. The question of keeping the corium inside the vessel is thus more complex if the mass of metallic zirconium is higher. The results are also sensitive to the databases used for thermodynamic calculations and the critical flux values outside the vessel. In particular, for reactors of greater than 600 MWe, a natural circulation system needs to be set up in order to remove a high heat flux.

It should be noted that the above studies were performed for a stationary corium pool configuration. The formation of metallic layers and the pool of oxides will necessarily involve transients of growth in the metal layer thickness and increase in the heat released by the pool of oxides. These transients were not incorporated in the calculations, but they could cause the critical heat flux to be reached.

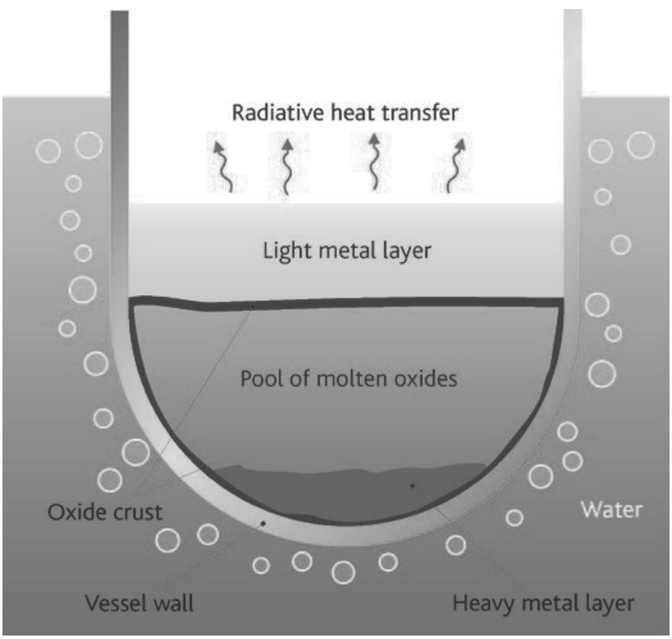

Figure 5.47. Corium recovery from the bottom of the vessel with inversed stratification of metals and oxides.

5.4.1.3.4. Possible progress for in-vessel corium retention with reactor pit flooding

The studies described in the foregoing paragraph do not yet demonstrate that reactor pit flooding would ensure, for any given reactor type, that the corium would be retained inside the reactor vessel for all foreseeable core melt accidents. Further studies are required on the following issues, in particular:

– more realistic corium configurations, in which the metallic layer can be in various positions (above or below the pool of oxides) over time;

– the possibility of simultaneous external (reactor pit) flooding and internal core reflooding;

– conditions that could lead to several successive corium flows within the reactor pit, for which the heat fluxes on the reactor vessel may differ significantly from the very schematic situation usually considered (see Figure 5.46).

An inflow of water onto the corium pool inside the vessel could eliminate the heat flux focusing effect. The ANAIS tests by CEA [8] have shown that, in this case, the metal layer on the surface could solidify, transferring a significant proportion of the residual heat to the water. These same ANAIS tests also showed that, under these conditions, the steam explosion risk would be limited to the area in which the water spray hit the liquid corium. A large explosion following accumulation of water seems unlikely because the surface of the corium pool would quickly be solidified by a significant inflow of water.

In order to better assess the possibility of in-vessel corium retention for reactor pit flooding, it will be necessary to improve the model of corium flow to the bottom of the reactor vessel and development of the corium pool at the vessel bottom.

For 900 MWe and 1300 MWe reactors, there are not currently provisions to ensure in-vessel corium retention for all foreseeable core melt accidents. Reactor vessel failure (Section 5.1.3) and the possibility of cooling the corium outside the vessel (Section 5.4.2) at the time of molten core-concrete interaction (MCCI) have therefore been studied in detail.

For the EPR reactor design, specific provisions have been adopted (the corium spreading and cooling compartment, presented in Section 5.4.3) in order to cool the corium outside the vessel.

5.4.2. Cooling of corium under water during MCCI

5.4.2.1. Physical phenomena involved

One possibility for accelerating cooling of a corium pool during MCCI (see Section 5.3), and stopping its development, would be to direct water into the reactor pit onto the corium surface.

Radiative heat transfer between the corium pool and the reactor-pit walls leads to the formation of a crust on the surface of the corium, due to the high solidification temperature of corium (around 2400 K for a corium containing little concrete). This crust would apparently be thicker if the corium were covered by water, but would also act as an insulator between the corium pool and the water, thereby restricting heat transfer between corium and coolant. Order of magnitude calculations show that if heat transfer between pool and water were only by conduction *via* the crust, then the slowing of concrete erosion due to directing water onto the corium would be minimal. For the cooling of corium under water to be truly effective, other heat transfer mechanisms would need to be involved. The purpose of R&D work (experiments and models) performed on the subject is to identify and quantify the effectiveness of these other modes of heat transfer.

5.4.2.2. Experimental programmes

The main experimental programmes that have been performed on this subject are: the Melt Attack and Coolability Experiments (MACE, see [26]) programme performed at the Argonne National Laboratory (ANL, USA) from 1989 to 2010 using real materials, MSET (see [27]) and OECD-MCCI (see [28] and [29]) – with the last programme divided into three sub-programmes, namely SSWICS, MET and CCI –, plus CEA's PERCOLA programme using simulation materials, see [30]. The ANL programmes involve both integral experiments and more analytical tests.

5.4.2.2.1. MACE and CCI tests

These integral tests aimed to study the possibility of cooling the corium during an MCCI by directing water onto the pool surface, using materials representative of corium formed during a core melt accident on a power reactor. Three tests were performed with 1D devices (concrete erosion only in the downwards direction: M1B, M3B and M4) and five tests with 2D devices (concrete erosion downwards and on the sides: M0 in the MACE programme, and CCI-1, CCI-2, CCI-3, and CCI-4 in the OECD-MCCI programme). Test performance was essentially the same for all tests; it initially involved forming a corium pool, with a composition representative of that of a core melt accident on a power reactor at the beginning of MCCI, by using a thermite reaction (a highly exothermic reaction, which for these tests involved a mixture of U_3O_8, CrO_3, CaO, SiO_2, silicon, zirconium and aluminium) that produces a molten mixture mainly made up of UO_2 and ZrO_2 along with a smaller proportion of oxides representative of concrete erosion (mainly SiO_2, CaO, etc.), alumina and chromium oxide. The corium pool was then maintained in a molten state by direct heating. Concrete erosion was initially obtained *via* a dry MCCI. Water was then directed onto the corium after a time delay or a maximum specified ablation, and MCCI continued under water. The effectiveness of directing water onto the corium can be understood by comparing concrete erosion rates with and without water, and by measuring the heat flux at the surface of the corium pool (associated with the quantity of steam produced). Pool temperature is also an indicator of the effect of water supply. However, it is important not to directly extrapolate test results to the case of a power reactor, to the extent that tests involve non-representative aspects: in particular, the corium is heated *via* the Joule effect in the liquid corium, whereas for a power reactor, the residual heat would be spread between the liquid (pool) and the solid (crust).

The tests mentioned above brought to light several possibilities for corium cooling by water:

– during numerous tests (M0, M3B and CCI2), part of the corium pool was entrained by the gases produced by concrete decomposition and ejected above the pool's upper crust, forming a bed of centimetre-sized debris. Furthermore, analytical tests have shown that it is possible to cool debris of this size dispersed in water but it has not been shown that cooling of a thick bed of debris emitting significant residual heat would be as effective as cooling of dispersed debris;

– the pool's upper crust could crack and water could penetrate under this crust, due to the effect of temperature differences between the water and the corium and thermomechanical stresses. This water could propagate in the pool and cool it completely (a mechanism called "water ingression"). However, models that describe this mechanism suggest that the cracks created by temperature differences would be too small for water ingression alone to effectively cool a corium pool, see [31]. Nevertheless, the presence of cracks plays an important role in the thermomechanical behaviour of the crust (cracks reduce the mechanical resistance of the crusts) and could contribute to corium cooling;

– effective cooling of the corium surface has been measured during direct contact between water and liquid corium. This highly transient phenomenon can occur during sudden mechanical failures of the crust or when water first arrives on the corium pool. However, during the tests performed, it may have been promoted by the geometry of the experimental set-up; it is not possible to directly extrapolate this result to a power reactor.

Under these conditions, it is not possible to conclude on the effectiveness of cooling of a corium pool during an MCCI by directing water onto the surface of the pool for a power reactor, although it would seem that the various phenomena mentioned above would tend to slow concrete erosion. The performance of more representative tests runs up against technological difficulties that limit the scope of the experiments and study of the phenomena:

– given the limited scale of existing test equipment, in most cases a crust forms on the upper part of the pool and bonds to the walls of the test section. As concrete erosion progresses, the liquid corium descends and separates from the crust; this separation limits the effectiveness of corium ejection. In the case of a power reactor, it is more likely that the crust would remain in contact with the liquid corium due to the size of the reactor pit;

– direct corium heating means that it is not possible to heat the solid crusts. The solidification observed during the tests is therefore not representative of that which might occur on a power reactor.

5.4.2.2.2. MSET test

The purpose of the MSET test, performed in 2001, was to study corium ejection through the crust, a phenomenon brought to light during the MACE tests. The MSET test was performed with materials representative of corium formed during a core melt accident on a power reactor, without concrete erosion and with water directed onto the upper part of the pool. Gas release was simulated by using a porous material at the base of the corium pool, through which gas was injected at a controlled rate.

The MSET test led to the formation of bed of debris but no corium ejection was observed for superficial gas velocities below 10 cm/s, which posed the question of the effectiveness of such a phenomenon for a power reactor, where the superficial gas velocity would be less than 5 cm/s during the long-term MCCI. However, analysis of the MSET results brought to light as possible causes of this behaviour:

- bonding of the crust onto the walls of the test section, leading to separation of pool and crust;
- the presence of a significant solid fraction, due to pool temperature (well below the liquidus temperature).

The results of this test do not therefore provide insight into the importance of corium ejection for the cooling of corium during an MCCI on a power reactor.

5.4.2.2.3. SSWICS tests

The purpose of the SSWICS tests (see [31]), performed with materials representative of corium formed during a core melt accident on a power reactor, was to study the mechanism of water ingression after thermomechanical cracking of the corium pool's upper crust. Water penetrates *via* this mechanism into the cracks which form in the upper crust when cold water comes into contact with the hot crust; the cooling of corium under the crust leads to its solidification, which increases crust thickness.

Under the SSWICS programme, separate-effect tests were performed without heating the corium pool, and with simulated release of concrete gases for some tests. The corium pool, which was produced in a test section using a thermite reaction similar to the one described in Section 5.4.2.2.1, sat on an inert support. Water was gradually directed onto the corium pool and the cooling kinetics were deduced from the water vaporisation rate. The effectiveness of water ingression was assessed by comparing the heat flux extracted during the tests with that obtained under conditions where only conduction was involved (thermite cooling without water). The permeability of the crust was measured after the tests, which meant that the removed heat fluxes could be assessed using specific models.

The tests performed (see Figures 5.48 and 5.49) enabled quantification of the influence of concrete type (siliceous or silico-calcareous), corium pool composition (between 4% and 25% concrete by mass), gas injection, and pressure (between 1 and 4 bar). The lumps of corium obtained at the end of the tests were cut into pieces and subjected to mechanical resistance tests.

The main lessons drawn from the SSWICS tests were the following:

– these tests confirmed that cracking of the upper crust and water ingression into these cracks cools the corium in certain cases (increasing crust thickness). With no gas injection, corium cooling is only effective for low concrete concentrations in the corium (less than 15% by mass); for a power reactor, these conditions would correspond to water arriving soon after the corium had flowed into the reactor pit during a core melt accident. The tests also demonstrated the effects of gas injection into the corium; indeed, the programme's last tests performed with counter-current gas flow in the corium showed more effective corium cooling for lower concrete concentrations. This was the case in the SSWICS-11 test performed with 15% concrete, which led to high heat-flux removal, similar to that obtained without gas injection for a low (4%) concrete concentration (Figure 5.48). Gas

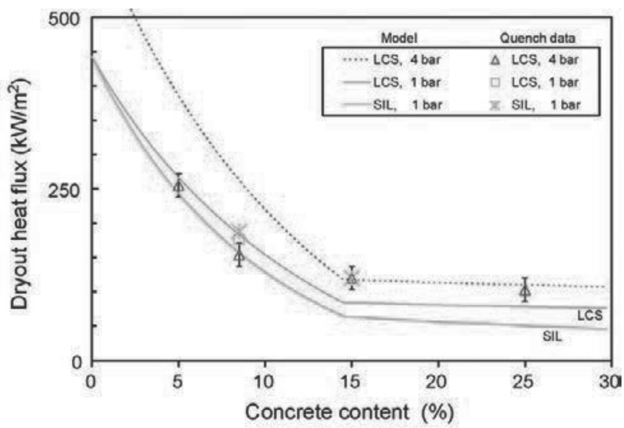

Figure 5.48. Measurement of heat fluxes removed by water ingression during SSWICS tests, as a function of concrete concentration in the corium (without gas injection into the corium), see [31].

Figure 5.49. Appearance of the resolidified lump of corium at the end of the SSWICS-11 test, performed with gas injection into the corium, see [32]; reproduced by permission of OECD.

flow could lead to the opening of pores in the corium during solidification, which would facilitate its cooling. However, it should be noted that, due to the lack of heating, SSWICS tests did not reproduce effects associated with residual heat in the pool and crust, effects which could be significant for a power reactor;

– measurements of mechanical stresses leading to crust failure (see [33]), in particular, the *in situ* measurements obtained during certain CCI tests show that crust failure occurs for low levels of stress. For a power reactor, it is very unlikely that the crust would remain a single block and bond to the reactor-pit walls.

During MCCI tests performed under more representative conditions with sustained corium heating (in particular during the CCI tests, see [29]), water ingression and corium ejection phenomena occurred simultaneously and were difficult to distinguish, as the corium ejections were close together.

5.4.2.2.4. PERCOLA programme

The PERCOLA experimental programme was performed by CEA between 1999 and 2002. Drawing lessons from the MACE tests and the results of calculations (see [34]) showing that it would be possible to cool a corium pool which would transform into a bed of debris, the purpose of this programme was to study corium ejection above a cracked crust caused by gases coming from concrete erosion. This analytical programme, performed with simulant materials (water, oil) brought to light several ejection regimes and meant that the influence of numerous parameters (see [30]) could be quantified, such as:

– fluid viscosity (a parameter representative of the increasing quantity of concrete in the corium as MCCI progresses);

– superficial gas velocity (a parameter representative of the type of concrete and the decreasing gas flowrate during MCCI);

– hole density in the crust (a parameter little understood for a power reactor);

– hole diameter (a parameter little understood for a power reactor);

– thicknesses of crust and bed of debris (parameters representative of the thickening of the crust and the bed of debris during MCCI after ejection).

The results of the PERCOLA programme have enabled development of an analytical model covering corium ejection during MCCI (this model is described in the next section).

5.4.2.3. Modelling

The main modelling work has covered corium ejection via holes in the upper crust and water ingression into corium.

An analytical model taking into account corium ejection[9] was developed in the context of the PERCOLA programme in 2004, see [35]. It takes into account the effect of

9. This model provides an estimate of the corium entrainment rate, i.e. the ratio of the volumetric flowrate of the liquid ejected and the volumetric flowrate of gases released during MCCI.

the major physical parameters for MCCI (superficial gas velocity, pool viscosity etc.) and the geometry of the ejection holes which were not covered in the Ricou and Spalding model (see [36]), which was used prior to the PERCOLA programme and describes liquid entrainment by a turbulent gas jet in a specific geometry. Application of the PERCOLA model to power-reactor scenarios tends to show that a bed of debris could quickly form if corium ejection is effective, see [37]. The stability of this bed of debris would then depend on the size of the debris particles formed. The PERCOLA model has been validated using the results of PERCOLA tests, but requires validation on the basis of more representative tests (with concrete erosion and prototypic compositions). To this end, large scale tests have been performed since 2012 at the Argonne National Laboratory (ANL in the USA); these are described in Section 5.4.2.4. Some of the model's input parameters are subject to very large uncertainties, such as the density and size of holes in the crust through which corium can pass. They are the subject of a specific model proposed by Farmer, see [38]. However, there is no experimental data that is sufficiently representative to validate these models for a power reactor.

In order to ensure the validity of the PERCOLA model in the long-term cooling phase for a power reactor, the model should also be supplemented to cover development of the bed of debris, in particular the effect of its thickening on corium ejection[10].

With regard to water ingression, it should be noted that a critical heat flux correlation, deduced from a model of crust cracking during water ingression, has been developed as part of the SSWICS programme, see [31]. This correlation has been adjusted using the results of tests performed without gas injection during corium solidification. On the basis of this correlation, it would appear that water ingression is much less effective for corium cooling than corium ejection.

As the possibility of cooling corium under water is strongly associated with MCCI, the modelling of cooling under water is covered by the same software as used for modelling MCCI. For example, the TOLBIAC-ICB code (see [39]) contains the corium ejection model developed after the PERCOLA test programme. Similarly, most models developed in the context of studies on the possibility of cooling corium by directing water onto it have been implemented in the CORQUENCH code (see [40]) developed by ANL to simulate 1D concrete erosion and the coupling between MCCI and heat transfer phenomena in the presence of water at the surface of the corium pool. Simplified models concerning water ingression into the upper crust and corium ejection, drawn from the first version of the CORQUENCH code, have been integrated into the MEDICIS code developed by IRSN, see [41]. More detailed models have subsequently been developed for the MEDICIS code, based on the PERCOLA model for hydrodynamics, see [35], and the literature available for assessing the geometry (density and diameter) of holes through which corium can be ejected, see [38] and [47]. Applications to a power reactor show that corium ejection is the dominant mechanism for corium cooling, and can significantly slow concrete erosion, especially in the case of a siliceous concrete, without stopping it completely, see [38] and [43].

10. The PERCOLA model assumes that gases and corium escape along vertical channels (or "chimneys") which develop in the bed of debris regardless of its thickness.

5.4.2.4. Summary and outlook

As shown by the overview above, in 2015, it is not possible to draw conclusions on the possibility of stabilising and cooling a corium pool during an MCCI by directing water onto the surface on the basis of the results of the tests performed (1D and 2D integral tests, corium ejection tests, and water ingression tests).

Progress in this area is hindered by the technological difficulties surrounding the performance of sufficiently large-scale tests with real materials (scale effects on the corium pool, crust bonding onto test set-up walls, representativeness of the corium heating mechanism etc.).

Given the results obtained and in the face of the difficulties encountered, other specific provisions aiming to cool corium were proposed and studied over the years 1995 to 2010.

Three very different types of corium cooling device have been considered:

- the first type is a corium spreader, that collects all the corium leaving the vessel and spreads it out on a large-surface "spreading compartment" to reduce the heat to be removed per unit surface area and to cool it using a passive water circulation system, as planned for the EPR, see [42]; this system has been studied in depth and is described in detail in Section 5.4.3;
- the second type of device is a core catcher in the form of a crucible, see [44], made up of a large cavity covered with a thick layer of "sacrificial" refractory materials (materials that are eroded by the corium), which reduces heat flux *via* corium "dilution" (due to the addition of the sacrificial materials) and cools the corium using a passive water circulation system outside the core catcher; an example of such a device is the one implemented in the VVER reactor on the Tianwan Nuclear Power Plant in China;
- a third type of device, based on corium cooling by directing water at the bottom, has been successfully tested on the COMET facility at Forschungszentrum Karlsruhe in Germany, using simulant materials, see [45], and also in Cadarache with materials more representative of corium that could form during a core melt accident on a power reactor, see [46]. In the tested device, corium is collected in a porous concrete core catcher, covered with sacrificial concrete. Once this layer has eroded, the corium is reflooded by a passive system that directs water *via* the porous concrete and fragments the corium; corium spreading is not necessary for cooling and such a device can be installed in the reactor pit, just under the reactor vessel.

Experiments performed on these types of devices, in particular those performed on the core catcher and spreader system described in Section 5.4.3, show that, during a core melt accident on a power reactor, the devices should be able to effectively cool the corium after reactor-vessel failure and prevent the basemat penetration which could result from MCCI. Such devices are implemented in some new generation reactors; in particular, this is the case for EPRs, which are fitted with a core catcher and spreader system.

At least in the short term, it is not planned that such corium-cooling devices be installed on second-generation reactors in the operating fleet, because such installation would involve expensive, complex modifications. Furthermore, the significant worker exposure to ionising radiation that would result from such installation work should be taken into account. For this reason, studies are also being pursued with regard to cooling corium by directing water onto it, especially by using existing spray systems in the containment building. In particular, a new large-scale test programme was launched at ANL in 2011, dedicated to the study of corium cooling during MCCI by directing water from above under representative conditions, especially with regard to changes in residual heat in the pool during its flooding with water; this programme is part of collaboration between EDF, IRSN and the US Nuclear Regulatory Commission (NRC). In the case where corium cooling by existing devices that can direct water from above is found to be inadequate, recourse to other corium cooling devices and, in particular, those studied in the COMET facility could be considered, including for second-generation reactors.

5.4.3. Corium spreading for the EPR

5.4.3.1. Physical phenomena involved

Development of a core catcher with a corium spreader for the EPR required a European R&D programme. The purpose of the spreading is to prevent reactor-building basemat penetration by facilitating corium cooling. To achieve this, spreading aims to ensure a sufficiently thin layer of corium, which minimises the surface heat flux due to residual heat to be removed.

Studies on corium spreading have therefore been performed to understand the ability of corium to spread on a substrate of fixed geometry and composition, with the corium flow conditions on the spreading surface determined by the accident sequence. The key parameters for corium spreading are the compositions of corium and substrate, the initial temperature and flowrate of the corium, and the geometry of the spreading compartment. References [60] and [76] give a summary of the work performed on this subject.

5.4.3.2. Description of the EPR core catcher

The concept used is based on spreading the corium over a large surface area, with the corium flooded and cooled by water from the In-containment Refuelling Water Storage Tank (IRWST) located in the containment building (see Figure 5.50 and Section 2.3.2.4 for a description of the EPR's engineered safeguard systems).

To promote corium spreading, the EPR core catcher temporarily retains the corium in the reactor pit before spreading. During this phase, the corium erodes a layer of "sacrificial" concrete, which is approximately 50 cm thick, before flowing into the melt discharge channel that connects the reactor pit to the "spreading compartment". This layer of sacrificial concrete is laid on a protective 10-to-14-cm-thick zirconium layer, which aims to ensure the integrity of reactor-pit concrete structures, even in the event

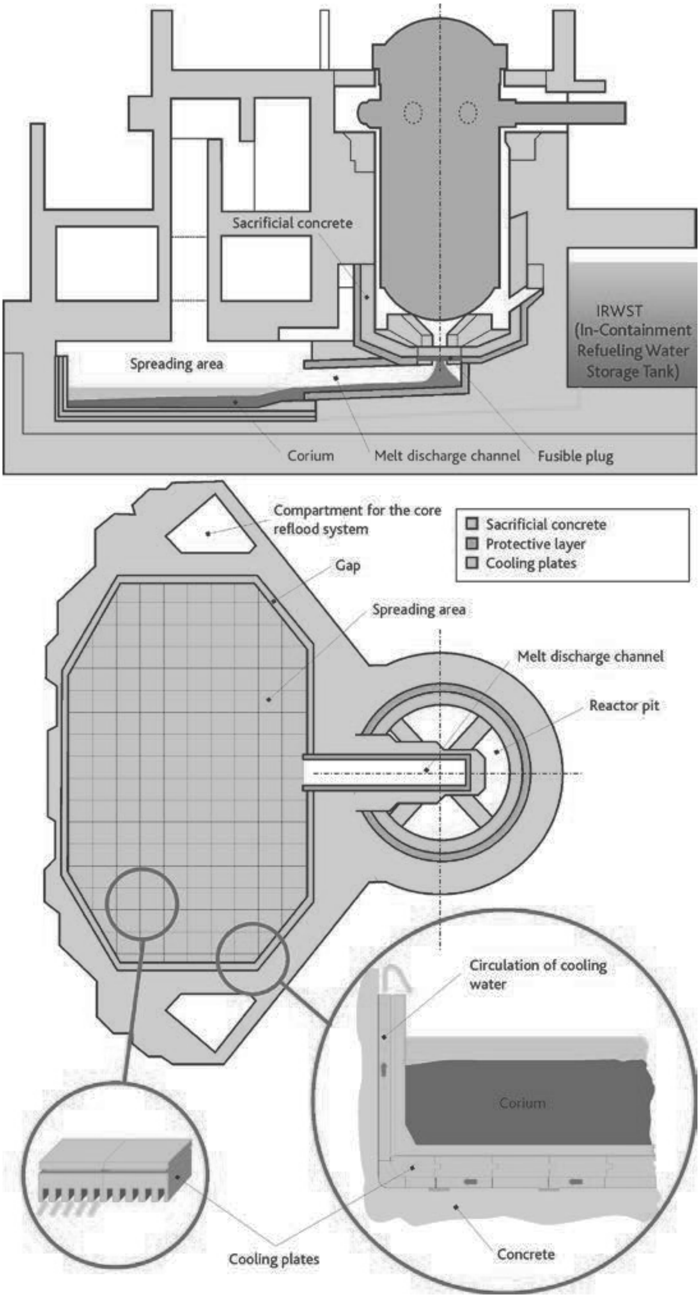

Figure 5.50. The upper part of the figure represents a cross-section view of the main components of the EPR core catcher (from [78]). The lower part of the figure represents the cooling system for the EPR core catcher with details: lower right, a vertical cross-section of the walls of the spreading compartment showing how the cooling water circulates under these walls; lower left, a vertical cross-section of the horizontal cooling channels located under the spreading compartment. The central part shows a top view of reactor pit, melt discharge channel and spreading compartment.

of non-uniform erosion of the sacrificial concrete by the corium (see Figure 5.50). During this temporary retention, the physico-chemical properties of the corium are modified (becoming more fluid, uniform in composition and of low viscosity) to facilitate its flow in the melt discharge channel and the spreading phase itself. If there are several successive corium flows following reactor-vessel failure, this temporary retention also means that the corium from the various flows can be gathered to obtain a uniform corium and a single flow towards the spreading compartment. In the zirconium layer under the sacrificial concrete at the bottom of the reactor pit, there is a wire-mesh insert which acts as a melt plug. This "gate" gives access to the melt discharge channel; it is a by-design weak point, as it is the only place where the sacrificial concrete is not reinforced by a protective layer, and it therefore fails relatively quickly on contact with the corium (after erosion of the sacrificial concrete), providing a sufficiently wide flow cross-section for rapid flow of all the corium into the spreading compartment.

The spreading compartment has a surface area of approximately 170 m^2. The floor and sidewalls of this compartment are assembled from a large number of individual elements made of cast iron. This structure is largely insensitive to thermal expansion and steep temperature gradients. The floor elements have rectangular, horizontal cooling channels. The inside of the spreading compartment is covered with a layer of sacrificial concrete. The arrival of corium triggers the opening of valves that initiate gravity-driven flow of water from the IRWST into the spreading compartment. The water first fills the horizontal cooling channels below the spreading compartment, and then fills the space behind the side-wall cooling structure before reflooding the corium from above. The system is shown in Figure 5.50.

5.4.3.3. Physics of corium spreading

Corium spreading is governed by competition between hydrodynamic driving forces (hydrostatic pressure and, to some extent, inertia), which promote progress and thinning of the flow, and gradual corium solidification, which leads to increasing apparent viscosity and the appearance of crusts in contact with the substrate and surface.

The hydrodynamics of lava spread has been studied by several authors in the field of volcanology, see [48], [49] and [50]. Numerical models and semi-analytical solutions have been developed for the flow of a fluid whose properties remain constant during the flow. Spreading on a horizontal surface is a free-surface flow, whose driving force is a function of the downslope. Corium flow during a core melt accident depends on gravity, inertia (at high flowrates) and viscous friction forces (at lower flowrates).

Corium rheology, see [73] and [74], changes strongly during its cooling, in particular below the liquidus temperature when crystalline phases appear. It depends on both the viscosity of the liquid phase (a mixture in which the silicate ions from the sacrificial concrete increases the viscosity by forming networks), which has been described by Urbain in [51] for example, and on the effect of crystals which solidify during the flow (the type of complex fluid formed, called semi-solid, is described by Flemmings in [52] and an empirical viscosity formula has been proposed for corium, see [73] and [77]).

Corium cooling is due to radiative heat transfer from the surface of the flow and by convection in contact with the substrate. Crusts may form at these two interfaces and contribute to slowing the flow. Nevertheless, there is significant thermal contact resistance, of around 5.10^{-3} m^2.K/W, at the corium-substrate interface, which contributes to reducing corium cooling in contact with the substrate, see [53]. The effect of residual heat is small, given the short duration of spreading (no more than a few minutes).

In reference [54], Griffiths and Fink have published a detailed study of the various models of the spreading of solidifying lavas as a function of dominant forces (gravity and inertia, gravity and viscosity, gravity and complex rheology, gravity and crust strength etc.). These models mean that the speed of corium spreading can be assessed as a function of its flowrate in the melt discharge channel and its viscosity. They are used to assess the validity of corium spreading calculations performed for simplified boundary conditions, that do not take corium cooling into account.

5.4.3.4. Experimental programmes, modelling and simulation software

5.4.3.4.1. Experimental programmes

The first test programmes regarding spreading corium from a core melt accident on a power reactor were performed at Brookhaven in the USA, see [55]. Their purpose was to study corium spreading on the bottom of a reactor pit on a Mark I BWR. In Europe, experimental and numerical studies of spreading have been performed with a view to development of a core catcher for the EPR. Most of this work has been performed in the context of European projects: COMAS, large-scale corium cooling tests performed by AREVA, see [57]; Corium Spreading and Coolability (CSC), qualification tests for the concept of a core catcher with corium spreading and of the COMET concept of reflooding from below, see [56]; and Ex-vessel COre melt STAbilisation Research (ECOSTAR), tests pertaining to the study of the physico-chemical phenomena that occur during spreading and the effectiveness of reflooding spread corium by directing water onto the top or bottom, see [58].

Experimental programmes include analytical experiments that aim to study the effect of various physical phenomena involved in corium spreading and cooling (for example, the CORINE programme using simulant materials performed at CEA Grenoble and jointly funded by IRSN, see [59] and [60]), semi-analytical experiments with simulant materials and tests with prototypic materials[11]. Tables 5.6 and 5.7 present the characteristics of main test programmes, whether with simulant materials or with prototypic corium compositions. As an example, Figure 5.51 illustrates a corium spreading test.

These experimental programmes (in particular, the CORINE, VULCANO and KATS programmes) cover the greater part of the range of possible variations of the parameters accessible to experimentation with regard to the geometry, properties of materials and boundary conditions.

11. (Non-radioactive) corium of identical chemical composition to that expected during a core melt accident, but of different isotopic composition (for example using depleted or natural uranium instead of enriched uranium).

Table 5.6. Experimental spreading programmes performed with simulant materials.

Programme	Laboratory	Materials	Scale (volume poured)	Geometry	Parameters or effects studied
CORINE [59, 60]	CEA (France)	Low-temperature simulant materials (water, glycerol, low-melting point metal alloys)	~ 50 litres	19° angular sector	• Flowrate (from 0.5 to 3 L/s) • Effect of material (viscosity, single substances or non-eutectic mixtures). • Cooling from above or below. • Effect of a gas flow coming from the substrate.
Greene [55]	BNL (USA)	Lead	~ 1 litre	Square cross-section	• Spread mass. • Heating. • Effect of water depth.
S3E [61]	KTH (Sweden)	Low- and intermediate-temperature (1200 °C) simulant materials	5 to 20 litres	Rectangular channels	• Flowrate. • Heating. • Effect of material. • Effect of (concrete) substrate. • Effect of water, with or without boiling.
SPREAD [62]	Hitachi Energy Research Laboratory (Japan)	Steel	1 to 15 litres	Rectangular channel Half-disk	• Spread mass. • Heating. • Flowrate. • Effect of inlet geometry. • Effect of substrate. • Effect of water depth.
KATS [63-65]	FzK (Germany)	Aluminium thermite (Al_2O_3 + Fe) around 2000 °C	Up to 850 litres	Rectangular channel 90° angular sector	• Spread mass. • Flowrate. • Effect of substrate. • Effect of adding "sacrificial materials" • Type of phase(s) spread (oxide or metal). • Reflooding.

Table 5.7. Experimental programmes performed with prototypic materials.

Programme	Laboratory	Materials	Scale (volume poured)	Geometry	Parameters or effects studied
COMAS [57]	Siempelkamp (Germany)	Corium-concrete-iron mixtures Liquidus temperature around 1900 °C	20 to 300 litres	Rectangular channels 45° angular sector	• High flowrates (> 150 kg/s). • Effect of silica. • Effects of substrate (ceramic, metal or concrete).

Programme	Laboratory	Materials	Scale (volume poured)	Geometry	Parameters or effects studied
FARO [66]	CCR Ispra (European Commis-sion)	$UO_2 + ZrO_2$ Liquidus temperature around 2700 °C	~ 20 litres	19° angular sector	• Presence or otherwise of a thin layer of water. • Effect of a metal sub-strate.
VULCANO [67]	CEA (France)	$UO_2 + ZrO_2$ + concrete erosion products Liquidus temperature of 1900 to 2700 °C	3 to 10 litres	19° angular sector	• Flowrate. • Corium composition. • Effects of the substrate.

The spreading experiments performed show that, for corium flows during solidification, the liquid and solid phases remain mixed (there is no macrosegregation, unlike that which occurs during slower transitions). The solid fraction varies continually during flow. Furthermore, for a corium where the difference between solidus and liquidus temperatures is large, a “skin” forms in a mushy (liquid-solid) state rather than as a solid crust, at least initially. Conversely, in the case of a more refractory corium where the solidus and liquidus temperatures are close, a solid crust forms on the upper flow surface, which cracks and lets molten corium pass. In this case, the phenomena observed depend strongly on the scale of the flow, which means that the available experimental data remains inadequate on this specific point of the effect of the crust on flow dynamics, as it only involves small-scale tests with masses at least 1000 times smaller than those which would be involved in the case of a power reactor. Erosion of the concrete substrate during spreading remains minor; an effect on the spreading speed has been brought to light but is of little importance.

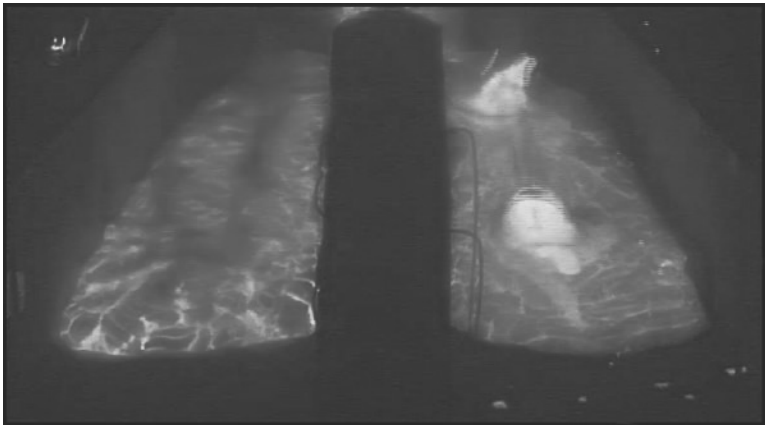

Figure 5.51. Spreading of corium representative of that produced during a core melt accident on a power reactor onto ceramic (left) and concrete (right) substrates, from the VULCANO VE-U7 spreading test performed by CEA with a mixture of UO_2, ZrO_2 and concrete erosion products, see [58]; credit: CEA.

Knowledge of dry corium spreading for the case of a large difference between solidus and liquidus temperatures (i.e. for corium rich in non-refractory materials from sacrificial concrete) is sufficient to validate calculation software and extrapolations to the case of a power reactor.

With regard to corium spreading under water, CORINE tests, performed with simulant materials and a water depth of around 10 cm, show that there could be an accumulation of corium whose thickness could reach that of the depth of water downstream of the corium flow, but that this would have little effect on the spreading. However, extrapolation of these results concerning spreading under water, which depend on scale (in particular the flow height), to the case of a power reactor is not possible with existing knowledge.

5.4.3.4.2. Models and simulation software

Several simulation software packages have been developed in Europe to model corium spread. Table 5.8 presents their main characteristics. This software has been the subject of significant validation work using the results of flow tests performed using simulant or prototypic materials. For example, comparison exercises for the calculation results produced using different software systems on the basis of the VULCANO VE-U7 test (with prototypic corium compositions, see [72]) and calculations performed on the basis of the ECOKATS-1 test (with simulant materials, see [64]) have shown that the software produces a good estimate of the spreading surfaces observed during tests; the uncertainty is around 20%.

The Stockholm Royal Institute of Technology (KTH) has developed a simplified analytical model for corium spreading, which has been satisfactorily validated (the mean precision is around ± 50% on the spreading surface calculations), see [61] and [76].

To supplement the modelling of corium spreading, an R&D programme was performed by CEA at the end of the 1990s, dedicated to the study of corium rheology during its solidification, taking into account the variation of corium viscosity as a function of the corium spreading speeds along the vertical axis, see [73]. This means that viscosity models applicable to liquids with or without silica can be applied to corium, and therefore the viscosity of semi-solid corium as a function of the solid volume fraction can be predicted with adequate accuracy (i.e. within a factor of 3), see [73], [74] and [75]. These calculation results for the viscosity of corium are used in the spreading calculations.

All this work has led to a level of corium-spreading modelling that is sufficient to predict the spreading kinetics under the conditions of a core melt accident on a power reactor. In particular, the software developed is used to check, with reasonable uncertainty, the correct spreading of corium as a function of the boundary conditions of various accident scenarios, especially as a function of corium temperature and changes in corium flowrate.

Table 5.8. The main simulation software for corium spreading.

Code	Origin	Geometry	Characteristics	Validation
MELTSPREAD [68]	ANL for EPRI (USA)	1D	• Covers substrate erosion and corium oxidation.	• Mainly based on the results of Greene's tests.
THEMA [60]	CEA (France)	2D temperature and horizontal speed averaged over the vertical axis	• Covers corium solidification (in mass and in crusts) and substrate erosion. • 3D resolution of heat equations in the substrate.	• Analytical tests. • Tests with simulant materials and prototypic corium compositions.
LAVA [69]	GRS (Germany)	2D temperature and horizontal speed averaged over the vertical axis	• Detailed analysis of corium cooling and rheology.	*Idem*
CROCO [70]	IRSN (France)	2D horizontal and vertical	• Detailed modelling of convection in the flow. • Calculation of the free surface using Lagrangian modelling and resolution of conservation equations on a Eulerian mesh.	*Idem*
CORFLOW [71]	FzK (Germany)	3D	• Detailed modelling of convection in the flow. • Free surface represented by a "corium height" function deduced from the equations for conservation of mass and momentum.	*Idem*

5.4.3.5. Summary and outlook

R&D programmes performed to study corium spreading have established that dry spreading of the corium formed during a core melt accident on a power reactor enables its later cooling (the corium layer produced is sufficiently thin). It has been found, in particular during VULCANO tests, that even when the temperature of a corium-concrete mixture is 100 to 200 °C below the liquidus temperature, this mixture spreads adequately, as long as the flowrate is sufficiently high.

The presence of a thin water layer (simulating the water which would condense in the reactor building during an accident) or that of a concrete substrate (releasing steam and CO_2 during its interaction with the corium) have little effect on spreading. However, the influence of a deeper layer of water on corium spreading cannot be determined on the basis of existing knowledge. In this case, corium flow depends on the mechanical behaviour of the crusts formed on the surface and at the front of the flow (in particular their cracking), and on fragmentation of the corium; study of the behaviour of crust and corium would require additional tests to provide a validated model.

Due to uncertainties concerning the ability of a layer of corium to spread under water, design provisions were taken for the EPR, aiming to ensure collection of corium from the reactor vessel in the reactor pit, followed by its dry spreading (no water in the spreading compartment before the corium arrives) then its cooling by water circulating in cooling channels located under the spreading compartment and finally its cooling by reflooding from above.

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