INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Research and development with regard to severe accidents in pressurised water reactors:

Summary and outlook

Rapport IRSN-2007/83

Rapport CEA-2007/351

FOREWORD

This report was produced conjointly by the French Institute for Radiological Protection and Nuclear Safety (IRSN) and the French Atomic Energy Commission (CEA). The French electric utility EDF also contributed to Section 8.2.

The following individuals participated in writing or editing all or part of the report:

For the IRSN:

Ahmed Bentaïb, Hervé Bonneville, Cataldo Caroli, Bernard Chaumont, Bernard Clément, Michel Cranga, Florian Fichot, Joëlle Fleurot, Richard Gonzalez, Vincent Koundy, Bruno Laurent, Jean-Claude Micaelli, Renaud Meignen, Frédérique Pichereau, Delphine Plassart, Emmanuel Raimond, Anne Tenaud and Jean-Pierre Van-Dorsselaere.

For the CEA:

Gérard Ducros, Michel Durin, Christophe Journeau, Daniel Magallon, Jean-marie Seiler, Etienne Studer and Bruno Tourniaire.

For EDF:

William Ranval and EDF's Severe Accident Teams.

CONTENTS

CHAPTER 1 : INTRODUCTION

1.1 SEVERE ACCIDENTS — DEFINITION AND RESEARCH OBJECTIVES

In this report, the term "severe accident" refers to an event causing significant damage to reactor fuel and resulting from more or less complete core meltdown. Such accidents are highly unlikely in light of the preventive measures implemented by operators. However, they are the focus of considerable research because the release of radioactive products into the environment would have serious consequences. This research also reflects a commitment to the defence-in-depth approach.

Severe accidents are generally caused by a cooling failure within the core, which prevents proper evacuation of residual power. Within a matter of hours, fuel element structure is damaged as the result of multiple dysfunctions, arising from equipment and/or human error, including the failure of safety procedures. A series of complex phenomena then occur, according to various scenarios and depending on the initial conditions of the accident and operator actions. These scenarios might ultimately lead to a loss of containment integrity and to the risk of large radioactive releases outside the containment building.

For the purposes of this document, "early releases" are those liable to occur before all the measures aimed at protecting the general public can be implemented.

The physical phenomena involved in severe accidents are extremely complex; they generally demand the development of specific research.

The research in this area thus aims to further understanding of these physical phenomena and reduce the uncertainties surrounding their quantification, with the ultimate goal of developing models that can be applied to reactors. These models, grouped together in computer codes, should allow predicting severe accident progression.

In this field, there is no way to conduct experiments on a real-world scale and reproduce all the possible situations. That is why elementary tests must be used, allowing each physical phenomenon to be studied separately, followed by global tests confirming the interactions between phenomena. All testing must be done on a scale within the technical and financial capacities of the facilities used, while remaining sufficiently representative to allow extrapolation to the reactor scale.

This research addresses reactors currently in operation as well as future reactors. The underlying phenomena are the same for existing pressurised-water reactors and those in the planning phases. However, for existing plants, severe accidents were not a design consideration. Consequently, modifications to these facilities are limited and the research in this area is primarily aimed at limiting the potential impact of severe accidents. Specifically, there are two complementary research orientations: 1) characterising releases and studying modes of containment failure, and 2) developing methods to deal with the consequences.

For the future EPR (European Pressurised-water Reactor), the French Nuclear Safety Authority has set the objective of significantly reducing the risk of radioactive releases for all conceivable accident situations, including core meltdown accidents. Specific design measures must be implemented in order to practically eliminate accidents with the potential for large early releases, and to reduce the impact of accidents involving low-pressure core meltdown. Research conducted within this framework should allow these objectives to be reached.

1.2 RESEACH FRAMEWORK

Most countries with well-developed nuclear power sectors, such as the United States, Japan, Germany, Belgium, Canada, South Korea, Switzerland, Sweden and Russia, also have severe accident research programmes. In general, each country has focused on one or more particular aspects of the issue; the field is too vast to allow investigation of all phenomena by any one national programme.

In France, the first major research programmes addressing severe accidents were initiated in the early 1980s, following the accident at the Three Mile Island Unit 2 reactor in 1979. Due to the size of its nuclear fleet, France undertook to develop its own programmes in nearly all areas related to severe accidents. Safety research in France is primarily conducted by the IRSN, the CEA, EDF and AREVA. All these entities develop or help develop simulation codes, and the IRSN, the CEA and AREVA (in Germany) all have testing facilities. Certain actions are also outsourced to research centres or component manufacturers.

Research in the area of severe accidents involves very substantial human and financial resources as well as collaboration between nuclear stakeholders, industry groups, research centres and safety authorities, at both the national and international levels. In France, the IRSN, the CEA, EDF and AREVA have initiated joint programmes on numerous subjects (involving bipartite, tripartite or even quadripartite agreements) and participate in international programmes, especially those initiatives supported by the European Commission through its Framework Programmes for Research and Development (FPRD), or those conducted under the auspices of the OECD (Organisation for Economic Co-operation and Development).

As part of the Sixth Framework Programme, a Network of Excellence called SARNET (Severe Accident Research Network of Excellence) and coordinated by the IRSN, was set up to optimise the use of resources in the area of severe accident research. SARNET brings together 49 participants from 19 different countries, either members of the European Union or new candidates for membership, as well as Switzerland. It aims to improve scientific knowledge, to define and oversee research programmes, to ensure the sustainability of outcomes and to disseminate information. Two of its “integration” activities are the ASTEC code (see Chapter 8) and methodologies for Level 2 Probabilistic Safety Assessments (PSA-2).

Many international collaborations have also been set up in the framework of the OECD. Through the Committee on the Safety of Nuclear Installations (CSNI), the OECD coordinates the launch and oversight of research programmes with the goal of building consensus on scientific and technical questions of general interest. Subjects are most often selected based on the general recommendations of expert reports on issues which have yet to be fully resolved, and on programmes or facilities which will involve international collaboration. In particular, the report [1.2_1] can be cited in reference; it is currently being updated. Since the OECD lacks its own budget for this type of action, it relies on contributions from participants. In the area of simulation tools, the CSNI facilitates the creation of expert workgroups to establish validation matrices. It also oversees ISP (International Standard Problems [1.2_2]) which compare experimental results for a given problem with the outcomes of various teams using various computer codes. Finally, State-of-the-Art Reports are produced on subjects of general interest (e.g. hydrogen distribution, hydrogen combustion, aerosol behaviour). These SOARs provide a complete vision of a given problem by reviewing current knowledge and the remaining uncertainties, and by presenting recommended orientations.

1.3 REPORT OBJECTIVES

This document reviews the current state of research on severe accidents in France and other countries. It aims to provide an objective vision, and one that's as exhaustive as possible, for this innovative field of research. It will help in identifying R&D requirements and categorising them hierarchically. Obviously, the resulting prioritisation must be completed by a rigorous examination of needs in terms of safety analyses for various risks and physical phenomena, especially in relation to Level 2 Probabilistic Safety Assessments. PSA-2 should be sufficiently advanced so as not to obscure physical phenomena that, if not properly understood, might result in substantial uncertainty. It should be noted that neither the safety analyses nor PSA-2 are presented in this document.

This report describes the physical phenomena liable to occur during a severe accident, in the reactor vessel and the containment. It presents accident sequences and methods for limiting impact. The corresponding scenarios are detailed in Chapter 2. Chapter 3 deals with in-vessel accident progression, examining core degradation (3.1), corium behaviour in the lower head (3.2), vessel rupture (3.3) and high-pressure core meltdown (3.4). Chapter 4 focuses on phenomena liable to induce early containment failure, namely direct containment heating (4.1), hydrogen risk (4.2) and steam explosions (4.3). The phenomenon that could lead to a late containment failure, namely molten core-concrete interaction, is discussed in Chapter 5. Chapter 6 focuses on problems related to in-vessel and ex-vessel corium retention and cooling, namely in-vessel retention by flooding the primary circuit or the reactor pit (6.1), cooling of the corium under water during the corium-concrete interaction (6.2), corium spreading (6.3) and ex-vessel core catchers (6.4). Chapter 7 relates to the release and transport of fission products (FP), addressing the themes of in-vessel FP release (7.1) and ex-vessel FP release (7.3), FP transport in the primary and the secondary circuit (7.2), aerosol behaviour in the containment (7.4) and FP chemistry (7.5). Finally, Chapter 8 presents a review of development and validation efforts for the main severe accident codes: ASTEC, MAAP and MELCOR.

In Chapters 3—7, for each of the theme areas, the phenomena involved are reviewed. The major relevant experiments are then briefly described, including recent, ongoing and future projects. The key models and specific codes (except for integral codes) used to simulate the phenomena in question are also discussed. Finally, the state of current knowledge is reviewed and an outlook for the future is presented, especially regarding experimental programmes and the development of modelling tools.

CHAPTER 1 REFERENCES

[1.2_1] Nuclear Safety Research in OECD Countries, Major Facilities and Programmes at Risk NEA/2001

[1.2_2] CSNI International Standard Problems, Brief description (1975-1999) NEA/CSNI/R(2000)5

CHAPTER 2 : SCENARIOS

This chapter provides an introduction to accident scenarios involving reactor core meltdown in a PWR.

The first section describes the main classes of Level 1 PSA accidents, initiated by internal events and liable to provoke substantial loss of coolant and core meltdown as a result. Accident scenarios resulting from an external event are not described.

The second section provides a general overview of the various physical phenomena liable to occur in a PWR once core meltdown is initiated. It also describes their consequences in terms of loss of containment for fission products.

Level 2 PSA conclusions (available only for the 900-MWe series) have been purposely excluded from this chapter: the IRSN esteems that the results and the comparative analyses of studies conducted independently either by its teams or by EDF have not been sufficiently consolidated. The next update of the PWR 900 PSA-2 (post-VD3 version) is planned for end-2006 and will either be carried out by EDF or the IRSN.

2.1 ACCIDENT TYPES POTENTIALLY LEADING TO CORE MELTDOWN

2.1.1 LEVEL 1 PSA ACCIDENT CLASSES

2.1.1.1 Introduction

Level 1 PSAs aim to identify accident sequences leading to partial or total fuel damage. It should be noted that EDF develops Level 1 PSAs for the 900-MWe, 1300-MWe and N4 series, and for the EPR system. As for the IRSN, it develops Level 1 PSAs for the 900-MWe and 1300-MWe series. EDF studies are used as benchmarks in terms of safety analysis; IRSN studies are used as counterstudies to analyse EDF's results and conclusions.

The rest of this chapter is based on the Level 1 PSAs developed by EDF and the IRSN for the 900-MWe series, in preparation for the third 10-year inspection of the reactors in this series (references [2.1_1] and [2.1_2]). Only internal initiators are considered.

The systems involved in normal and accident situations are presented schematically in Figure 2.1-1.

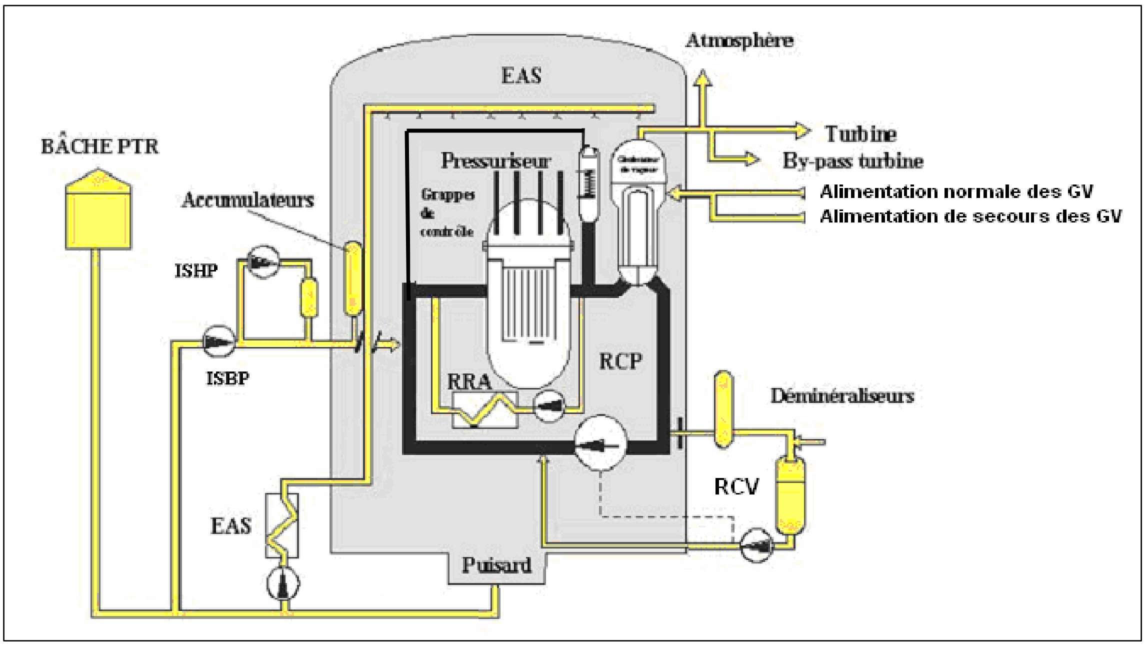


Figure 2.1-1: Schematic diagram of various reactor systems

2.1.1.2 Loss of coolant accidents (LOCA sequences, divided into Large Breaks, Intermediate Breaks and Small Breaks)

These accidents are initiated by a break in the reactor coolant system (RCS) or one of the connecting circuits (except for vessel rupture, or ruptures of one or several steam generator tubes, see Section 2.1.1.5). Such a break would cause loss of RCS coolant and depressurisation of the reactor coolant system. The Level 1 PSA conducted by EDF distinguishes between several situations depending on the reactor's initial state, and the break's location and size.

In the event of a LOCA, reactor coolant system depressurisation automatically causes the reactor to shut down and the safety injection system to actuate. For large breaks, rapidly increasing pressure in the containment automatically actuates the spray system.

To bring the accident under control, the reactor's protection and safety systems must perform the following functions:

- Controlling reactivity by automatically shutting the reactor down and injecting borated water.
- Using the safety injection system to maintain the reactor vessel's water inventory, first in the direct injection phase (using the refuelling water storage tank), then in the recirculation phase (using the sumps in the containment building).
- Cooling and evacuating residual power, first via the break itself or the steam generators (depending on the break's size), then via the residual heat removal system (RHR) if it can be connected to the reactor coolant system; for safety injection involving sump recirculation, energy

is evacuated from the containment by heat exchangers integrated in its spray system, or by RHR for small RCS breaks.

Accident sequences causing core meltdown are the result of failure in one or more of these safety systems. For reactors at full power, mitigation sequences exist for the following situations:

- Failure of safety injection or of the containment heat removal system (CHRS) in the direct injection or recirculation phases.
- Failure of the control and monitoring system preventing the launch of safety injection, containment heat removal or recirculation (in the absence of corrective action by the operating team).

When reactors are shut down (RHR circuit connected), postulated accident sequences vary depending on the initial state of the reactor (RCS closed, partially open or completely open), but in all cases there is failure to maintain the water inventory due to human error(s) or equipment dysfunction(s).

2.1.1.3 Loss of coolant accidents occurring outside the containment building (V-LOCA class)

These accidents, due to loss of coolant via a break occurring outside the containment building, but in a circuit connected to the RCS loop rather than independent of it, are postulated to have two particular characteristics:

- Since coolant loss occurs outside the containment building, the recirculation phase of safety injection might prove impossible.
- In the event of a core meltdown, fission products would be released outside the containment building if there was no way to isolate the break.

A ruptured thermal barrier in one of the RCS pumps, for example, might lead to this sort of situation.

2.1.1.4 Accidents involving secondary system line breaks (includes FWLBs and SLBs)

The initiating events considered for this class include:

- A break in steam generator feedwater piping (FWLB).
- A steam line break (SLB) in the secondary system of a generator.

A rupture in steam generator feedwater piping results in drainage of the generator's secondary system, accelerated RCS cooling, automatic reactor shutdown and initiation of safety injection.

Operating personnel must isolate the defective steam generator to prevent the drainage of the other steam generators and maintain the secondary system's cooling capacity.

A steam line rupture sharply increases flowrate to the secondary system and thus results in greater energy extraction from the reactor coolant system, whose pressure and temperature in turn begin to drop. The available anti-reactivity margin is reduced as a result of this drop in temperature. If the steam line rupture is substantial, safety injection is rapidly initiated (injection of borated water) and the generators are automatically isolated through action at the steam lines.

The following functions must be performed in the event of an SLB or FWLB:

- Reactivity control; initiated by the reactor's automatic shutdown and completed if necessary by safety injection of borated water.
- Evacuation of residual power; performed by the unaffected steam generators, supplied by the auxiliary feedwater system.

For an FWLB, the dominant accident sequences leading to core meltdown involve several assemblies getting stuck outside the core, preventing control of reactivity, or a failure of the feed-and-bleed function.

Regarding SLBs, two examples of sequences leading to core meltdown are as follows:

- *In the event of major SLBs*, CHRS failure may cause bypass of the instrumentation qualification profile required for initiating standby conditions; likewise, if at least two assemblies get stuck outside the core due to mechanical failure, it may be impossible to control reactivity.
- *In the event of minor SLBs,* mechanical blockage of an assembly, combined with failure by the operating personnel to isolate the faulty generator, may also hinder control of reactivity.

2.1.1.5 Accidents involving steam generator tube ruptures (SGTR class, combined SLB-SGTR incidents)

The initiating event for this class is a major leak or rupture in one or several steam generator tubes (SGTR class), or a secondary line break (feedwater or steam) leading to the quasi-immediate rupture of one or several steam generator tubes (combined SLB-SGTR).

Rupture or leaking in steam generator tube(s) causes a drop in pressure in the reactor coolant system, leading to automatic reactor shutdown, followed by actuation of the safety injection and auxiliary feedwater systems.

Operating personnel must then identify and isolate the defective steam generator, shut down safety injection and cool the RCS via the unaffected steam generators, in order to establish the operating conditions for the residual heat removal system. If there is a delay in shutting down safety injection or isolating the defective steam generator, this generator fills with water and secondary system discharge components are in turn solicited, which may cause them to become stuck in open position, leading to containment bypass. In this situation, the reactor coolant system must be depressurised to counteract the loss of RCS coolant.

Accident sequences postulated for a SGTR (one or two tubes) primarily involve complete loss of coolant by the secondary system and failure of the feed-and-bleed function.

2.1.1.6 Accidents involving complete loss of cooling source or associated systems (H1 class)

Initiating events for this class of accidents are loss of the terminal cooling source or of the cooling systems allowing evacuation of energy towards this source.

Failure of intermediary cooling systems results in:

- RCS pump shutdown
- Suspended cooling of RCS pump thermal barriers, potentially resulting in an RCS break should injection fail at the RCS pump seals
- Loss of cooling capacity for CVCS discharge water
- Loss of cooling capacity in certain ventilation systems, potentially resulting in equipment loss
- Loss of cooling capacity in the exchangers of the containment heat removal system and, consequently, loss of energy evacuation if safety injection involves recirculation

Operator intervention in this situation aims to bring the reactor to standby conditions (45 bar) so that injection at the RCS pump seals can be suspended without risk. RCS pumps are shut down and RCS cooling is performed by the secondary system via natural circulation on the RCS side.

The accident sequences postulated for a reactor at full power are primarily related to failure of the emergency feedwater system (EFWS) to supply the steam generators, followed by deficient feed and bleed, or failure to maintain water in the reactor coolant system in the event of an RCS pump seal break.

During reactor outages, the postulated sequences depend on the reactor's initial state: dysfunction in the EFWS or failure to maintain the RCS water inventory, if the reactor is closed or partially open; water supply dysfunction or U5 decompression and filtration system not used to cool the containment, if the reactor coolant system is initially open.

2.1.1.7 Accidents involving a complete loss of feedwater to steam generators (H2 or TGTA class)

Initiating events for this accident class involve equipment failure resulting in the combined loss of main and auxiliary feedwater systems for the steam generators.

If no mitigating action is taken by operating personnel, the steam generators drain rapidly on the secondary side and become inefficient. The reactor coolant system also overheats and its pressure rises until reaching the set pressure of the SEBIM valves within the pressuriser. The reactor coolant system is then emptied, via the SEBIM valves. Its pressure remains elevated until core uncovering is complete. Core meltdown may occur with the reactor coolant system at pressure, constituting a short-term containment hazard (ejection of corium into the containment during vessel rupture and subsequent DCH, risk of steam generator tube rupture).

In this situation, operating personnel must:

- Open the pressuriser safety valves, then initiate safety injection (feed-and-bleed function).
- Ultimately, restore feedwater to the steam generators so as to establish the operating conditions for the residual heat removal system.

The dominant accident sequences for this situation lead to core meltdown and result primarily from failure of the feed-and-bleed function, either in the short term (operator error, safety injection fails to initiate) or over the long term (operational failure of safety injection, CHRS fails to cool containment building).

2.1.1.8 Accidents involving a complete loss of electrical power (H3 class)

These accidents are initiated by the quasi-simultaneous loss of the two 6.6 kV emergency switchboards LHA and LHB, or the loss of external then internal power due to a succession of events damaging electrical power sources, which in turn cuts off all electrical power to the reactor's safety systems.

In this situation, if the reactor coolant system is initially closed, operating personnel must initiate a standby mode in which injection at the RCS pump seals is no longer necessary. This is accomplished using the LLS alternator turbine, the test pump (injection at RCS pump joints), the TDAFWP (turbine-driven auxiliary feedwater pump) and the MSRT dump valves. These conditions involve an RCS temperature of 190 °C and 45 bar of pressure.

If the reactor coolant system is initially partway open, operating personnel must establish an intermediary mode characterised by a temperature of 190°C, in which the test pump compensates for water loss via the reactor coolant system vents.

If the reactor coolant system is initially open, a gravity-based backup system must be implemented as a short-term measure, completed in the medium-term by water injection via the adjacent unit's charging pump.

In any event, the standby generator sets (SGS) must be connected very rapidly.

Postulated accident sequences include situations resulting from:

- Failure of the turbine-driven auxiliary feedwater pump, or of injection at the joints (resulting in a break due to lack of thermal barrier cooling), if the reactor is initially closed.
- Failure of emergency water supply to the reactor coolant system, if the RCS is open.

2.1.1.9 Loss of internal electrical power

This class includes accident sequences caused by a power outage at a low-voltage switchboard. The events considered include failures of an LB. or LC. switchboard, or failure of the LDA switchboard.

The impact on the plant has been assessed for each type of electrical power outage. Depending on the type of outage, the following may occur: partial unavailability of the steam generator feedwater function and of the maintenance of RCS pump seal integrity (thermal barrier cooling, injection at the seals).

Core meltdown is generally caused by H2 sequences (complete loss of feedwater to steam generators and failure of the feed-and-bleed function) and H1 sequences (break at RCS pump seals and failure to maintain water inventory).

The dominant sequences are the H2 type, caused by loss of the LCA switchboard.

2.1.1.10 Transients involving automatic shutdown failure (ATWS class)

This class of accidents encompasses various situations involving failure of automatic reactor shutdown following an initiating event within the facility. All the initiators described above may in fact lead to an ATWS accident.

These situations result in a loss of normal feedwater to steam generators, and the auxiliary feedwater system may not be sufficient to evacuate the nuclear power still present in the core.

Three consequences are postulated:

- Loss of reactor coolant system integrity due to pressure in excess of the design limits.
- Core damage (especially if secondary system cooling of the steam generators and feed-and-bleed initiation both fail).
- Steam generator tube rupture due to the large pressure difference between the reactor coolant system and the secondary system.

2.1.1.11 Reactor coolant system transients

This accident class encompass various situations, such as CVCS failures, untimely dilution incidents, RCS pressure incidents, assemblies falling or being inserted in an untimely manner and uncontrolled withdrawal of assemblies.

With the exception of dilution incidents and, to a lesser degree, untimely safety injections, these accidents do not have significant consequences, unless the situation deteriorates (pressuriser safety valve stuck in open position, ATWS).

For dilutions, a distinction is made between homogenous dilutions (progressive untimely dilution) and heterogeneous dilutions (insufficiently borated water slug sent into the core). According to Level 1 PSAs (for EDF or the IRSN), homogeneous dilutions resulting in core damage are relatively infrequent compared to heterogeneous dilutions. Based on the studies of heterogeneous dilution conducted to date, it is not yet possible to evaluate containment leaktightness risks. Therefore, in Level 2 PSAs for EDF and the IRSN, a simplified hypothesis of short-term containment loss has been adopted. Consequently, heterogeneous dilutions contribute substantially to the risk of massive early release in these PSA-2.

2.1.2 MELTDOWN FREQUENCY BY CLASS IN EDF’S 900-MWE LEVEL 1 PSA

According to the level 1 probabilistic study conducted prior to the third 10-year visits to 900-MWe reactors [2.1_1], core meltdown frequency is estimated by EDF at around $5*10^{-6}$ per year and per reactor, for all initial reactor states. Frequency for the various accident classes is provided in Table 2.1-1. The conclusions of the IRSN's analysis of this study are not reported here but figure in the report GP n°DSR-50 cited in reference [2.1_2].

Table 2.1-1: Distribution of core meltdown frequency by accident class, according to the results of the PSA-1 conducted by EDF (taken from [2.1-1])

	EDF (pre-VD3 update)	
Accident class	**Core meltdown frequency (/year.reactor)**	**% of overall core meltdown frequency**
H3 (Complete Loss of External Electrical Power)	2.4×10^{-6}	49.0%
Small Break LOCA	9.1×10^{-7}	18.5%
Intermediate Break LOCA	3.3×10^{-7}	6.5%
H2 or Secondary System Transients	3.3×10^{-7}	6.5%
ATWS	2.3×10^{-7}	4.7%
Reactor Coolant System Transients	2.0×10^{-7}	4.0%
Large Break LOCA	1.7×10^{-7}	3.4%
H1 (Loss of Cooling Source)	1.6×10^{-7}	3.3%
SLB	9.0×10^{-8}	1.8%
FWLB	8.3×10^{-8}	1.7%
Loss of Internal Electrical Power	2.2×10^{-8}	0.4%
SGTR	1.7×10^{-8}	0.3%
Combined FWLB-SGTR	5.5×10^{-9}	0.1%
Overall core meltdown frequency	**4.9×10^{-6}**	**100%**

2.1.3 ACCIDENT PROGRESSION

The above paragraphs present a wide variety of scenarios liable to produce limited or complete core damage. However, it should be noted that these scenarios, although initiated by different events, can lead to similar severe accident progressions.

In fact, certain characteristics of the reactor state when core uncovering occurs are considered sufficient to determine the subsequent progression of the accident. These characteristics are primarily used for interfacing between Level 1 and Level 2 PSAs. For example:

- **The point at which core meltdown occurs, because it influences residual power within the core** and therefore the kinetics of severe accident progression.
- **Pressure in the reactor coolant system during core meltdown:** In particular, accident sequences due to deficient RCS energy evacuation, resulting in core meltdown under high-pressure conditions, which involve specific risks to containment integrity.
- **State of safety systems,** especially the availability of the containment heat removal system, which ensures energy is evacuated out of the containment building and radioactive products in the containment atmosphere are sprayed down.
- **Core sub-criticality**

- **Status of the containment,** especially whether isolation has been successful, containment has been bypassed or the access hatch has been closed.

Given these similarities in the postulated progression of severe accidents, the various physical phenomena liable to occur can be investigated by severe accident R&D in a rather general manner.

2.2 GENERAL PROGRESSION OF SEVERE ACCIDENTS

2.2.1 PHYSICAL PHENOMENA POSTULATED FOR PWRs

Figure 2.2-1 schematically presents the major physical phenomena liable to occur during a severe accident, as well as the safety systems involved.

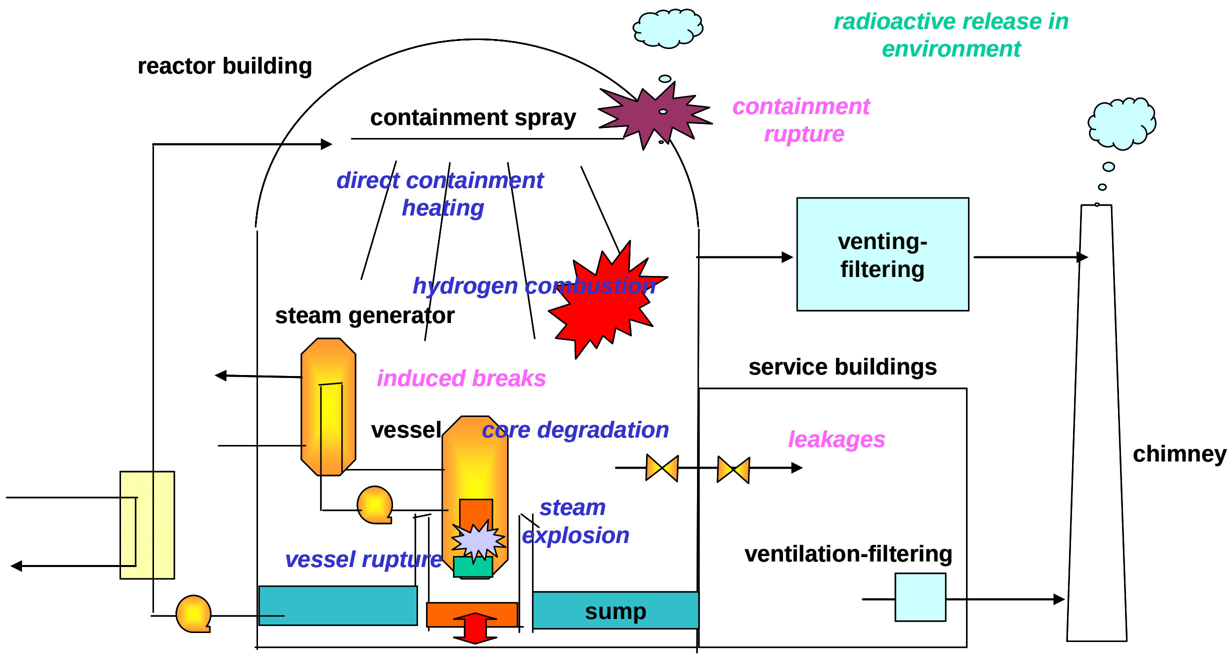


Figure 2.2-1: Physical phenomena during a severe accident

If the reactor core remains unwatered for an extended period of time (**see Section 3.1 on core degradation within the reactor vessel**), nuclear fuel progressively overheats due to unevacuated residual power. Steam initiates an exothermic oxidation of zircaloy fuel cladding, resulting in substantial production of hydrogen and thermal power. Additionally, metal reactions between fuel and its cladding produce low-melting-point eutectics, resulting in relocation of material in the core. As a result of overheating, the fuel first releases the most volatile fission products, then the semi-volatile products (**see Section 7.1 on FP release in the reactor vessel**).

Progressively, a pool of molten material called corium forms in the core and sinks to the lower head of the vessel. When it reaches the water remaining there (**see Section 3.2 on corium behaviour in the lower head**), a corium-water interaction occurs, resulting in coarse fragmentation of the corium, which may be followed by a more violent phenomenon called steam explosion (**see Section 4.3 on steam explosion**). This explosion may cause lower head failure (**see Section 3.3 on vessel rupture**), or part of the core may be transformed into a projectile and break the vessel head. Structural elements might then be projected towards the containment building, threatening its leaktightness.

During core degradation, standby supplies of water can be delivered to the reactor coolant system or the secondary system via "lineups" between the various circuits. Reflooding a degraded core (**see**

Section 6.1.2 on RCS reflooding to enable in-vessel retention) is a complex phenomenon which may enable the accident's progression to be halted under certain conditions. In contrast, reflooding may also increase hydrogen production and cause further release of fission products.

During core degradation, in the absence of secondary system cooling, there is substantial overheating of the inner surfaces of the reactor coolant system and the steam generator tubes. If overheating is coupled with elevated pressure in the reactor coolant system, the latter may rupture in a so-called “induced break” (**see Section 3.4 on high-pressure core meltdown**). This break depressurises the reactor coolant system, but steam generator tubes are involved and RCS valves are open, containment bypass occurs.

Hydrogen produced by core degradation is released into the containment, where it burns on contact with oxygen, provoking a pressure and temperature spike which may damage the containment building (**see Section 4.2 on hydrogen risk and mitigation strategies**). This combustion can either be slow-acting (slow deflagration) or more rapid and, in some cases, explosive (rapid deflagration, detonation). Hydrogen combustion and ensuing containment loss during a reactor core meltdown accident constitutes a risk, and commitment to making this risk residual has been demonstrated by the progressive implementation of catalytic hydrogen recombiners in all units of the operational fleet.

Corium melts accumulate in the lower head and eventually cause its rupture, either by thermal erosion, creep or plastic fracture, depending on pressure conditions in the reactor coolant system. During vessel rupture, and especially under high-pressure conditions, some of the corium is released into the containment and may provoke a pressure spike, resulting in substantial heat exchange between the air and the corium, oxidation of the corium's metallic components and, in some cases, simultaneous combustion of the hydrogen present in the containment building. This phenomenon is called “direct containment heating” (**see Section 4.1**).

Following vessel rupture, corium accumulates in the reactor pit and causes progressive thermal erosion of the concrete basemat, potentially penetrating it, a situation that would cause loss of containment (**see Section 5.1 on molten corium-concrete interaction**). During this phase, a substantial quantity of incondensable gas is liberated, resulting in a progressive increase in pressure within the containment building. To avoid a potential break in this structure, a ventilation-filtration system (U5) has been installed in pressurised-water reactors and can be activated 24 hours after an accident begins, in the event the containment heat removal system fails.

During a severe accident, leaks may occur via penetrations in the containment building as a result of pre-existing leaks or those generated during containment isolation. These leaks may also occur in the sump recirculation circuits, especially in the spray system. Material may be leaked into auxiliary buildings and picked up by ventilation circuits, which are fitted with various types of filtration devices.

For all modes of containment rupture, the release of fission products into the environment depends on the conditions affecting their transfer within the facility. Transfer of fission products depends primarily on their physical and chemical properties — i.e. whether they are gases or aerosols and their chemical form. Iodine behaviour requires particular attention, given its complexity and the significant short-term radiological impact (**see Chapter 7 on release and transport of fission products**). Regarding longer-term accident consequences, particular attention must be paid to caesium releases.

2.2.2 DIFFERENCES BETWEEN 900-MWE AND 1300-MWE PWRs WITH REGARD TO SEVERE ACCIDENTS

There are major design similarities between 900 PWRs and 1300 PWRs, but certain differences exist which may shape severe accident research and its impact. A few examples will be discussed below, including the differences in containment design and the design of control rods in the reactor core.

In the 900-MWe series, the containment building includes a steel liner which, if no defects are present, reinforces leaktightness. For the 1300-MWe series, the containment's mechanical strength under severe accident load conditions is evaluated in different terms because there is no internal liner, but rather a dual concrete containment equipped with an annulus ventilation system which, in the event of an accident, serves to:

- Establish and maintain sufficiently low pressure levels between the internal and external containments so as to force leaks from inside and outside the reactor building towards the annulus.
- Avoid direct transfer of contaminated air outside the reactor building.
- Ensure purification (iodine filtration and trapping) of contaminated air leaking in from the internal containment before release through the stacks.

Therefore, the containment annulus ventilation system plays a critical role in protecting the public by helping limit radioactive releases into the environment.

In 1300-MWe PWRs, Ag-In-Cd and B_4C can be found in the control rods within the core; in 900-MWe PWRs, only Ag-In-Cd is found.

The silver in the 900-MWe PWR control rods combines with iodine to form an insoluble compound in the containment sumps. The presence of Ag-In-Cd in large quantities in 900-MWe units serves to trap iodine.

2.2.3 EPR SPECIFICITIES

The provisions adopted for the EPR project aim to significantly enhance accident prevention. For EPRs, this question is addressed from the design phase. To this end, the technical directives specify that core meltdown accidents (particularly under pressurised conditions) postulated to cause large early releases must be “practically eliminated”; while such accidents remain physically possible, design measures must be implemented to prevent them. The technical directives also indicate that low-pressure core meltdown sequences must be executed in such a way that the maximum conceivable releases only require measures very limited in duration and scope to protect the public.

Regarding core meltdown accidents, the following points can be made:

- The commitment to “practically eliminating” all accidents involving pressurised reactor core meltdown has led designers to integrate a dedicated pressurisation valve, coupled with an isolation valve, for the reactor coolant system under core meltdown conditions, in addition to the standard safety mechanisms protecting this system from overpressure.

- Concerning low-pressure core meltdown accidents, the EPR project involves the implementation of a system for collecting molten fuel, built into the containment building and linked to the reactor pit (**see Sections 6.3 and 6.4 on corium spreading (EPR) and ex-vessel catchers**).

2.2.4 SEVERE ACCIDENT MANAGEMENT

In the event of a severe accident, operating personnel would be called upon to abandon accident procedures in order to follow the recommendations in the Severe Accident Management Guide (SAMG [2.2_1]) and apply the “Containment” and “Standby Measures” sections of the Guide to Emergency Response Team Action (GAEC). The GAEC is not specific to severe accidents; it is used both by operating teams and by local and national emergency teams.

The GAEC provides information on the emergency measures that might be implemented in an accident situation, regarding water injection into the reactor coolant system and the secondary system, as well as identification and isolation of leaks in the containment building.

It details all the procedures operating personnel must follow in the event of a severe accident. The guide distinguishes between:

- Actions requiring no prior assessment by emergency teams, referred to as immediate actions.
- Actions requiring prior assessment by emergency teams, referred to as delayed actions.

Actions recommended in the SAMG serve primarily to maintain containment, aiming to:

- Avoid or minimise airborne releases outside the containment building.
- Provide sufficient time before potential containment loss to allow implementation of the public protection measures described in emergency plans (PUI and PPI in France).

CHAPTER 2 TABLES

CHAPTER 2 FIGURES

CHAPTER 2 REFERENCES

[2.1_1] Note de synthèse concernant l'Etude Probabiliste de Sûreté de niveau 1 du palier 900 MWe
Note EDF CIPNE-EMESM040011 – 24 February 2004 – Non-public reference

[2.1_2] Utilisation des études probabilistes de sûreté dans le cadre du réexamen de sûreté VD3 900 MWe
Tome 1 – Synthèse / Démarche d'utilisation des Etudes probabilistes de sûreté
Tome 2 – Etude probabiliste de sûreté de niveau 1
Tome 3 – Etude probabiliste de sûreté de niveau 2
Rapport IRSN/DSR n°50 – January 2005 – Non-public reference

[2.2_1] Guide d'Intervention en Accident Grave (GIAG) – Version 4 – CPY. Note EDF BE/DPN n°D4510 LT BEM ONC 05 0068 du 25 April 2005 – Non-public reference

CHAPTER 3 : IN-VESSEL ACCIDENT PROGRESSION

3.1 IN-VESSEL CORE DEGRADATION

3.1.1 DEFINITION, OVERALL PHENOMENOLOGY

If the core remains uncovered for a considerable length of time, there is an increase in local fuel rod temperature, which may eventually lead to significant and irreversible core degradation. The mechanisms producing such degradation may be chemical or mechanical. Depending on the temperature attained locally, the consequences may be more or less severe: hydrogen production, fission product release, formation of molten corium and propagation towards the lower head. These phenomena have been studied as part of several national and international programmes [3.1_1], [3.1_2], [3.1_8], [3.1_9], [3.1_11], [3.1_12], [3.1_15]. The main degradation mechanisms and their consequences are described below.

3.1.2 PHYSICAL PHENOMENA

3.1.2.1 Cladding oxidation and hydrogen formation

At temperatures above 1300 K, the zircaloy (Zr) in the fuel cladding is oxidised by steam. This exothermal reaction is very important because the amount of energy released is comparable to that of residual power.

$$Zr + 2H_2O \xrightarrow{\Delta H \approx 132 kcal / mol} ZrO_2 + 2H_2$$

This oxidation produces a zirconia (ZrO_2) layer on the external surface of the cladding. The mass of oxygen absorbed by the cladding and the thickness of the oxide formed are governed by a parabolic time law. The reaction rate varies exponentially with temperature (Arrhenius rate law). Several experimental and theoretical studies have examined thoroughly this phenomenon, which is now well understood. The hydrogen produced may escape from the reactor coolant system (via a break) and mix with containment air, which may lead to explosion and a direct threat to containment integrity. The capacity to evaluate hydrogen production (instantaneous and cumulative) is a key issue in safety studies.

3.1.2.2 Molten metals and their interactions with intact rods

Control rods undergo degradation at lower temperatures than fuel rods, either via melting (Ag-In-Cd) or oxidation/liquefaction (B_4C). Control rod materials (including steel) then relocate towards lower parts of the core, effectively weakening some of the still-intact fuel rods as a result of chemical interactions. It should be noted that spacer grids containing Inconel may also prematurely interact with cladding. The key dissolution reactions include Ag-Zr and Fe-Zr interactions, both of which form eutectics with a melting point below that of zircaloy. Experimental studies have been conducted on these interactions and knowledge levels as well as modelling are satisfactory, especially with regard to Ag-In-Cd. There are still uncertainties, however, as to the influence of B_4C on fuel rod degradation.

3.1.2.3 Cladding rupture

Rising temperature and fission gas formation within the pellets increase pressure inside the fuel rods. Depending on the accident scenario, internal rod pressure can surpass RCS pressure. This overpressure within the fuel causes the cladding to swell as a result of creep. This phenomenon, called ballooning, may lead to cladding rupture. Larger deformations, known as flowering, have also been observed. They are caused by the increased volume of oxide, which puts additional strain on cladding. There are sufficient experimental data on these phenomena, and modelling is satisfactory. In contrast, the mechanisms of cladding rupture above the melting point of zircaloy are much less understood. According to current assumptions, the zirconia scale breaks above a certain temperature (typically 2300—2500 K). Another type of rupture occurs if ZrO_2 thickness is below a critical value (around 300 µm). This remains a very approximate model, however, and as this phenomenon is amongst those still poorly understood. Unfortunately, the experiments required to improve the level of knowledge are both difficult and expensive (real material necessary).

3.1.2.4 Zircaloy melting and fuel dissolution

When the melting point of zircaloy is reached, the fuel is partially dissolved by liquid metal (which does not flow out of the cladding as long as the zirconia layer remains intact). This dissolution may lead to premature collapse of fuel rods, at temperatures below the melting point of UO_2. Melting/dissolution followed by molten corium relocation are the processes which have the main impact on core geometry. They determine whether the core can be cooled by water injection (reflooding) or, if not, whether it can at least be cooled locally. Several experimental studies have been conducted in this area and while the level of knowledge is considerable, modelling is not yet satisfactory. Some experimental results are still difficult to explain or interpret using existing models.

3.1.2.5 Corium progression

The progression of molten material through the core modifies local porosity and flow area, thus having a direct impact on coolant flow (see Figure 3.1-1). This progression depends primarily on the viscosity of the molten mixture, which is a function of its oxidation degree. In the 2100—2900 K range, the effective viscosity of a U-Zr-O mixture is an increasing function of the oxygen level. Knowing how to calculate the oxidation of such mixtures is thus particularly critical to determine corium progression. Understanding of this phenomenon is incomplete, especially since most of the experiments conducted to date (PHEBUS, CORA, PBF) have involved one-dimensional corium relocation. In contrast, it is likely that radial corium progression is also important in reactor cores (the example of TMI-2 tends to prove it, even though the scenario was not typical). There are various models of corium progression and, in general, their results are globally satisfactory. One of the main shortcomings is the lack of validation for 2D and 3D tests. Furthermore, there are still uncertainties regarding the physical properties of corium, particularly solidus and liquidus temperatures, as well as apparent viscosity in the two-phase region (solid-liquid). These properties have a direct impact on corium relocation.

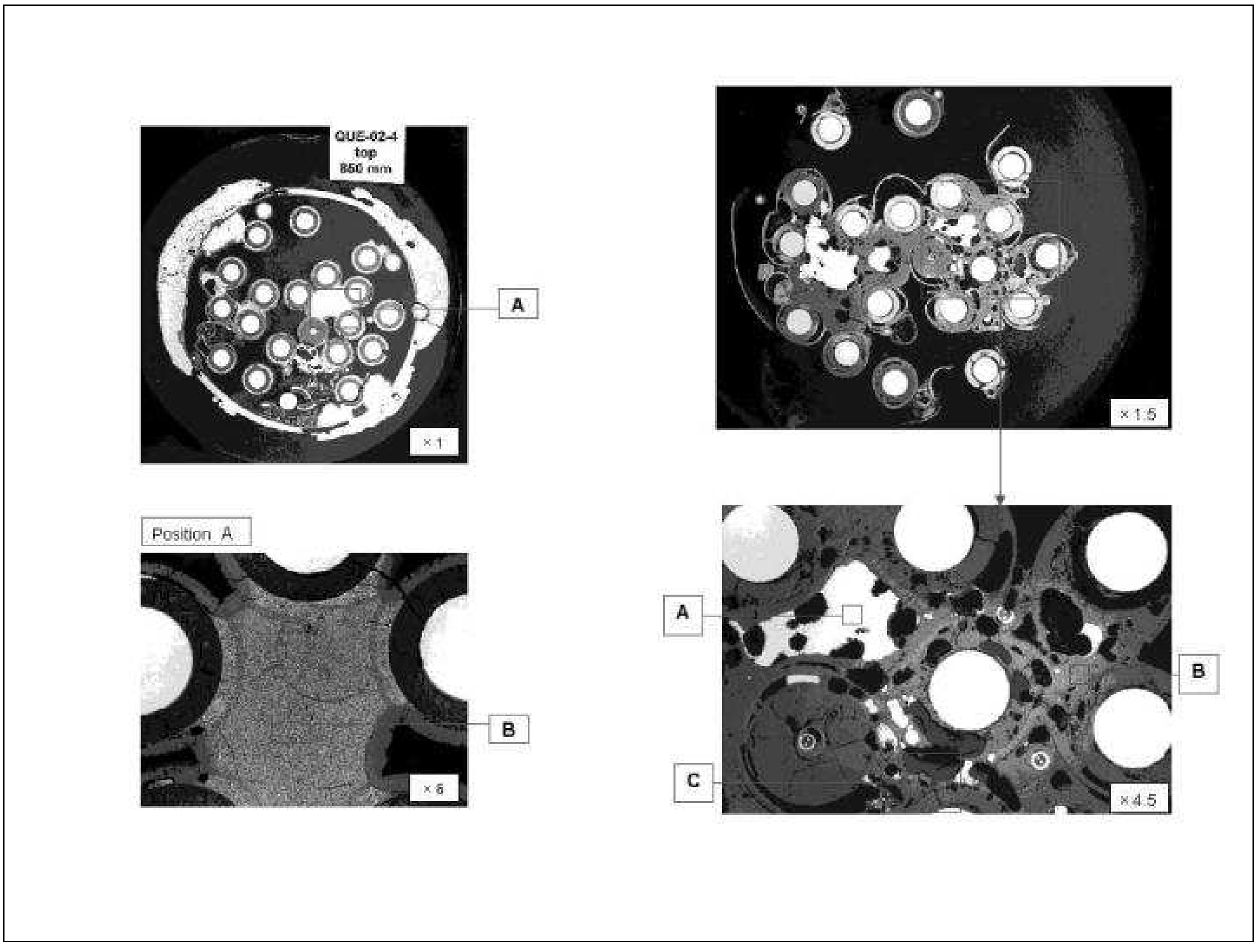


Figure 3.1-1: Photos of two cross-sections from the QUENCH-02 test showing melt distribution in the assembly and impact on degradation

3.1.2.6 Oxidation of molten mixtures

Corium moving through the core contains zircaloy which has not been completely oxidised. The mixture oxidises upon contact with steam. There are no experimental measurements of oxidation kinetics for a liquid U-Zr-O mixture. However, during integral tests such as CORA, substantial hydrogen production has been observed over a very brief period for scenarios involving reflooding or a local increase in steam flowrate. This is particularly important for safety studies. At this time, the main experiments used to understand this phenomenon are QUENCH tests, in which corium is a mix of Zr-O, rather than the U-Zr-O found in real situations. Moreover, these experiments are integral tests, making it difficult to distinguish oxidation from other processes (melt relocation, cooling, etc.). There is a general understanding of the phenomenon, however, and models have been developed for most codes. But some of these models are faulty, particularly those involving spalling of the zirconia layer; this process has not been observed significantly or confirmed in QUENCH testing. For most models, validation is still minimal. In particular, there is a need for more analytical tests aimed at validating models and determining oxidation kinetics for molten mixtures.

3.1.2.7 Formation of a molten pool and corium relocation to the lower head

If the temperature is high enough, fuel melts and a molten pool forms in the core. Due to eutectic formation, temperature remains below the UO_2 melting point (3100 K) by several hundreds of degrees. As the molten mass increases, the pool expands axially and radially in the core until it reaches either the baffle or the lower core support plate, leading to corium relocation towards the lower plenum. It should be noted that a molten pool is very difficult to cool. As a result, it may expand by incorporating the rods located around it. Predictions of the mass, composition and temperature of the materials relocated to the lower plenum, as well as relocation times, are critical to the study of subsequent accident phases. At present, there are no experimental data allowing characterisation of the entire formation and relocation phase for the molten pool originating in the core. Only partial characterisations exist, either in quasi-1D assemblies (PHEBUS), or in preformed debris beds (RASPLAV, ACRR, PHEBUS-FPT4). Models have been developed in most codes. The level of validation and detail is satisfactory with respect to the experimental data available. Nonetheless, more representative data should be obtained to enable the characterisation of pool growth through multidimensional rod assemblies.

3.1.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

3.1.3.1 Experimental programmes

Below is a rapid description of major experiments including recent, ongoing and future projects. Most test programmes have produced data used for code validation. All tests used for this purpose are catalogued in an OECD summary report [3.1_10], including:

- Separate effects tests on chemical interactions: Numerous tests, conducted by various teams (FZK and AECL in particular), have helped determine kinetics of zircaloy oxidation, UO_2 dissolution by molten zircaloy, B_4C oxidation, zircaloy dissolution by molten steel, etc.

- Separate effects tests on cladding rupture mechanisms: These tests (e.g. EDGAR tests) helped determine creep laws for cladding based on cladding temperatures and oxidation states.

- LOFT-FP: This project, completed in 1985, was conducted by the Idaho National Laboratory (INL/INEL, USA) on an assembly of 121 UO_2 rods with nuclear heating (in-pile). It consisted of tests on rod degradation and fission product release, and involved temperatures up to 2400 K (locally). Steam cooling was used, followed by water reflooding.

- PBF-SFD: This project, completed in 1985, was conducted by INEL (USA), on an assembly of 32 rods of non-irradiated UO_2 with nuclear heating (in-pile). It included tests on degradation and FP release, at temperatures up to 2600—3100 K (locally). Steam cooling was used, followed by water reflooding (for some tests).

- NRU-FLHT: This project, completed in 1987, was conducted by AECL (Canada), on an assembly of 16 rods of non-irradiated UO_2 with nuclear heating (in-pile). It involved degradation tests which were unique because of their use of full-scale rods (3.7 metres in height).

- ACRR-MP: This project, completed in 1992, was conducted by SNL (USA). It involved in-pile tests of debris bed melting (UO_2+ZrO_2) in a neutral atmosphere, with temperatures up to 3000—3200 K. The formation and growth of a molten pool were observed.
- CORA: This project, completed in 1993, was conducted by FZK (Germany), on an assembly of 25 rods of non-irradiated UO_2 with electrical heating (out-of-pile). It involved degradation tests at temperatures up to 2200 K (locally). Steam cooling was used, followed by water reflooding (for some tests) [3.1_13], [3.1_14].
- QUENCH: This project, managed by FZK (not finished yet, two tests remain to be conducted), involved an assembly of 25 rods of non-irradiated ZrO_2 and electrical heating (out-of-pile). It consisted of degradation tests and involved temperatures above 2000 K (locally). Steam cooling was used, followed by water or steam reflooding.
- PHEBUS-SFD and -FP: This project, completed in 2004, was conducted by IRSN on an assembly of 21 UO_2 rods (irradiated in the case of PHEBUS-FP) with nuclear heating (in-pile). It involved degradation and/or FP release tests with temperatures up to 2600—3100 K (locally). Steam cooling was used.
- ISTC 1648 (QUENCH): This is a project of the ISTC (International Science and Technology Center, overseen and funded by the European Commission) which is conducted by the NIIAR (Russia). It aims to study reflooding under post-LOCA conditions and includes three items: degradation and reflooding tests using irradiated VVER fuel segments, reflooding tests using a new VVER assembly of 31 rods (without UO2), and the development of a reflooding module for the SVECHA code by IBRAE (Nuclear Safety Institute of the Russian Academy of Sciences).
- PARAMETER: This ISTC project, initiated by LUCH (Russia), involves degradation of prototypical non-irradiated VVER assemblies of 19 rods (similar to QUENCH, but with UO_2 pellets) and allows bottom and/or top-down reflooding, with temperatures up to 2300 K. Two tests are planned, one with slow degradation and top-down reflooding, and a second involving a mix of bottom and top-down reflooding.

As indicated, there are few experimental programmes on late-phase effects, apart from PHEBUS-FP and ACRR. The LOFT and PBF tests attained late-phase degradation, but did not involve detailed analysis of corium progression and fuel rod melting.

Regarding knowledge of reactor core evolution after late-phase degradation, the TMI-2 accident remains the sole point of reference. Detailed analyses of the TMI-2 reactor have been conducted and are available in the literature [3.1_4], [3.1_11], [3.1_19], [3.1_20]. Figure 3.1-2 illustrates the condition of the core following the accident. Of particular interest are the large molten pool within the core, the collapse of a large portion of the rods above the pool (forming a debris bed) and partial corium relocation towards the lower head. Some characteristics of the accident scenario are also worth noting, particularly the high pressures involved and the fact that corium relocation to the lower head occurred after core reflooding (the bottom part of the core was already quenched).

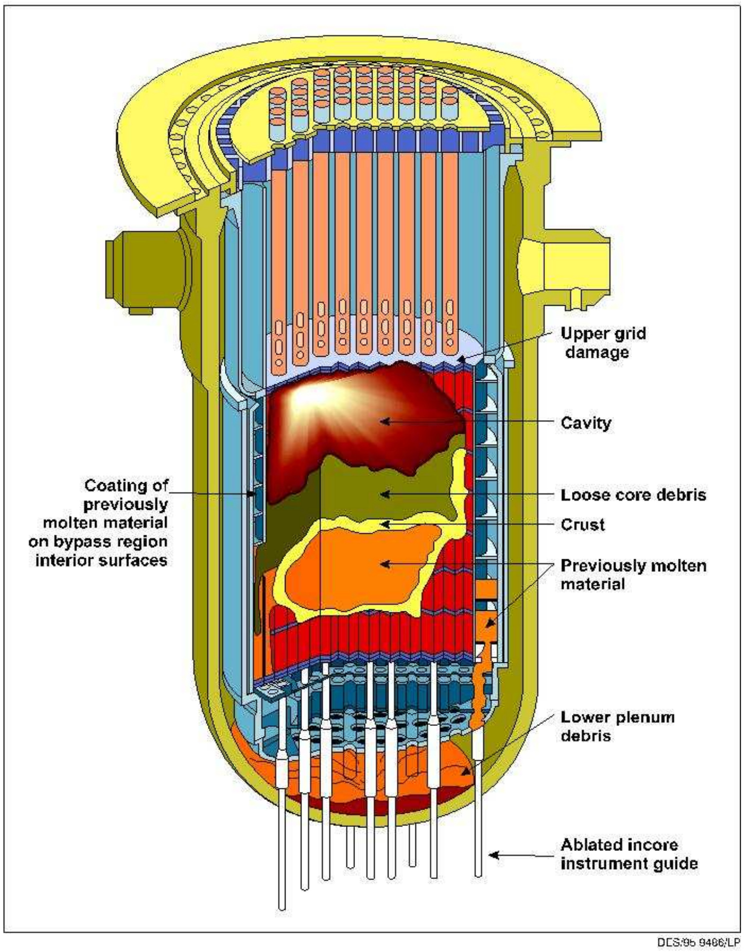


Figure 3.1-2: Schematic diagram of reactor core following TMI-2 accident

3.1.3.2 Modelling and computer codes

The following is a brief description of the main models and mechanistic codes (excluding integral codes) used to simulate the phenomena described above.

SCDAP/RELAP (NRC) is a mechanistic code developed by INEL. It is the result of coupling the RELAP5 thermal-hydraulic code with the SCDAP degradation code and allows core modelling using parallel 1D channels. It includes several models for simulating rod changes: heat transfers, residual power, cladding oxidation, fuel dissolution, cladding rupture and FP release. The development of this code is now complete [3.1_3], [3.1_5].

ATHLET-CD (GRS) is a mechanistic code coupling the ATHLET thermal-hydraulic code with a degradation module. Very similar to SCDAP/RELAP, it allows core modelling using parallel 1D channels and includes several models for simulating rod changes: heat transfers, residual power, cladding oxidation, fuel dissolution, cladding rupture and FP release. The development of this code is ongoing and involves, amongst other components, a dedicated module for the molten pool progression in the core [3.1_16], [3.1_18].

ICARE/CATHARE (IRSN) is a mechanistic code for simulating PWR severe accidents. It is the result of coupling the CATHARE thermal-hydraulic code with the ICARE degradation code. Although similar to SCDAP/RELAP, it has undergone considerable development with regard to late degradation phases. Moreover, it allows axisymmetrical 2D modelling of the core and the reactor vessel. It includes models of several phenomena to allow simulation of rod changes as well as corium progression in the core and lower head: heat transfers, residual power, cladding oxidation, fuel dissolution, cladding rupture, FP release, 2D corium relocation, melt oxidation, rod collapse, formation and growth of the molten pool. Development of this code is ongoing and involves, amongst other components, a model of degraded core reflooding and complete modelling of corium behaviour in the lower head [3.1_6], [3.1_7].

RATEG/SVECHA (IBRAE) is a mechanistic code coupling the RATEG thermal-hydraulic code with the SVECHA degradation module. The major strength of this code is the SVECHA module and its very detailed modelling of some phenomena, particularly cladding oxidation, fuel dissolution, cladding rupture and FP release. Its principal disadvantage is that it was designed for very detailed description of degradation at the assembly scale (or of a representative rod); it is not suited to describe large scale radial degradation propagation and late-phase effects (especially molten pool growth). Development of this code is ongoing and involves, amongst other components, a dedicated module for corium melt oxidation [3.1_17].

3.1.4 SUMMARY AND OUTLOOK

The physics of accident progression in the reactor core is currently well understood and modelled, in particular the phase of rod oxidation and cladding rupture. However, there are still several uncertainties concerning specific late-phase phenomena. This is particularly true for fuel rod collapse and liquid corium oxidation. Additional experimental data are needed to improve modelling, especially for the study of degraded core reflooding (aimed at demonstrating in-vessel coolability, see Section 6.1). But aside from those already underway, no new programmes are currently planned for these themes. Given the high cost of real materials, it seems unlikely that new experimental programmes (national or international) will be initiated in the near future. To reduce the remaining uncertainties, the only alternative is to perform additional analysis of past tests (often insufficiently used) and develop more detailed modelling based on advanced physical analysis.

3.2 CORIUM BEHAVIOUR IN THE LOWER HEAD

3.2.1 DEFINITION, OVERALL PHENOMENOLOGY

It is generally assumed that the lower plenum is filled with water when corium enters that area from the core zone. It also seems likely that this corium contains a substantial fraction (between 25%—80%) of unoxidised zircaloy (cladding material), which is said to be substoichiometric until it attains the composition $(U\text{-}Zr)O_2$. The interaction of this corium (above 2500 K) with water at saturation leads to more or less fine-grained fragmentation of corium jets into particles, causing intense steam production capable of substantially increasing pressure in the reactor coolant system. When the partially fragmented corium accumulates in the lower head, it forms what is called a debris bed. This bed is either very compact, if there is little cooling, or composed of porous solid debris. Debris bed coolability therefore involves a great deal of uncertainty. In any case, corium continues to progressively evaporate residual water. Assuming there is no additional water supply and debris configuration is such that it cannot be cooled, the temperature rises progressively until it reaches the melting point of the steel structures (plates, tubes, etc.) located in the lower head. A substantial quantity of molten steel is gradually incorporated into the corium. As the temperature rises, first zircaloy then oxide debris melt and either form a pool or become part of the existing pool. Corium pool formation in the lower head is a critical step in PWR core meltdown accidents. In such scenario, heat flux at the pool-vessel interface is a key parameter for assessing the vessel mechanical resistance.

3.2.2 PHYSICAL PHENOMENA

There are two main risks when corium reaches the lower head. One is that the steam produced when hot corium comes into contact with residual water will cause a pressure spike, or even an in-vessel steam explosion (see Section 4.3). The other risk is that upon contact with corium, the vessel will undergo a heat flux, which may be of considerable magnitude locally, potentially resulting in vessel rupture. For such scenarios, safety studies should be able to determine the likelihood of in-vessel corium retention and the conditions under which vessel rupture would occur (time, location, characteristics of the corium transferred into the containment). There is thus a critical need to predict the changes corium will undergo, from its relocation towards the lower plenum until its cooling or transfer out of the reactor vessel. The main phenomena governing these changes are briefly described below.

3.2.2.1 Corium jet fragmentation, debris formation

Corium descending as a jet is fragmented upon contact with water (see Figure 3.2-1). This phenomenon is discussed in Section 4.3, which deals with steam explosions. Modelling the fragmentation process is very complex, as this later chapter shows, and remains marked by substantial uncertainty [3.2_7], [3.2_13].

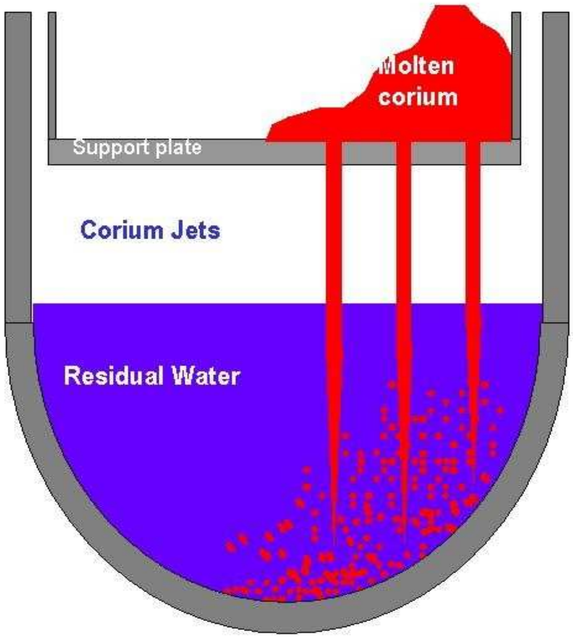


Figure 3.2-1: Schematic diagram of molten corium arriving in the lower head and fragmenting upon contact with water

3.2.2.2 Direct contact of corium jets with the vessel

If there is little water in the lower head or the mass of relocated corium is substantial, corium jets interact only partially with water; there is some direct contact with the vessel. In such a case, the vessel may rupture very rapidly during contact with the corium jets. Although there are few experimental studies of this phenomenon (a few CORVIS tests), it is relatively well understood. One of the key parameters is the degree of corium overheating. According to estimations, corium temperature cannot exceed 150 K over its melting point. An insulating crust may thus form between corium and the vessel, which is in turn thermally protected during this phase of melt progression.

3.2.2.3 Steam explosion

Under certain conditions that are generally understood but not yet precisely quantifiable, the interaction between corium jets and water may provoke a steam explosion (see Section 4.3).

3.2.2.4 Debris bed dryout, reflooding possibilities

Following corium jet fragmentation, particles settle in the lower head and form what is commonly referred to as a debris bed. The debris bed may be very compact if there is insufficient cooling of the droplets formed during jet fragmentation. The main area of investigation focuses on cooling this bed by injecting water into the vessel (see Section 6.1.2). If this is not done, the debris bed progressively dries out and melts, forming a pool which is much more difficult to cool down. There is currently a renewed interest in understanding debris bed reflooding or/and dryout. One-dimensional models, derived from the Lipinski model, provide estimates of the “critical dryout heat flux” based on debris parameters (size, power, etc.). However, 1D modelling is a simplification, and characterising these phenomena relative to the geometry under study still involves considerable uncertainty. Two-dimensional simulations are now used to represent these phenomena.

3.2.2.5 Molten pool formation

As mentioned above, debris bed dryout is a key step in accident progression because it influences molten pool formation or, if only a portion of the corium is fragmented into solid particles, propagation of the existing pool. Given the results of the tests ACRR-MP, PHEBUS-FPT4 and RASPLAV AW-200 ([3.2_12], [3.2_5], [3.2_1]), this phenomenon is now well understood and modelled, for inert atmosphere conditions. Although interactions between liquid steel and (U-Zr)O corium have been studied for many years, the importance of their impact on corium progression in the lower head was recently highlighted (see below). The OECD MASCA project in particular has provided new insights. The progression of a steel-containing debris bed in an oxidising atmosphere has yet to be studied (see Figure 3.2-2).

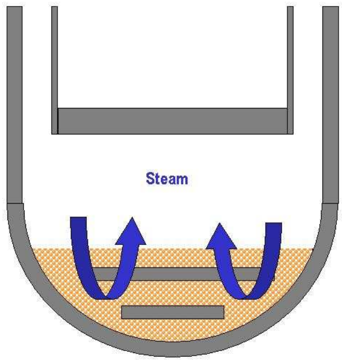


Figure 3.2-2: Schematic diagram of corium configuration following dryout in the lower head; formation of a debris bed (more or less porous depending on fragmentation efficiency) around steel structures

3.2.2.6 Natural convection in the molten pool

The power generated in the molten pool can be evacuated by its lateral edges (thus via the vessel) and by its upper surface (via radiation and convective exchange with water if present). These heat transfers generate the movements of natural convection (see Figure 3.2-3). One of the key parameters for safety studies is the ratio between the upward heat flux and the outward heat flux through the vessel wall. The flow regime is mainly turbulent, except in some areas with a very stratified temperature field (e.g. at the bottom of the lower head). This phenomenon is relatively well understood for simple pool configurations, and correlations for heat exchange at the pool boundaries have been established (see Section 6.1 and [3.2_6], [3.2_14]).

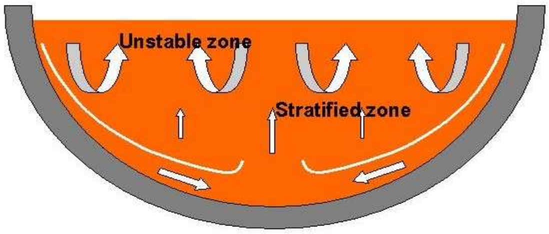


Figure 3.2-3: Schematic diagram of convective structures in a turbulent molten pool with top-down and lateral cooling

3.2.2.7 Focusing effect

In the molten pool, there may be separation between non-miscible materials, resulting in a metal layer floating above an oxide pool. In such a case, if the upper layer is thin, it may result in a local concentration of the heat flux. This phenomenon, well understood and modelled ([3.1_17]), represents one of the main hazards to vessel integrity, particularly if the vessel is not cooled externally. It is explained in greater detail in Section 6.1.

3.2.2.8 Corium oxidation (particles or molten pool), hydrogen production

Aside from particle size, one of the main areas of investigation regarding corium fragmentation is corium oxidation, because of the associated hydrogen production. It also has an impact on subsequent corium progression because the oxidised Zr fraction influences corium viscosity and therefore its rate of relocation. This fraction also affects molten pool stratification in the presence of steel (cf. MASCA results). The ZREX/ZRSS tests (SNL, Zr/ZrO2 or Zr/stainless steel mixture) and the CCM tests (ANL, UO2/ZrO2-corium + 24% steel) have partially answered this question. These tests suggest that, in the absence of a steam explosion, fragmentation is not fine enough to result in significant oxidation of debris. However, oxidation may reach 30% if water is at saturation. In the event of a steam explosion, oxidation may be complete. It should be noted that there have not been enough tests to allow sufficient quantification of this phenomenon (such experimental studies have been discontinued because of the risks involved).

As to molten pool oxidation, this phenomenon has not been extensively studied and most of current models are inadequate. The MASCA-2 tests (evolution of a stratified pool in an oxidising atmosphere) may shed light on this area (after results are interpreted) even though they were not designed to measure oxidation kinetics.

3.2.2.9 Metal/oxide stratification in the molten pool

The MASCA MA and STFM tests, which investigated metal-oxide separation in the U-O-Zr-Fe system at high temperatures, shed light on an equilibrium between two non-miscible liquid phases, one metallic and the other consisting of oxides. Depending on overall composition, the metallic phase, consisting mainly of steel, may incorporate uranium and zirconium, becoming denser than the oxide phase. This leads to stratification with the metallic phase at the bottom (Figure 3.2-4). Phase composition at equilibrium can be predicted using thermochemical databases such as NUCLEA (developed by

Thermodata for the IRSN and the CEA). However, there is practically no modelling of this pool stratification in severe accident simulation codes. Although stratification of two non-miscible liquids is a well-known phenomenon, the coupled interaction between mass exchange (thermochemistry) and flow dynamics (natural convection and stratification) remains a difficult transient process to model.

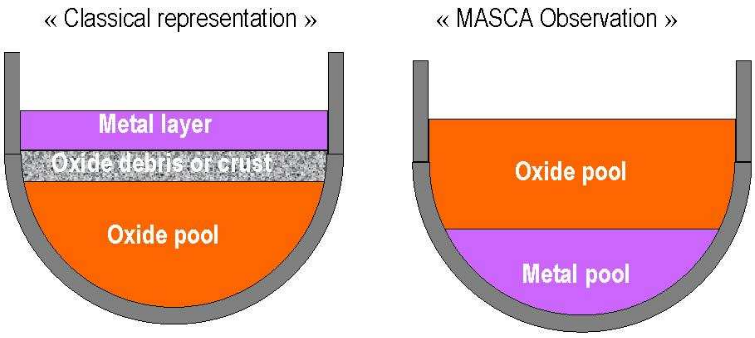


Figure 3.2-4: Layer arrangement assumed in the "classical" envelope approach (left) and observed during MASCA tests (right)

3.2.2.10 Dissolution of reactor vessel steel at sub-melting point temperatures

The formation of eutectic mixtures (Fe-U-Zr) makes it possible to dissolve steel starting at 1360 K. This process may lead to premature corrosion of the reactor vessel steel if it comes into contact with corium (UO_2+ZrO_2+Zr). Corrosion kinetics have been estimated (METCOR test), but the need remains for a more detailed understanding of the process. However, the rate of creep above 1300 K is such that steel no longer has any mechanical strength at such temperatures (see Section 3.3). This phenomenon is therefore of secondary importance.

3.2.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

3.2.3.1 Experimental programmes

Below is a rapid description of major experiments including recent, ongoing and future projects.

DEBRIS: This project, conducted by IKE (Stuttgart), aims to characterise two-phase flows in a debris bed with volumetric heating. The experimental system is one-dimensional and consists of steel balls heated by induction. The first phase involved characterising pressure drops, which is necessary to predict the critical dryout heat flux. The experimental system was recently modified to allow reflooding tests. As the preliminary tests were satisfactory, more quantitative tests are planned ([3.2_16]).

SILFIDE: This project, conducted by EDF, was aimed at characterising dryout of a debris bed with volumetric heating. The debris bed was large enough to observe 2D flow effects, which made these tests a unique reference for validation of 2D models. The debris bed consisted of steel balls heated by induction. Interesting results were obtained despite the challenge of establishing a homogeneous distribution of power. In particular, local flux was at times observed to be higher than the theoretical critical flux (for 3-mm particles, the maximal flux measured during SILFIDE was 1.7 MW/m^2 compared with the 1 MW/m^2 predicted by the Lipinski correlation). Researchers also observed temporary dryout in localised zones, followed by rewetting. This experimental programme was completed in 2000 ([3.2_2]).

RASPLAV: This OECD project was carried out by the Kurchatov Institute (Moscow), with the participation of the IRSN, the CEA and EDF. It was aimed at studying the thermal hydraulics of a molten pool (2D) composed of "real" materials (UO_2+ZrO_2+Zr). The tests, involving up to 200 kg of corium, demonstrated that the heat flux at the boundaries of the molten pool was in agreement with correlations developed from tests with simulant materials. However, it was shown that material interactions could lead to a non-homogeneous distribution, resulting from stratification, for example. Tests using simulant materials (binary salts) were also conducted to study crust formation. This experimental programme was completed in 2000 ([3.2_1]).

MASCA: This OECD project was carried out by the Kurchatov Institute (Moscow), with the participation of the IRSN and the CEA. The RASPLAV experimental facilities were found to provide interesting results on material effects and their impact on heat flux distribution in a pool. The experiments thus focused on how material interactions affected potential stratification of the molten pool and, consequently, flows and heat exchange at the boundaries of the molten pool. The main tests were designed to study the addition of steel and/or fission products and B_4C to corium (UO_2+ZrO_2+Zr). At the same time, a few thermophysical properties of metallic alloys (U-Zr-Fe) or oxides were measured, such as density, viscosity, and solidus and liquidus temperatures. The second phase of the MASCA project was aimed at studying the progression of a stratified corium pool under an oxidising atmosphere [3.2_10]. The Kurchatov Institute has proposed a new programme called CORTRAN to study the progression of a transient debris bed (UO_2+ZrO_2+Zr+Fe) up to the formation of a potentially stratified molten pool, under an inert or oxidising atmosphere. It will not be possible to proceed with this programme under the auspices of the OECD, but it will probably be proposed as an ISTC project.

SIMECO: This project, conducted by KTH (Stockholm), aims to study the heat flux in stratified pools with different power density in each layer. In tests using simulant materials (salts and/or paraffins), three-layer pool configurations were reproduced (similar to those observed during MASCA tests, involving a heavy metal layer, an oxide layer and a light metal layer). This allowed measurement of heat flux distribution along the vessel wall. The results require further analysis but, according to the interpretation of KTH, they show a heat flux distribution different from what could be predicted by classical correlations ([3.2_17]).

METCOR: The objective of this ISTC project, conducted by the NITI Institute (Saint Petersburg), is to study corrosion of a steel sample (representing the vessel) by corium (UO_2+ZrO_2+Zr). The sample undergoes external cooling and is submitted to a heat flux representative of the conditions in a large molten pool (over 1000 K of variation across the sample). Interesting results have been obtained and NITI plans to request a contract extension from the ISTC ([3.2_3]).

LIVE: The overall objective of this project, to be carried out by the FZK with support from the European Commission, is to study the behaviour of a corium simulant in a hemispheric "lower head" (around 1 m in diameter). The simulant is a binary mixture of $NaNO_3$-KNO_3. The first test was designed to study the thermal hydraulics of a pool under steady-state conditions (flux distribution across the inner surface). A second test, proposed by the IRSN and the CEA, was designed to study relocation towards the lower plenum and spreading phenomena, with crust formation.

INVECOR: The aim of this ISTC project, to be conducted by the IAE-NNC-RK (Kazakhstan), is apparently to study corium interactions with reactor vessel steel by pouring premelted corium (UO_2+ZrO_2+Zr) into a hemispheric "lower head" (about 80 cm in diameter) and by using the appropriate setup to maintain power density (this point requires further definition).

3.2.3.2 Modelling and computer codes

The following is a short description of the main models and specific codes (excluding integral codes) used to simulate the phenomena described above.

CFD (computational fluid dynamics) codes: These codes solve Navier-Stokes equations for compressible or incompressible fluids, either in 2D or 3D. They are usually easy to use, and offer several modelling options (turbulence, transport of chemical species). They are used increasingly to study molten pools. However, these codes are intended for rather generic applications and may prove limited for modelling a particular phenomenon (e.g. solid particle formation or stratification).

MC3D (CEA/IRSN): This code is intended for detailed modelling of the corium-water interaction (fragmentation, steam explosion). It is presented in Section 4.3 ([3.2_4]).

CONV 2D/3D (IBRAE): This code solves Navier-Stokes equations for incompressible fluids, in any geometry (2D or 3D). It can be used to calculate molten pool progression and ex-vessel corium spreading, and is similar to a CFD code. At present, its main shortcomings are the lack of turbulence modelling (critical for large molten pools) and the absence of models for the chemical interactions within corium (transport of chemical species and chemical kinetics not modelled) ([3.2_8]).

TOLBIAC (CEA): This is a dedicated model for simulating corium pools in the lower head. It integrates the existence of two non-miscible liquids having the capacity to stratify in either of the two possible configurations; it also accounts for potential crust formation on the pool's upper surface or along its edges. It can be used to calculate transient evolutions in axisymmetrical 2D domains ([3.2_18]).

SURCOUF (CEA/IRSN, under development for ASTEC): This ASTEC module is designed to model debris progression in the lower head by integrating the coupled interaction between thermochemistry and thermal hydraulics. The 0D approach accounts for the existence of several layers (light metal, heavy metal, oxide, solid debris) and calculates their respective positions, based on changes in density.

ICARE/CATHARE (IRSN): This mechanistic code, which calculates core degradation under severe accident conditions, offers axisymmetrical 2D modelling of the reactor vessel and includes several models designed to simulate corium behaviour in the lower head: corium jet fragmentation, debris bed dryout, debris melting, metal/oxide stratification, corium oxidation, debris reflooding. The lower plenum meshing is still rather crude and the numerical methods used are not as accurate as the CFD models ([3.2_9], [3.2_11], [3.2_15]).

3.2.4 SUMMARY AND OUTLOOK

There are still many uncertainties in the prediction of corium behaviour in the lower head. The effects of material interactions (miscibility gap, oxidation, dissolution) appear considerable, and adequate models do not yet exist for all of them (mainly because the experimental results are recent). The programmes currently underway will provide additional experimental data (cf. INVECOR, CORPHAD [dedicated programme to enhance thermodynamic databases], etc.). This should improve current models. After analysis, complementary tests may prove necessary, particularly if there are difficulties transposing the results to a full-scale reactor.

3.3 VESSEL RUPTURE

3.3.1 DEFINITION, OVERALL PHENOMENOLOGY

During a severe accident, vessel integrity may be threatened by various phenomena. When corium from the molten core relocates to the lower head, the vessel wall may be eroded by direct contact with corium jets, or a steam explosion may occur when corium interacts with the residual water in the lower head. If the vessel resists this transient relocation phase, it may then be threatened by the formation of a molten pool in the lower head.

In the event of vessel wall erosion by corium jets, the phenomenon is more intense if there is a large quantity of corium or if the residual water level is low. This may lead to very rapid rupture when the vessel comes into contact with a jet. However, certain experiments, carried out for jet temperatures of almost 2500 K, indicate that a crust forms between the jet and the molten metal, substantially slowing the rate of erosion ([3.3_1]). Other factors may attenuate the consequences of contact between corium jets and the vessel, namely the variable location of impact, the presence of water in the lower head and the generally limited duration of impact. Contact between corium jets and residual water may also lead to very rapid and intense steam generation, resulting in an extreme internal pressure peak and, in some cases, a steam explosion. Such an explosion could be followed by a very large shock wave, capable of damaging the vessel. Despite the low probability of such a rupture, it cannot be ruled out and remains an area of investigation ([3.3_2], [3.3_3]).

Should molten corium form a pool in the lower head, heat exchange between the pool and the vessel may provoke localised overheating (or partial melting), thereby resulting in vessel penetration. This heat exchange is even greater for high-mass corium pools. Corium pool formation and relocation to the lower head was in fact observed during the TMI-2 accident in 1979 ([3.3_4]). However, during this accident, the vessel integrity remained completely intact and later analyses posited that 1) the corium debris was porous, allowing some cooling; and 2) a gap existed between the pool and the vessel inner surface. The gap is believed to have allowed circulation of water or steam. Calculations showed that a small amount of coolant and a narrow gap would suffice in reducing heat exchange between the pool and the vessel ([3.3_5]). It should also be noted that high pressure may have a favourable impact on corium cooling in such scenarios (increased critical flux and vessel deformation from creep or plasticity, potentially enlarging the gap).

Moreover, in PWRs the lower head contains tubular penetrations for guide thimbles, used to introduce instruments for measuring neutron flux. Vessel rupture may be initiated in the zones around these penetrations.

3.3.2 PHYSICAL PHENOMENA

The scenarios examined below involve relocation of molten corium to the lower head and pool formation. The vessel failure time as well as the failure location and size of the breach are considered as key elements because of their role in ex-vessel accident progression.

The physical variables that influence the failure time are mainly the RCS pressure and the internal vessel temperature (related to corium pool mass and configuration). The RCS pressure is generally uniform throughout the vessel, but it may increase rapidly if water is injected into the vessel. The temperature of the structure is strongly linked to the heat flux transferred through the vessel thickness. The failure location depends essentially on temperature distribution inside the vessel, and the zone likely to fail first is the hottest area. Other vulnerable zones are those for which the vessel thickness has been eroded during the corium jet phase (if it occurred), as well as zones that are non-typical due to the presence of penetrations and their welding. Vessel failure can be triggered either by plastic instability or creep. Plastic instability occurs when membrane stress acting in the vessel thickness is greater than the ultimate material strength, which decreases considerably at higher temperatures. Creep, which is an active mechanism of deformation, generally occurs at temperatures above 800 K. When higher temperature reaches throughout the vessel thickness, creep may occur even if pressure levels remain low. Once initiated vessel cracking spreads, and final breach size is highly dependent on the mode of crack propagation. It will be shown in a later section that the propagation mode is directly tied to the metallurgical characteristics of vessel materials. Differences in chemical composition (even in trace elements) may change rupture behaviour at high temperatures; the rupture can be either brittle or ductile. Tests conducted on lower head mockups ([3.3_6] and [3.3_7]) have shown that if two materials behave differently at high temperatures (hot shortness vs. ductility), the final breach sizes will also differ considerably.

3.3.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

As part of their experimental research on lower head rupture behaviour, specifically the scenario of corium relocation and pool formation, Sandia National Laboratories in the US conducted two experimental programmes, LHF (1994—1999) and OLHF (1999—2002) ([3.3_6], [3.3_7]). The second programme, an extension of the first, was carried out with international cooperation via the OECD. LHF involved eight tests and OLHF involved four. Although the same type of 1/5th scale mockup was used for both programmes, thickness was doubled for OLHF to study the impact of the temperature gradient across the vessel wall thickness. Several methods were used to heat the mockup, namely overheating of an azimuthal band (representing a lower head corium pool with maximum heat flux at its free surface), overheating of a localised zone (representing a lower head hotspot) and finally, uniform heating throughout the lower head area (representing scenarios where core melt gradually relocates to the lower head). The experimental protocol called for increasing temperature at a constant rate until mockup rupture occurred. The LHF tests were carried out under constant pressure (seven tests at 10 MPa and one test at 5 MPa). Two of these tests were aimed at studying the behaviour of penetrations. For OLHF tests, only uniform heating was used (see Figure 3.3-1), along with two pressure levels: 5 MPa and 10 MPa. The last two tests of the OLHF programme focused respectively on the behaviour of penetrations (pressure at 5 MPa) and the impact of a 5—10 MPa pressure jump on the rupture mode. During tests with penetrations, weld leaks generally occurred, resulting in termination of the experiments prior to actual rupture of the lower head.

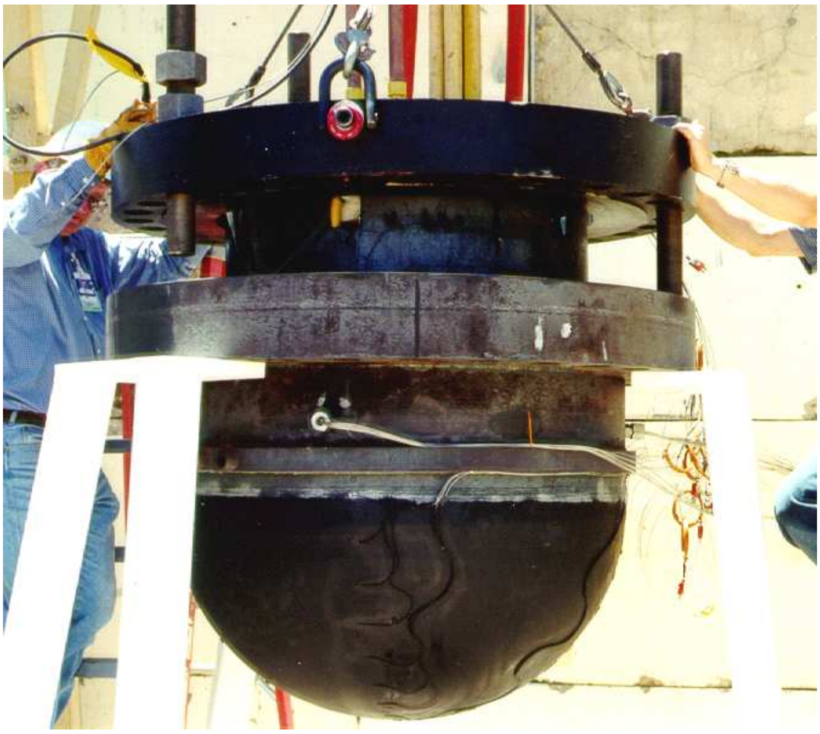


Figure 3.3-1: OLHF 1/5-scale lower head and installation of its induction heating system

In these series of tests, particular attention was paid to failure times and modes, as well as breach size. It should also be noted that the LHF and OLHF programmes enabled development and validation of numerical models needed to study the thermomechanical behaviour of the lower head prior to rupture. The various models developed are briefly presented below:

- Two simplified models (1D and 2D) were developed by the IRSN. The 2D simplified model was recently implemented in the ASTEC and ICARE/CATHARE computer codes. As part of a SARNET initiative, the results obtained using this 2D simplified model, and those from the EDF code (Code_Aster) as well as from the FZD code, are currently being compared.
- 2D finite element models have been developed by other partners of the OLHF programme, namely the AVN (Samcef), the CEA (Cast3m), the GRS (Adina), SNL (Abaqus), the UJV (Systus) and the VTT (Pasula). A benchmarking exercise comparing the 1D and 2D results with the experimental findings of the first OLHF test (OLHF1) concluded that predictions of failure times and rupture locations are generally accurate and consistent with experimental data ([3.3_8]). Figure 3.3-2 compares the final elongation of the lower head in OLHF1 with predictions from various numerical models.
- 3D finite element models were developed by the AVN, the CEA and SNL. Their predictions of failure times and rupture locations were also completely aligned with experimental results. Additional work by the CEA on the Cast3m code enabled analysis of crack propagation for the OLHF1 test and led to estimations of final breach size that were totally in keeping with experimental data. However, this propagation model cannot be applied to the other LHF and OLHF tests.

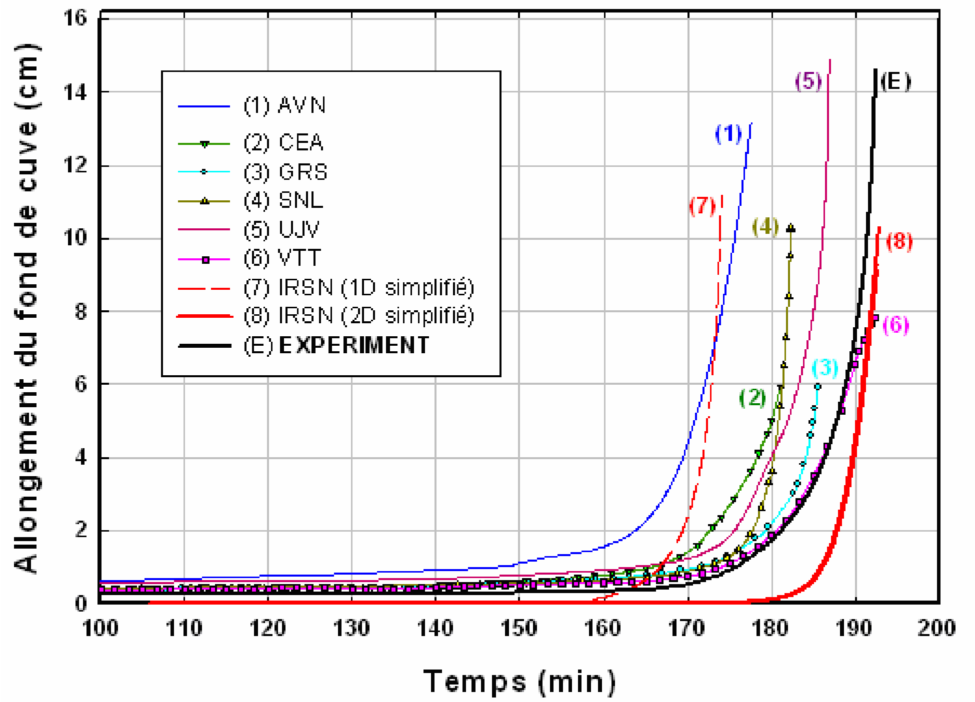


Figure 3.3-2: Comparison of the OLHF1 lower head elongation using different models

Below are additional conclusions based on the results of analyses and interpretation of the LHF and OLHF tests ([3.3_9]):

- LHF and OLHF tests demonstrated variations in the behaviour of vessel materials (hot shortness vs. ductility) at around 1000 °C. This variability seems strongly linked to the presence of certain components (sulphur, aluminium nitride, etc.). The tests also pointed to the impact of this variability on final breach size and the difficulty of integrating it into existing numerical models.

- Thus the experimental results do not provide a means of estimating breach size relative to the loads under study (or extrapolating to a reactor scenario); 3D finite element analysis is at present inevitable and requires a failure criterion capable of accounting for this variability in material behaviour.

In addition to the LHF and OLHF tests, the FOREVER tests ([3.3_10] and [3.3_11]), conducted by the Royal Institute of Technology in Stockholm (KTH), should also be mentioned. These tests used 1/10th scale mockups of a reactor pressure vessel. The experimental protocol involved pouring a molten pool of binary oxide (30% by weight of CaO, 70% by weight of B_2O_3) into the mockups, the binary oxide simulating corium at a temperature of around 1200 °C. With the pool maintained close to this temperature, pressure inside the vessel was then increased to 2.5 MPa until lower head rupture ensued.

The RUPTHER programme ([3.3_12]) is also worth mentioning. Conducted by the CEA from 1995 to 1999 as provided for in the CEA/EDF/FRAMATOME cooperative document, this programme aimed to characterise vessel steel and model vessel behaviour in a PWR subjected to severe accident loads. RUPTHER consisted of three parts: the first focused on characterising 16MND5 steel (tensile strength and creep from 20 °C to 1300 °C), the second involved modelling and the third dealt with experimental validation. The specimens used for the validation tests were tubes subjected to internal pressure and heated to very high temperatures (700—1300 °C). The cylindrical form was chosen for its simplicity and because it allowed control of parameters while remaining representative of the essential elements under study (material, loading and rupture modes). The RUPTHER programme was the first attempt at material characterisation for French reactor vessels (16MND5) in typical severe accident conditions. However, as work progressed certain deficiencies came to light (both in the modelling and the mechanical characterisation of 16MND5 steel). There were also other difficulties, mainly related to the metallurgical complexity of this material. Not surprisingly, the results showed that its metallurgical properties were strongly coupled to its rupture behaviour.

To elucidate the variability in rupture behaviour observed for the vessel materials during the LHF/OLHF tests, the IRSN launched a research programme in collaboration with the CEA and INSA Lyon. The programme focused on materials in French reactor vessels and had a twofold objective: to complete the characterisation database for these materials and to apply study results to French reactors.

The programme began with an inventory of the properties and compositions of the materials used in French vessels (carried out by Framatome), then turned to the selection of five study materials with sufficiently different metallurgical and mechanical properties.

Samples of these five materials were then subjected to high-temperature characterisation tests, which confirmed that certain materials exhibited brittleness around 1000 °C (ductility trough). Identification of the metallurgical factors responsible for this hot shortness also revealed aluminium nitrate precipitates and manganese sulphide precipitates at the grain boundaries and provided insight into their role. This study is in the process of being finalised and work on its summary report is underway. Concurrently, high-temperature characterisation tests (900—1000 °C) on CT (compact tension) specimens have also been initiated to characterise vessel steel properties that are essential for crack propagation kinetics. These tests are very useful for developing crack propagation models. High-temperature testing on tubes will also be conducted by INSA Lyon to characterise the rate of propagation.

Incidentally, several theoretical studies and the CORVIS tests ([3.3_14]), conducted at the PSI in Switzerland, have focused on the behaviour of penetrations under severe accident conditions involving relocation of corium to the lower head. These investigations targeted the length of corium penetration in the tubes and the various possibilities of their rupture (see comprehensive study [3.3_12]). It was found that even if corium penetrates quite far into the tubes, the resulting heat flux is usually not sufficient to bring about the ablation of the tube walls and that RCS pressure and temperature conditions should not cause plastic instability leading to their rupture. Tube ejection following rupture of the welded joint between the lower head and the penetration is also unlikely, as is melting of the retaining flange. It should be noted that the CORVIS tests were primarily carried out on BWR penetrations, which are greater in diameter than PWR penetrations. Therefore, the results obtained tend to penalize the situation in PWRs.

Two finite element models for studying the rupture behaviour of lower head penetrations were also developed by the VTT as part of the OLHF programme ([3.3_15]). The results from these models are consistent with experimental findings.

3.3.4 SUMMARY AND OUTLOOK

The computational models (simplified or finite element) developed by the various partners (AVN, CEA, GRS, IRSN, SNL, UJV and VTT) as part of the LHF and OLHF programmes have demonstrated their capacity to determine the lower head failure time and the zone in which the crack is initiated. The various results are generally consistent with each other as well as with experimental data. For a more precise investigation of crack propagation and final breach size, the only option at present is a 3D finite element model. However such a model, capable of integrating the effects of additional metallurgical elements on the vessel rupture mode, is not yet available. The IRSN/CEA/INSA Lyon experimental programme aimed at material characterisation in French reactor vessels is currently being finalised. It should enable:

- Diagnosis of the brittle/ductile behaviour of vessel steel based on chemical and metallurgical composition.
- Enhancement of the 3D finite element model so that it takes this variable behaviour into account.

In this experimental programme, high-temperature characterisation tests of vessel steel did in fact show ductile/brittle variability around 1000 °C. This variability seems strongly linked to the presence of aluminium nitrate and manganese sulphide. Another related project is the characterisation of vessel materials with regard to crack propagation. The project is currently underway, but there are major technological obstacles to conducting tests of this type (at too high temperatures). The next step should involve developing a model of crack propagation at high temperatures.

3.4 HIGH-PRESSURE CORE MELTDOWN

3.4.1 DEFINITION, OVERALL PHENOMENOLOGY

A severe accident scenario in a PWR involving loss of pressure control may lead to a core meltdown under high-pressure conditions. In such an accident, the steam leaving the core and circulating by natural convection in the reactor coolant system causes considerable overheating which, in conjunction with high pressure, may provoke creep-induced rupture of part of the hot leg piping, the steam generator tubes or, in some cases, another RCS component. The situation is referred to as **high-pressure core meltdown** and the resulting ruptures are traditionally called "**induced breaks**" in the reactor coolant system.

Experiments have shown that in such situations, there is a natural circulation of steam between the vessel and the steam generators (SGs) via the hot legs, due to the presence of a water slug in the RCS crossover legs. Overheated steam from the core moves along the loops by the hot leg and some of the SG tubes where it cools down; it then returns towards the vessel via other SG tubes and the hot leg (where countercurrent flow thus occurs) (Figure 3.4-1). This circulation helps evacuate heat out of the core and transfers heat towards the various structures.

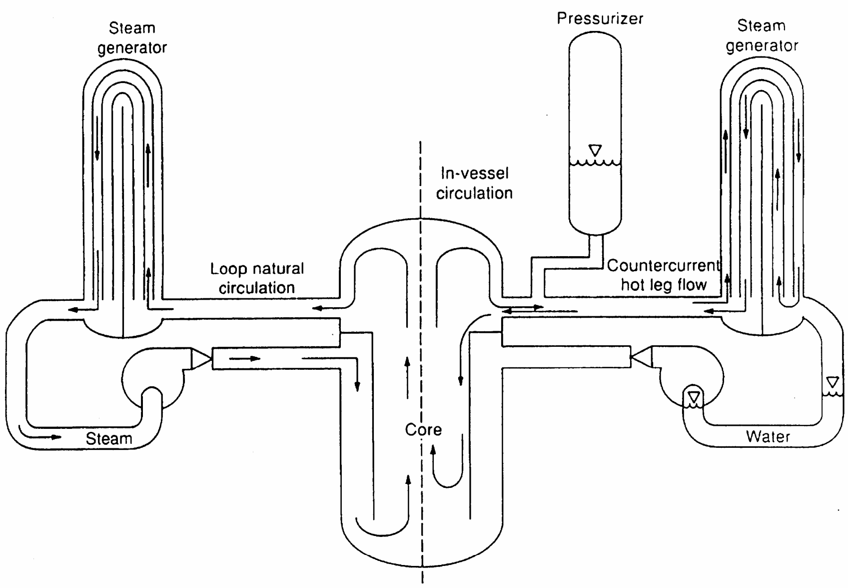


Figure 3.4-1: Modes of steam circulation in the reactor coolant system

The consequences of a high-pressure rupture depend on the location and size of the earliest break, from which point vessel rupture is inevitable. A high-pressure rupture of the vessel lower head can result in direct heating of containment gases and potential loss of containment (see Section 4.1). A rupture in a steam generator (SG tube or tubesheet) results in containment bypass. These phenomena can be avoided if the rupture is located in an area other than the lower head or the SGs. To assess the consequences, break location must therefore be determined.

3.4.2 PHYSICAL PHENOMENA

Estimating break location in the reactor coolant system is based on estimating the chronology of potential RCS ruptures and identifying the earliest one. Such estimation requires knowledge of the thermal and mechanical loads as well as how the structures will react to these loads. This in turn requires knowledge of the mechanical behaviour of component materials at high temperatures. Given that mechanical stress results from internal pressure and the coupling between thermal expansion and boundary mechanical conditions, three-dimensional aspects play a significant role in assessing the collapse of the various structures.

To estimate thermal loads, the power generated in the core (residual power and energy released through the exothermic oxidation of Zr) must be known. Thermal loads are also dependent on convective transport of fission products (FP) in the reactor coolant system. Therefore, modelling the various convection loops and phenomena involved in FP release, transport and deposition is particularly important.

3.4.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

Although all programmes targeting core degradation, fission product release, corium relocation and lower head failure are more or less directly tied to studies of high-pressure core meltdown, various experimental and modelling programmes specific to problems of this type have been conducted. Historically, they were initiated in the United States. One of the first outcomes was the demonstration of gas circulation flows in mockup tests (Section 3.4.3.1). Various computer codes have been adapted to model these circulation flows (Section 3.4.3.3) and can be used to determine changes in thermal loads. Mechanical studies using finite element analysis and based on these simplified loads allow more detailed modelling of structural response (Section 3.4.3.2.2). Mechanical tests (Section 3.4.3.1) reinforce this approach and provide additional modelling for the welding in RCS loops. In recent years, IT development has also made it possible to perform CFD simulations of hot leg circulation flows at a given time (Section 3.4.3.2.1). This partially compensates for the lack of experimental data.

3.4.3.1 Experimental programmes

Westinghouse tests: These tests were aimed at simulating gas circulation and heat exchange during a severe accident in a PWR system. Conducted in the early 1980s, they were financed by the EPRI (Electric Power Research Institute) and involved a simulant fluid in a 1/7th scale mockup of one side of a four-loop Westinghouse PWR (mockup thus included the vessel, two hot legs and two SGs). In particular, these tests revealed recirculation flows in the hot legs and SG tubes, the mixing of hot and "relatively cold" gases in the SG inlet plenums, and gas stratification in the hot legs. They also enabled quantification of the associated phenomena. The tests also estimated the mix ratio in the SG inlet plenums as well as the ratio of "direct" SG tubes (i.e. with gas circulation from the inlet plenum to the outlet plenum) to the number of "indirect" SG tubes (i.e. with gas circulation in the opposite direction). These tests are described in several publications ([3.4_1], [3.4_2] but the results are only partially available), and were used to qualify certain computational tools ([3.4_3], [3.4_4]).

MECI programme: Conducted by the CEA between 2000 and 2004 and financed by the IRSN, this programme was aimed at characterising the mechanical aspects of the “induced break” problem ([3.4_5], [3.4_6]). It entailed material characterisation, burst tests on tubes representative of hot legs, and SG tube burst tests. Concerning material characterisation, the MECI tests enhanced existing data on the various grades of steel comprising the hot legs. They also made it possible to assess uncertainties under creep conditions, determine the properties of materials used in the burst tests and compare these properties with those available in the literature (in particular, the inventory of RCS material properties compiled by Framatome). The main objective of the high-pressure tube burst tests was to validate methods of assessing collapse times for the various structures. They were carried out on SG tubes, and on tubular test specimens representative of hot leg geometry (straight tube, half scale) and hot leg materials. At constant pressure, the test specimens were subjected to a thermal load with a “temperature ramp” (constant rate of temperature increase) until they ruptured. The hot leg tests focused in particular on characterising the behaviour of the various material grades present in the circuits and included tests on mono-material mockups (i.e. entirely comprised of 16MND5 steel or 316L steel) and welded mockups representing real welded joints. A distinction was made between “homogeneous” joints [LH in table below], consisting of a welded assembly of two 316L steel half-models, and bimetallic joints [LBM in table below], consisting of the welded joint between a 316L steel half-model and a 16MND5 steel half-model. The test grid is provided in Table 3.4-1, taken from [3.4_6]. In left-to-right order, the four columns respectively list test materials, mockup thickness, membrane stress and heating curve slope.

Table 3.4-1: Grid for burst tests on hot leg mockups

Material	Thickness (mm)	$\sigma_{\theta\theta}$ (MPa)	ΔT/Δt (°C/s)
316L	10.5	107	0.2
316L	10.5	107	0.05
316L	15	75	0.2
316L	15	75	0.05
16MND5	10.5	107	0.2
16MND5	10.5	107	0.05
16MND5	15	75	0.2
16MND5	15	75	0.05
LBM (16MND5L/316L)	15	75	0.2
LBM (16MND5L/316L)	15	75	0.2
LBM (16MND5L/316L)	15	75	0.05
LBM (16MND5L/316L)	15	75	0.05
LH (316L/316L)	15	75	0.2
LH (316L/316L)	15	75	0.2
LH (316L/316L)	15	75	0.05
LH (316L/316L)	15	75	0.05

Regarding SG tube tests, two pressure loads were studied, representing respectively pressurised and depressurised conditions in the secondary system. These tests were repeated for various temperature slopes and on tubes that were either in good condition or defective (notches or recesses). The test grid is provided in Table 3.4-2, taken from [3.4_6].

Table 3.4-2: Grid for SG tube burst tests

Sample number	Internal pressure (bar)		Temperature ramp (°C/s)		Defect geometry		
	80	150	0.05	0.1	None	Notch	Recess
0	•			•	•		
1	•			•	•		
2		•		•	•		
3		•	•		•		
4	•		•		•		
5		•		•		•	
6		•		•		•	
7	•			•		•	
8	•		•			•	
9		•	•			•	
10		•	•				•
11	•		•				•
12	•		•				•

Figure 3.4-2 below presents the experimental setup and the post-test condition of the hot leg mockup.

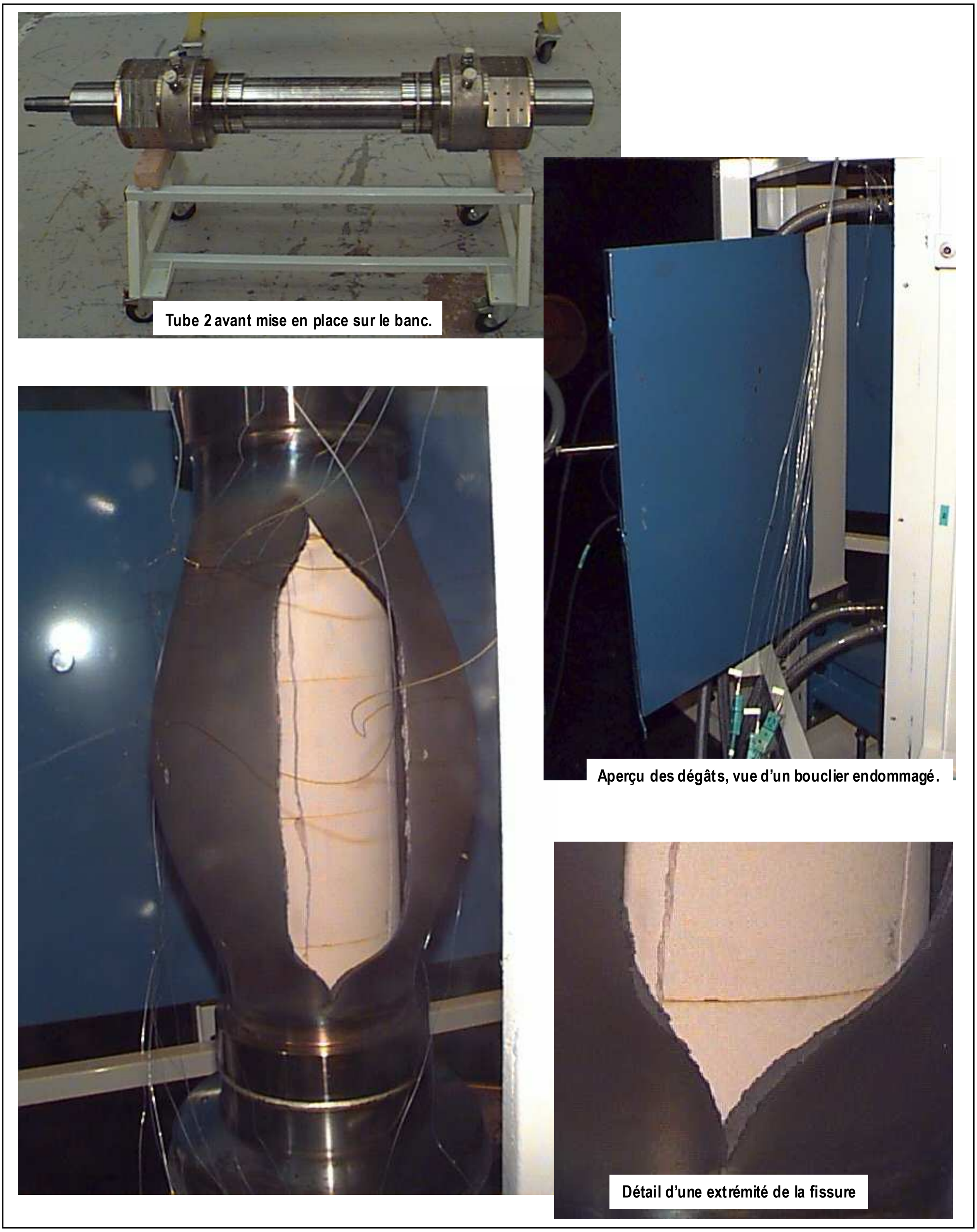


Figure 3.4-2: Rupture via longitudinal cracking in tube NO. 2 — 316 L — 0.05 °C/s — 107 MPa

3.4.3.2 Modelling using CFD or finite element analysis

Computational fluid dynamics and finite element analysis are used to compensate for the shortcomings of large-scale experimental testing. These approaches are used respectively for thermal-hydraulic studies and thermomechanical studies.

3.4.3.2.1 Thermal-hydraulic phenomena

The CFD approach, which involves numerical simulation of gases in an RCS loop, was recently adopted by the NRC using the FLUENT code ([3.4_7]), and by the IRSN with the CFX and TRIO codes (for the latter, as part of a collaboration with the CEA) ([3.4_8]). Using current computational methods, around 10% of SG tubes are modelled; their characteristics are adapted to represent the entire SG bundle. Modelling is limited to the reactor coolant system, and exchanges with the secondary system are accounted for via a boundary condition. This type of analysis provides a detailed vision of flows at a given time. In particular, it enables reassessment of certain flow characteristics (mix ratio in the SG inner plenum, distribution of direct and indirect SG tubes).

One of the noteworthy findings in the results produced for the IRSN is the existence of a third category of SG tubes, characterised by the quasi-absence of gas circulation. The analysis also provides three-dimensional temperature profiles of gases in the reactor coolant system, which are of course impossible to obtain using systems-level analysis. In particular, they point to a potential triple stratification in the hot leg, with a “warm” layer between the hot and cold layers. Figure 3.4-3 shows a thermal profile (hot leg and inlet plenum) from a TRIO-U calculation.

The NRC recently conducted similar studies ([3.4_7]). In particular, and in addition to determining characteristic mixing parameters, the studies analysed the thermal-hydraulic consequences of SG tube leaks based on flowrates and break location (impact on mix ratio and relative number of “direct” and “indirect” tubes).

It should also be noted that in 1998, prior to the NRC studies, Framatome conducted comparative work on gas circulation using the MAAP4 and TRIO VF codes ([3.4_9]). Two-dimensional TRIO VF modelling was used.

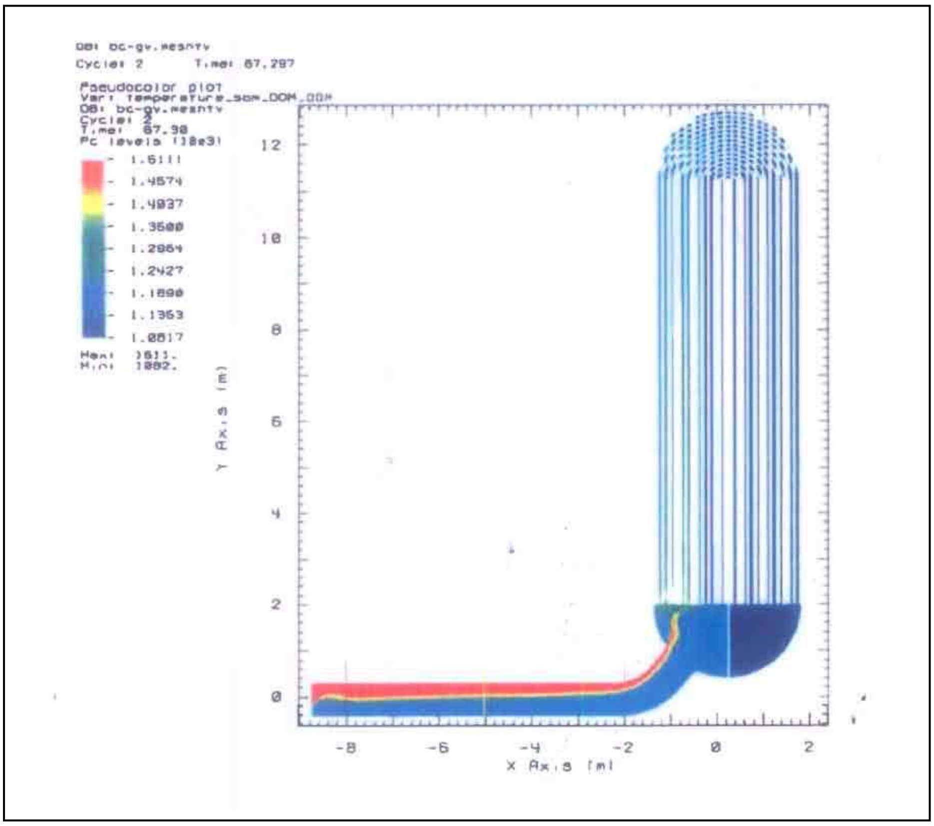


Figure 3.4-3: Example of a thermal field produced by a TRIO-U calculation

3.4.3.2.2 Mechanical phenomena

Finite element analysis for the zone including the hot leg piping and the lower steam generator area was carried out by the CEA for the IRSN ([3.4_10]). It focused on the impact of expansion and mechanical jamming on the state of mechanical stress and, consequently, on rupture times and locations. The resulting model involved varying the geometry of hot leg thermal loads according to the indications provided by CFD analysis. It also accounted for hot leg weld strength via a specific calculation for the damage produced, which involved introducing a corrective factor for creep damage (modification of the Larson-Miller parameter). Creep-induced variations of properties of the various materials were handled in a similar way. Figure 3.4-4 shows the damage to a hot leg (sectional view) at the time of “rupture” (the damage reached a value of 1). Here, the rupture was initiated at the entrance of the elbow upper surface before the SG.

Framatome has focused closely on the tubesheet, developing a special orthotropic elastic model for its study. The IRSN has added to this model by extending it to the plastic domain (perfect-plastic model).

A more specific model was also developed to study the leaktightness of SG manholes ([3.4_11]). It should be noted that all the mechanical calculations are dependent on thermal loads, which must be provided.

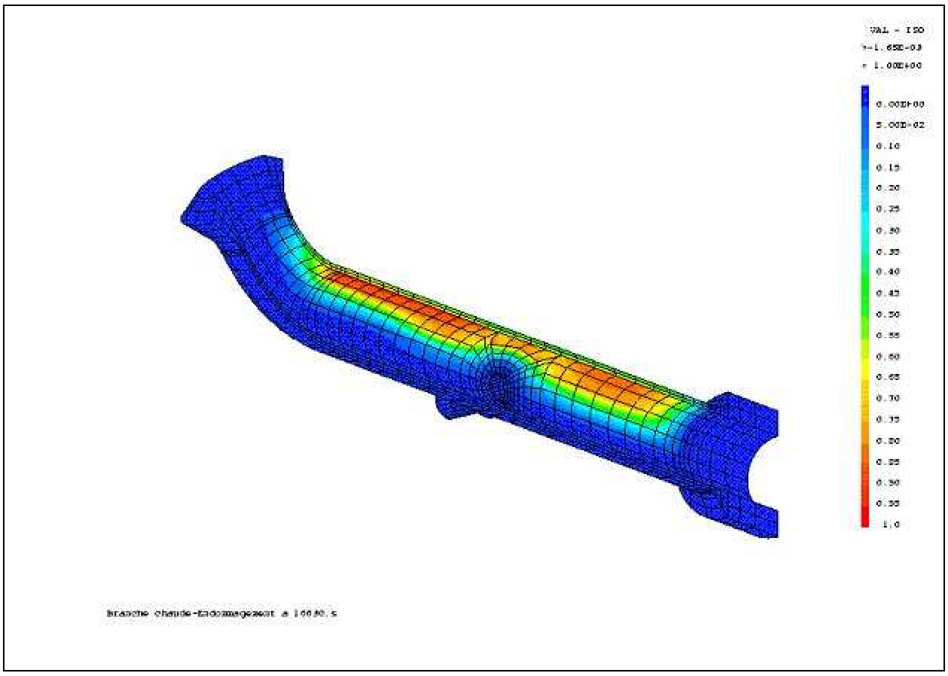


Figure 3.4-4: Mechanical analysis of hot leg strength using the Cast3M code; damage at the time of rupture

3.4.3.3 Systems-level analysis

Given the computational time required, it is not possible to carry out a full thermal-hydraulic analysis of an accident transient using a CFD code. As a result, mechanical strength behaviour of the various structures is still investigated using thermal fields based on systems-level codes with 1D models of the circuits. The codes which have been used for this purpose are RELAP5 (NRC [3.4_12]), MAAP (EPRI [3.4_13], Framatome and EDF), MELCOR (IRSN, NRC) and ICARE/CATHARE (IRSN). All these codes model core degradation as well as circulation flows in the RCS and the secondary system. “Traditional” models of the reactor coolant system must be adapted to allow simulation of recirculation flows and of hot and cold gas mixing in the SG inlet plenums. These modifications involve duplicating legs and volumes.

3.4.4 SUMMARY AND OUTLOOK

3.4.4.1 Needs

The IRSN has enhanced R&D results through its Level 2 PSA studies. The assessment of uncertainties in these studies demonstrates that it is still difficult to reliably predict induced break location. Regarding this difficulty, the following phenomena are particularly relevant:

- Core degradation (power evacuated laterally towards the downcomer, power evacuated towards the upper plenum, power stored in the core, corium relocation).
- Gas mixtures in the vessel upper plenum and the steam generator inlets (SG).
- Phenomena involved in release, transport and deposition of FP.

However, the multiple improvements made to the ICARE/CATHARE systems-level code via 2D axisymmetrical vessel modelling should enable considerable advances in prediction of break location. Nonetheless, there are still major uncertainties concerning the mixing of gases in the steam generator inner plenums. These points can only be ameliorated through complex CFD studies.

3.4.4.2 Future projects

Project ROSA: This OECD project involves a quadripartite agreement between its four participants — EDF, Areva, the CEA and the IRSN. It aims to further development and validation of thermal-hydraulic models in the systems-level codes used for analysing accidental transients in PWR nuclear plants. It will involve tests on the ROSA/LSTF experimental loop at the JAERI institute (Japan), which uses a 1/48th scale mockup with two loops to represent a four-loop 1100-MWe PWR. The tests will simulate six classes of accidental transients. One of the accident groups simulated, involving natural circulation of overheated steam in the reactor coolant system during high-pressure transients ([3.4_14]), may then be used to validate computer codes. This programme began in 2005 and should be completed by 2009. As of this writing, the natural circulation tests have not yet been conducted.

CHAPTER 3 TABLES

CHAPTER 3 FIGURES

CHAPTER 3 REFERENCES

[3.1_1] Adroguer B. et al., "Core Loss During a Severe Accident (COLOSS project)", Proceedings of the FISA-01 meeting, Nov. 2001, Luxembourg.

[3.1_2] Adroguer B. et al., "Corium Interactions and Thermochemistry", CIT project, FISA-99 Symposium, Luxembourg, Nov. 1999, EUR 19532 EN.

[3.1_3] Allison C. M., Rempe J. L., Chavez S. A., "Final design report on SCDAP/RELAP5 model improvements – debris bed and molten pool behavior", INEL-96/0487, December 1996.

[3.1_4] Broughton J., Kuan P., Petti D. and Tolman E., "A Scenario of the Three Mile Island Unit 2 Accident", Nuclear Technology, 1989, 87, pp. 34-53.

[3.1_5] Coryell E. W., "Summary of Important Results and SCDAP/RELAP5 Analysis for OECD LOFT Experiment LP-FP-2", NUREG/CR-6160, NEA/CNSI/R(94)3, EGG-2721, April 1994.

[3.1_6] Fichot F., Gonzalez R., Chatelard P., Lefèvre B. and Garnier N., "ICARE2 Late phase degradation models: Application to TMI-2 accident", Int. Seminar on Heat and Mass Transfer in Severe Reactor Accidents, Cesme (Turkey), May 1995.

[3.1_7] Guillard V., Fichot F., Boudier P., Parent M. and Roser R., "ICARE/CATHARE coupling: three-dimensional thermal-hydraulics of severe LWR accidents", Proceedings of ICONE-9, Nice (France), 2001.

[3.1_8] Haste T. et al., "Degraded Core Quench: A Status report", August 1996, OCDE/GD(97)5, NEA/CSNI/R(96)14.

[3.1_9] Haste T., Trambauer K., "Degraded Core Quench: Summary of Progress 1996-1999", February 2000, NEA/CSNI/R(99)23.

[3.1_10] Haste T., Adroguer B., Hozer Z., Magalon D., Trambauer K., and Zurita A., 2000, "In-Vessel Core Degradation Code Validation Matrix", Update 1996-1999, OECD/GD(94)14, NEA/CSNI/R(95)21, 1996.

[3.1_11] Hobbins R., Russel M., Olsen C., Mc Cardell R., "Molten Material Behaviour in the Three Mile Island Unit 2 Accident", 1989, Nuclear Technology , Vol. 87, pp.1005-1012.

[3.1_12] Hobbins R., Petti D., Osetek D. and Hagrman D., "Review of experimental results on light water reactor core melt progression",1991, Nuclear Technology, Vol. 95, pp.287-307.

[3.1_13] Hofmann P. et al., "Chemical-Physical Behaviour of Light water reactor core components tested under severe reactor accident conditions in the CORA facility", Nuclear Technology, 1997, Vol. 118, 200-224.

[3.1_14] Hofmann P., Hagen S., Schanz G. and Skokan A., "Reactor core Materials Interactions at Very High Temperatures", Nuclear Technology, August 1989, Vol. 87.

[3.1_15] Shepherd I., et al., "Investigation of Core Degradation", COBE project, FISA-99 Symposium, Luxembourg, Nov.1999, EUR 19532 EN.

[3.1_16] Trambauer K., "Coupling methods of thermal-hydraulic models with core degradation models in ATHLET-CD", ICONE-6, © ASME 1998.

[3.1_17] Veshchunov M.S., Mueller K. and Berdyshev A.V., “Molten corium oxidation model”, Nuclear Engineering and Design, Volume 235, Issue 22 , November 2005, Pages 2431-2450.

[3.1_18] Wahba A.B., "International activities for the analysis of the TMI-2 accident with special consideration of ATHLET calculations", Nuclear Engineering and Design 118 (1990) 43-53.

[3.1_19] Wright R., "Current understanding of in-vessel core melt progression", Proceedings of the Dubrovnik meeting, 1995, IAEA-SM-296/95.

[3.1_20] Progress Made in the Last Fifteen Years through Analyses of TMI2 Accident Performed in Member Countries - Rapport NEA/CSNI/R(2005)1, 2005.

[3.2_1] Asmolov V. et al., "RASPLAV Application Report", OECD RASPLAV Seminar, Munich (Germany), 2000.

[3.2_2] Atkhen K, Berthoud, G., "Experimental and numerical investigations on debris bed coolability in a multidimensional and homogeneous configuration with volumetric heat source", Nuclear Technology., Vol.142, 3 (2003).

[3.2_3] S.V. Bechta, .B. Khabensky, V.S. Granovsky, E.V. Krushinov, S.A. Vitol, V.V.. Gusarov, V.I. Almiashev, D.B. Lopukh, W. Tromm, D. Bottomley, M. Fisher, P. Piluso, A. Miassoedov, E. Alstadt, H.G. Willschutz, F.Fichot, "Experimental Study of Interactions Between Suboxidized Corium and Reactor Vessel Steel", Proceedings of ICAPP’06, Reno, NV USA, June 4-8, 2006.

[3.2_4] Berthoud G. and Valette M., "Description des lois constitutives de la version 3.2 du logiciel de pré-mélange MC3D", NT SMTH/LM2/99-39, 1999 – Non-public reference.

[3.2_5] Chapelot P., Grégoire A.C. and Grégoire G., "Final FPT4 Report", DPAM-DIR 2004-0135, PH-PF IP-04-553, 2004 – Non-public reference.

[3.2_6] Chawla T.C., Chan C.H. 1982, "Heat Transfer from Vertical/Inclined Boundaries of Heat Generating Boiling Pools", J.of Heat Transfer, vol. 104, pp. 465-473.

[3.2_7] Cho D.H., Armstrong D.R., Gunther W.H. and Basu S., "Experiments on interactions between Zirconium-containing melt and water (ZREX): Hydrogen generation and chemical augmentation of energetics", Proceedings of JAERI Conference, 97-011, (Japan), 1997.

[3.2_8] Chudanov V.V., Aksenova A.E., Pervichko V.A., 2003, "Development of 3D unified computational tools to thermalhydraulic problems", Proc. 10-th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10) Seoul, Korea, October 5-9, 2003.

[3.2_9] F. Fichot, V. Kobzar, Y. Zvonarev and P. Bousquet Mélou, The Use of RASPLAV Results in IPSN Severe Accident Research Program, in OECD-NEA, editor, Proceedings of RASPLAV Seminar, Munich (2000).

[3.2_10] Fichot F., Seiler, J.-M. and Strizhov, V., "Applications of the OECD MASCA Project Results to Reactor Safety Analysis", MASCA Application Report, OECD-NEA, 2003 – Non-public reference.

[3.2_11] Fichot F., Duval F., Trégourès N. and Quintard M., "The impact of thermal non-equilibrium and large-scale 2D/3D effects on debris bed reflooding and coolability", Proceedings of NURETH-11 Conference, Avignon (France), 2005.

[3.2_12] Gasser B.D., Gaunt R.O. and Bourcier S., "Late Phase Melt Progression Experiment MP-1. Results and Analyses", NUREG/CR-5874, SAND92-0804, 1992

[3.2_13] Magallon D., "The FARO programme recent results and synthesis", Proceedings of CSARP Meeting, Bethesda (USA), 1997.

[3.2_14] Mayinger F. et al. 1975. "Examination of thermo-hydraulic processes and heat transfer in core melt", Final Report BMFT RS 48/1. Technical University, Hanover, W. Germany.

[3.2_15] Salay M. and Fichot F., "Modelling of metal-oxide corium stratification in the lower plenum of a reactor vessel", Proceedings of NURETH11 Conference, Avignon (France), 2005.

[3.2_16] Schäfer, P., Groll, M., Schmidt, W. Widmann, W. Bürger, M., "Coolability of Particle Beds: Examination and Influence of Friction Laws", 2004 International Congress on Advances in Nuclear Power Plants (ICAPP '04), Pittsburgh, PA, USA, June 13-17, 2004.

[3.2_17] A.V. Stepanyan, A.K. Nayak, B.R. Sehgal, "Experimental Investigations of Natural Convection in a Three-layer Stratified Pool with Internal Heat Generation", Proceedings of NURETH11 Conference, Avignon (France), 2005.

[3.2_18] Vandroux-Koenig S. et al., "TOLBIAC version 2.2 code description", NT SMTH/LM2/99-36, 1999 – Non-public reference.

[3.3_1] Saito M. et al., "Melting attack of solid plates by a high temperature liquid jet – effect of crust formation", *Nuclear Engineering and Design*, Vol. 121, 1 (1990).

[3.3_2] Theofanous T. G. et al., "Lower head integrity under steam explosion loads", Nuclear Engineering and Design, Vol. 189 (1999).

[3.3_3] Sehgal B. R. et al., "Assessment of reactor vessel integrity (ARVI)", Nuclear Engineering and Design, Vol. 235 (2005).

[3.3_4] Wolf J. R. et al., OECD-NEA-TMI-2 Vessel Investigation Project. Report TMI V(93)EG10 (1993).

[3.3_5] Stickler L. A. et al., OECD-NEA-TMI-2 Vessel Investigation Project. Calculations to estimate the margin-to-failure in the TMI-2 vessel. Report TMI V(93)EG01 (1993).

[3.3_6] Chu T. Y. et al., "Lower Head Failure Experiments and Analyses". NUREG/CR-5582, SAND98-2047.

[3.3_7] Humphries L. L. et al., "OECD Lower Head Failure Project Final Report". OECD/NEA/CSNI/2(2002)27.

[3.3_8] Nicolas L. et al., "Results of benchmark calculations based on OLHF-1 test", Nuclear Engineering and Design, Vol.223 (2003), pp. 263-277.

[3.3_9] OLHF Seminar 2002 - Nuclear Safety – NEA/CSNI/R(2003)1.

[3.3_10] Sehgal et al., "Assessment of reactor vessel integrity (ARVI)", Nuclear Engineering and Design, Vol. 221 (2003).

[3.3_11] Sehgal et al., "Assessment of reactor vessel integrity (ARVI)", Nuclear Engineering and Design, Vol. 235 (2005).

[3.3_12] Devos, J., et al., 1999, "CEA programme to model the failure of the lower head in severe accidents", Nuclear Engineering and Design, Vol. 191, pp 3-15.

[3.3_13] B. Autrusson, G. Cénérino, "Synthèse des études concernant le comportement mécanique du fond de cuve", Note technique DPEA/SEAC/97-069 – Non-public reference.

[3.3_14] H. Hirschmann, "State of progress", December 1996, CORVIS, Report N°7 February 1997, Paul Scherrer Institute – Non-public reference.

[3.3_15] K. Ikonen, R. Sairanen, "FEM Analysis of OLHF tests with and without penetration", OLHF Seminar 2002, Madrid, June 26-27, 2002 - (Paper from VTT, Nuclear Energy, Finland).

[3.4_1] "Experiments on natural circulation flows in steam generators during severe accidents". Stewart, Pieczynski & Srinivas. Proceedings of the international ANS/ENS topical meeting on thermal reactor safety. Sans-Diego, 1986.

[3.4_2] "Experiments on natural circulation flow in a scale model PWR reactor system during postulated degraded core accidents". Stewart, Pieczynski & Srinivas. Proceedings of the 3rd international topical meeting on reactor thermal hydraulics, Newport, Rhode Island (October 1985).

[3.4_3] "Experiments on natural circulation during PWR severe accidents and their analysis". Seghal, Stewart & Sha, International European Nuclear Society / ANS, Meeting on Reactor Safety. Avignon, France, 1988.

[3.4_4] "CFD predictions of severe accident steam generator flows in a 1/7th scale pressurized water reactor". Boyd & Hardesty. Proceedings of the 10th international conference on nuclear engineering (Icone10), April 14-18, 2002, Arlington, Virginia.

[3.4_5] Programme de travail proposé par le DPEA/SEAC et le DES/SAMS sur l'étude des brèches induites. Dietrich, Dupas, Durin, Goetsch, Gomane. Fiche technique DPEA/SEAC/2000-04 – Non-public reference.

[3.4_6] Brèches induites – Réacteurs à eau pressurisée type 900 MWe – "Synthèse du programme analytique et expérimental réalisé au CEA de 1999 à 2003". J.M. Stelmaszyk. Rapport SAMS n°2. – Non-public reference.

[3.4_7] "CFD analysis of full-scale steam generator inlet plenum mixing during a PWR severe accident. Boyd, Helton & Hardesty". NUREG-1788.

[3.4_8] "Brèches Induites – Simulation « best estimate » d'écoulement dans la branche chaude et le générateur de vapeur". U. Bieder. Note technique CEA/SSTH/LMDL/2004-003 – Non-public reference.

[3.4_9] "Comparaison MAAP4.03/TRIO.VF sur la circulation naturelle des gaz chauds dans le primaire en cas de scénario de fusion de cœur sous haute pression". P. Lalanne. 1998. Document Framatome référence EPDA DC 0152 – Non-public reference.

[3.4_10] "Calculs d'interprétation de l'essai MECI-LBM n°12". Finalisation du modèle global branche chaude et fond du générateur de vapeur. T. Laporte. Rapport CEA/DM2S/SEMT/LM2S/RT/03-028 – Non-public reference.

[3.4_11] "Accidents graves – Etude locale du trou d'homme". J. Kichenin. Rapport SEMT/LM2S/RT/03-004 – Non-public reference.

[3.4_12] "Assessment of the potential for high-pressure melt ejection resulting from a Surry station blackout transient". Knudson & Dobbe. 1993. NUREG/CR-5949.

[3.4_13] "Risks from severe accidents involving steam generator tube leaks or ruptures". Rapport EPRI TR-106194, 1996 – Non-public reference.

[3.4_14] http://www.nea.fr/html/jointproj/rosa.html.

CHAPTER 4 : PHENOMENA LIABLE TO INDUCE EARLY CONTAINMENT FAILURE

4.1 DIRECT CONTAINMENT HEATING

4.1.1 DEFINITION, OVERALL PHENOMENOLOGY

In the event of a PWR severe accident resulting in core meltdown, a corium pool containing uranium and zirconium oxides as well as non-oxidised metals (zirconium and steel) may form in the lower head. If lower head rupture occurs, and vessel pressure exceeds containment pressure, corium is ejected along with steam and, in some cases, the remaining liquid water. The ejected materials are sent first towards the reactor pit (also called cavity), then towards the various containment compartments linked to the pit. Depending on the level of pressure in the vessel, the ejection will finely fragment the corium and disperse it outside the reactor pit. This is accompanied by very efficient heat exchange between corium and gases, as well as oxidation of corium's metallic components, producing hydrogen. The temperatures involved allow more or less rapid combustion of the hydrogen present at the time of vessel rupture, combined with the hydrogen produced by oxidation. These processes lead to overheating and pressurisation of the containment atmosphere, and may result in damage to the containment building and loss of containment integrity. Collectively these phenomena are referred to as “Direct Containment Heating” (DCH).

4.1.2 PHYSICAL PHENOMENA

High-pressure ejection of corium and steam out of the vessel is characterised by various stages: single-phase liquid corium jet, two-phase corium and steam jet, gaseous steam jet. The duration of these stages depends on the mass of corium relocated to the lower head, the sectional area and location of the break, and the pressure in the reactor coolant system. Pressurised ejection initially causes fragmentation of corium into liquid droplets. A flow of steam and corium through the cavity is then established. This very complex flow is strongly influenced by the pit geometry. It is also subject to various phenomena (Fig 4.1-1): projection of corium onto the pit walls and formation of a liquid film along these walls, entrainment and fragmentation of the film by the flow of steam and formation of corium droplets, coalescence and/or fragmentation of these droplets, etc. As a result, part of the corium is entrained by the steam into the areas adjoining the pit, whereas a portion remains trapped in it. During this entrainment phase, the steam and the droplets interact thermally and chemically. Consequently, steam temperature and cavity pressure increase considerably. Hydrogen is produced by the exothermic reaction between the corium metals and steam. However, hydrogen combustion is not possible in the cavity because its atmosphere contains little oxygen which is swept out by the steam flow. When hot gases and corium particles enter in the containment, they contribute to the overheating and rapid pressurisation of its atmosphere. The larger the mass of corium dispersed and the finer its fragmentation, the larger the increase in heat transfer and containment pressure. The distribution of corium in the various containment areas and the duration of discharge also play an important role in the degree of pressurisation. Moreover, when the extremely hot gases and corium particles enter the containment, they will provoke more or less rapid and complete hydrogen combustion. This combustion is very complex because it combines the characteristics of a diffusion flame (the flow causes a jet of corium and hydrogen and the entrance sections of the containment) with the characteristics of a premixed flame (combustion of the initially present hydrogen). It will increase the pressure spike in the containment if the characteristic combustion times are close to those of corium discharge and dispersion. Furthermore, determining characteristic combustion times is complicated by the complex nature of combustion during DCH.

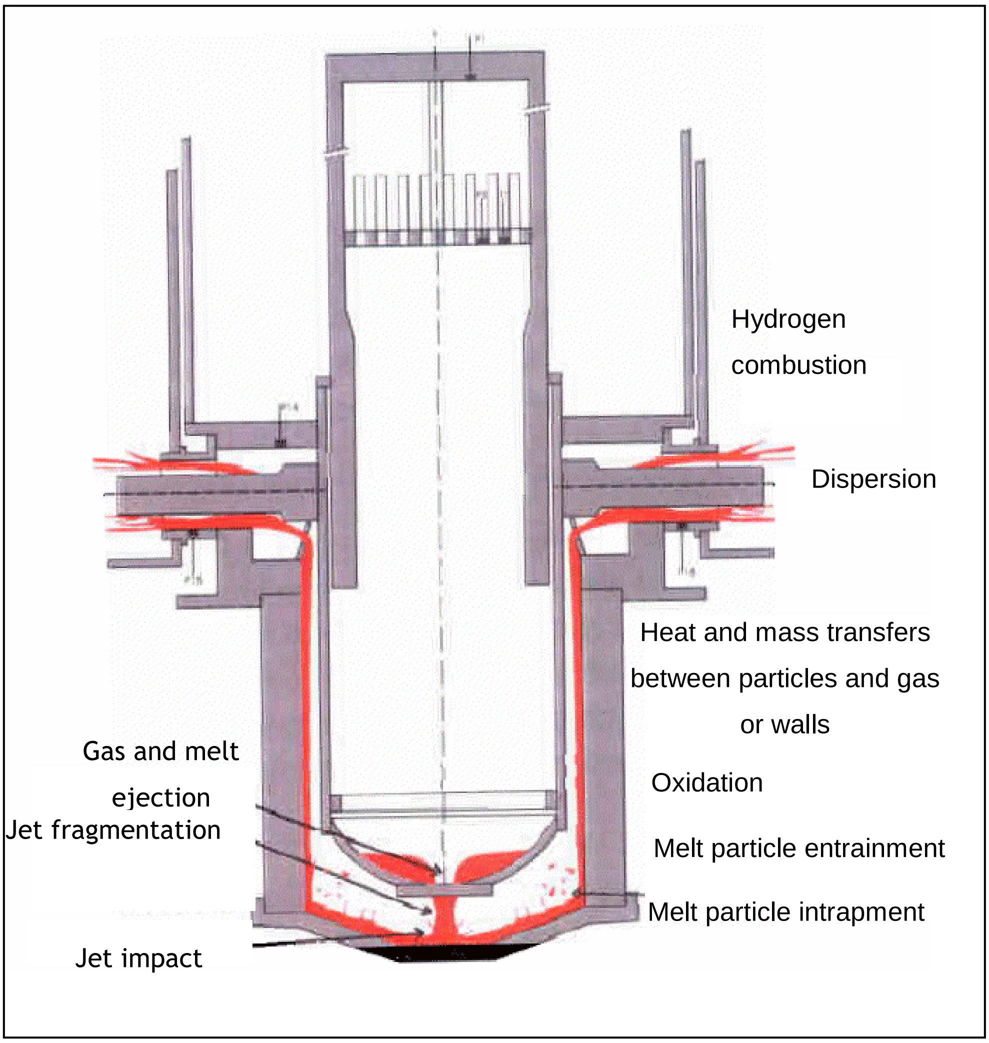


Figure 4.1-1: Physical phenomena during direct containment heating

4.1.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

One of the ways to investigate DCH phenomena is to perform tests on mockups, which provide small-scale reproduction of the main geometrical characteristics of reactors. Based on the results of these tests, models and correlations are developed for simulation software, which is then used to study reactor-scale phenomena.

4.1.3.1 Major experimental programmes

Numerous experimental programmes have been developed to study DCH phenomena [4.1_1]. Only the main tests, which fall into three categories, are presented below:

Simulator tests at low melting temperatures (dynamic aspects)

These experiments focus specifically on the dynamic aspects of DCH. They are separate effects tests aimed at characterising corium dispersion and establishing correlations for entrainment of corium droplets out of the cavity towards the containment and the adjacent compartments. These correlations are a function of the experimental parameters, which generally include the size of the lower head break, vessel pressure, and physical properties of simulant materials and the carrier gas exiting the vessel. Several cavity geometries have been studied and various simulant materials have been used for this type of test: water, oils, Wood's metal, gallium. The advantage of these last two metallic alloys is that they possess properties (density, viscosity, surface tension) similar to those of corium. The key results of these tests are characteristic melt times and the distribution of the dispersed material, as well as the pressure changes over time in the vessel and the cavity. In the late 1980s and early 1990s, several tests were conducted on mockups of American reactors at scales varying from 1:40 to 1:25 [4.1_2]. Two types of geometry were analysed. The Zion reactor is representative of the first type. Its geometry is referred to as “closed”, with no direct pathway towards the dome (there is an indirect pathway between the cavity and the dome via an instrument tunnel leading to the intermediate compartments). The Surry reactor is representative of the second geometry, referred to as “open”: there is a direct pathway towards the dome, via the annular section around the vessel. This geometry is closer to that of French reactors. In the closed geometry (Zion), a substantial fraction of simulant material remained trapped in the intermediate compartments, allowing only a small quantity to enter the containment. In 1997, KAERI (South Korea) conducted a test campaign for the IRSN. These tests were the first to study the phenomenon in a geometry close to that of a French 900-MWe reactor, at a scale of 1:20 [4.1_4]. What makes this geometry unusual is the existence of a major, direct pathway between the cavity and the containment via the annular section. These tests showed that, for the case of water, when vessel pressure is above 1.5 MPa, 80% of the simulant material is entrained through the annulus, then evacuated towards the containment. In contrast, at pressures below 0.2 MPa, more than 90% of the simulant material remains in the cavity. These tests were limited to water as the simulant material. More recently, tests were conducted at the FZK in Germany, using the DISCO-C facility, which represents the EPR cavity at a scale of 1:18, with a direct pathway from the cavity to the containment. These tests were aimed at studying corium dispersion at vessel pressures below 1.5 MPa and how the mode of lower head rupture (break in the centre, lateral break or partial unzipping and tilting of the lower head) affected the dynamics of simulant material ejection ([4.1_6]. These tests also showed that a greater mass of material was dispersed in cases of central vessel breach. Since these tests, the EPR geometry has evolved, and does not include a direct pathway between the cavity and the containment.

Figure 4.1-2: DISCO-C test representing P'4 geometry

Recently, the DISCO-C facility was modified to enable study of dispersion phenomena for the geometry of P'4 reactors (Figure 4.1-2). The cavity is largely deeper than that of the EPR system, and there are three possible outlets, leading respectively into the containment, the compartments and the cavity access corridor. For this type of cavity, a smaller fraction of material is dispersed via the annulus (60%), since a substantial portion of the fuel (~30%) is trapped in the access corridor [4.1_7]. It should also be noted that of the 60% ejected, around 1/3 goes directly towards the dome; the rest moves towards the compartments.

Simulator tests at high melting temperatures (thermal and chemical aspects)

In addition to the dynamic aspects, these tests provide insight into heat exchange phenomena and, to a lesser degree, chemical interactions. The simulant material used for this type of test is a mixture of iron and alumina (Al_2O_3), the product of a thermitic reaction. Many of the tests have been conducted in the United States, mainly by Sandia and Argonne National Laboratories (SNL and ANL respectively). They involve two types of geometry ("closed" for Zion and "open" for Surry) and various experimental conditions. For the closed geometry (Zion), the integral effects tests confirmed that the intermediate compartments retained 90% of the simulant material and that containment overpressure was due solely to the ejection of hot gases out of the vessel, and was thus very limited.

In addition to the DISCO-C facility, the FZK also has another facility, DISCO-H, specially designed for simulant tests at high melting temperatures. The EPR and P'4 geometries were studied for vessel pressures varying from 0.7 MPa to 2.5 MPa. For the P'4 geometry, the compartment effect seemed to play an important role, and only the fraction dispersed directly into the containment seemed to have a measurable impact on containment heating. For reasons not yet elucidated, a substantial portion of the fuel moves towards the subcompartments. The small size of these compartments lowers heat transfer efficiency, which strongly limits pressurisation [4.1_5].

The impact of the presence of water, either in the cavity or directly in the vessel, has been summarily studied in the United States. In the three CES tests (Surry open geometry, [4.1_3]), the corium simulant started off in the lower head, and water or steam was ejected from the vessel, which was first pressurised to 4—8 MPa by a break 4 cm in diameter (1/10th scale). According to the findings, water that was initially saturated (thus undergoing flash vaporisation as it left the vessel) had no particular impact. For water at 300 K (does not vaporise following depressurisation), a considerable reduction (30%) in the pressure loads was observed. In these tests, around 60% of simulant material was dispersed towards the containment dome via the annulus. Several tests were conducted by adding a small quantity of water to the cavity (IET and WC tests), which did not produce a significant impact either.

Very early on, the chemical phenomena of oxidation and combustion were recognised as potentially having a major impact. The first significant experiments to focus on these effects have been the IET tests for the two geometries. The initial concentrations of hydrogen (in the containment) were on the order of 2%—3%. This always produces very intense oxidation. The level of combustion is around 70%, producing a two- or three-fold increase in pressure (Figure 3.4-3).

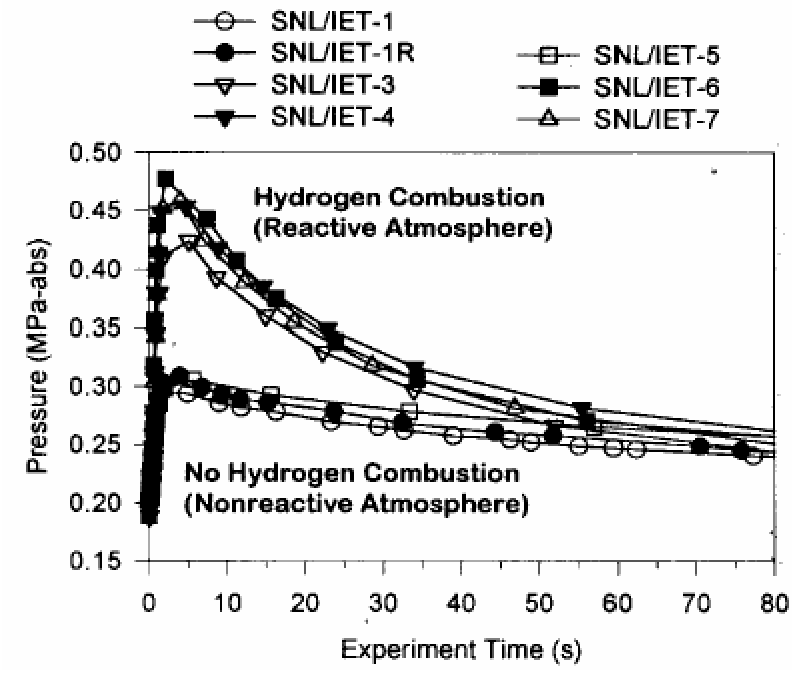


Figure 4.1-3: Impact of combustion on pressure levels in IET tests (SNL)

The DISCO-H tests conducted by the FZK (EPR and P'4 geometries) have confirmed earlier findings as to the role of combustion, which appears critical. In these tests, high initial levels of hydrogen increase the likelihood of complete reaction. Burnup is on the order of 80% for initial hydrogen levels of 4.5%—

6%. Combustion characteristics remain rather vague, and modelling burnup as well as combustion kinetics is a key challenge.

Tests using real materials

These tests are a precious source of information because they constitute a first step towards extrapolating dynamic, thermal and chemical aspects to reactor scenarios. Following IET tests using thermite, similar experiments were conducted with a mixture of UO_2, Zr, ZrO_2, Fe and Cr at the COREXIT facility (1/40th scale mockup of the Zion reactor) in order to demonstrate the effect of using real materials in place of thermite. The tests showed that low vessel pressure (0.3 MPa and 0.4 MPa), together with the mixture's low specific energy (~1.2 MJ/kg for real corium vs. 2.7 MJ/kg for thermite) and its higher melting point, result in reduced pressurisation of the dome compared with results from the same tests conducted with thermite in alumina (0.11 MPa for U1B vs. 0.15 MPa for IET-1RR). However, hydrogen production was higher, by around 20% in moles. This outcome is related to substantial quantities of metals which are more reactive (Zr). Furthermore, tests with real corium indicate that pre-existing hydrogen in the dome does not burn as rapidly as hydrogen produced by DCH, a finding supported by tests with simulant corium.

4.1.3.2 Modelling and simulation codes

Two classes of codes exist:

- 0-D “parametric” codes (MAAP, CONTAIN, RUPUICUV), which uses simple models and offers broad flexibility for input data; generally included in integral codes.
- “Mechanistic” codes for multidimensional multiphase simulation (AFDM and MC3D), which is important for interpreting experiments, understanding various phenomena and developing simple models.

4.1.3.2.1 Parametric models

Parametric models are not aimed at studying and understanding phenomenology, but they do reflect the level of knowledge for a given process.

Using the DCH module of the American code CONTAIN ([4.1_9]), it is possible to describe corium debris transport, the flows between containment building compartments, debris trapping by various structures, chemical reactions (corium oxidation), heat transfers between debris and the atmosphere (through convection or radiation) and combustion of hydrogen produced by DCH as well as pre-existing hydrogen. This 0-D code is by far the most advanced of its kind, offering an impressive collection of computational options and backed by a solid qualification database. However, validation was limited to American reactors with Zion and Surry geometries, and recent use of the codes by the GRS for DISCO tests on EPR and P'4 geometries did not produce satisfactory results. This may also be due to the code's complexity: due to the many options, it requires a high level of expertise. The DCH module of MAAP is based on the correlation known as Kim (KAERI) and used to assess the total fraction of corium dispersed as a function of initial state (vessel pressure and break diameter). The droplets are assumed to be in dynamic and thermal equilibrium with the gases. The distribution of droplets between the various cavity outlets thus depends on gas flowrates. Only steam is consumed by oxidation of the dispersed corium droplets (Zr, Cr, Fe and Ni successively). Neither the oxygen in the air nor the oxidation of spread corium is taken into account.

The ASTEC code assesses pressure loads during DCH using the RUPUICUV, CORIUM and CPA modules. The cavity phenomena are addressed by the RUPUICUV module ([4.1_8]). The total dispersed fraction is given by a correlation developed by the KAERI based on their tests with the 900-MWe geometry (similar to the Kim correlation used in MAAP). The droplets are assumed to be in thermal equilibrium with the gas in the cavity, but containment heat exchanges are assessed using simple convection models. Beyond a threshold temperature, steam oxidation of corium metals becomes possible, producing hydrogen and heat. Pressure and temperature loads as well as hydrogen combustion are also modelled.

4.1.3.2.2 Simulation codes

The IRSN and the FZK have decided to use multiphase thermal-hydraulic simulation codes to enhance knowledge and facilitate the development of simple models, at the same time enabling extrapolation to other scales, materials or reactor types.

AFDM is a fluid dynamics code currently used by the FZK. Initially developed for safety analyses of Fast Breader Reactors, it was a precursor to the SIMMER III code. Physical DCH models have been added to AFDM, including the model for the exothermic reaction between metals and steam or oxygen, and the parametric model for hydrogen combustion in the containment ([4.1_10]). The code also addresses entrainment and formation of corium films on the cavity walls, the thermal exchanges between the films and the walls, and crust formation. Unfortunately, the code is limited to 2D geometry (axisymmetrical). Promising results have been attained for DISCO experiments.

MC3D is a multiphase thermal-hydraulic code developed by the IRSN and the CEA. It is mainly used for assessing pressure loads during a steam explosion (see Section 4.3.3.1), but it can also be used to study a number of multiphase phenomena, including DCH. The defining feature of this code is its detailed description of corium: the droplet field (dispersed corium) is separate from the “jet” field (continuous corium) (see Figure 4.3-6). A detailed model of fragmentation and coalescence allows the fuel to move from one field to another. A parametric model of oxidation exists, but as of this writing, it does not address combustion. The latest modifications to the code, making it possible to use several incondensable gases, should correct this shortcoming. MC3D also enables three-dimensional assessment, which means the complex geometries of French reactors can be taken into account. As an example, Figure 4.1-4 shows a representation of reactor geometry (simplified P'4) as well as the results for dispersion via the annulus, as a function of vessel pressure. Providing more than simple direct comparison with a few experimental points, the code makes it possible to study functional dependence on parameters such as vessel pressure. For the conditions described in Figure 4.1-4, the code makes it possible to predict a dispersion threshold pressure of around 5 bar.

It would also be quite natural to study with MC3D water's impact, given that the code's primary purpose is to study interactions between corium and water.

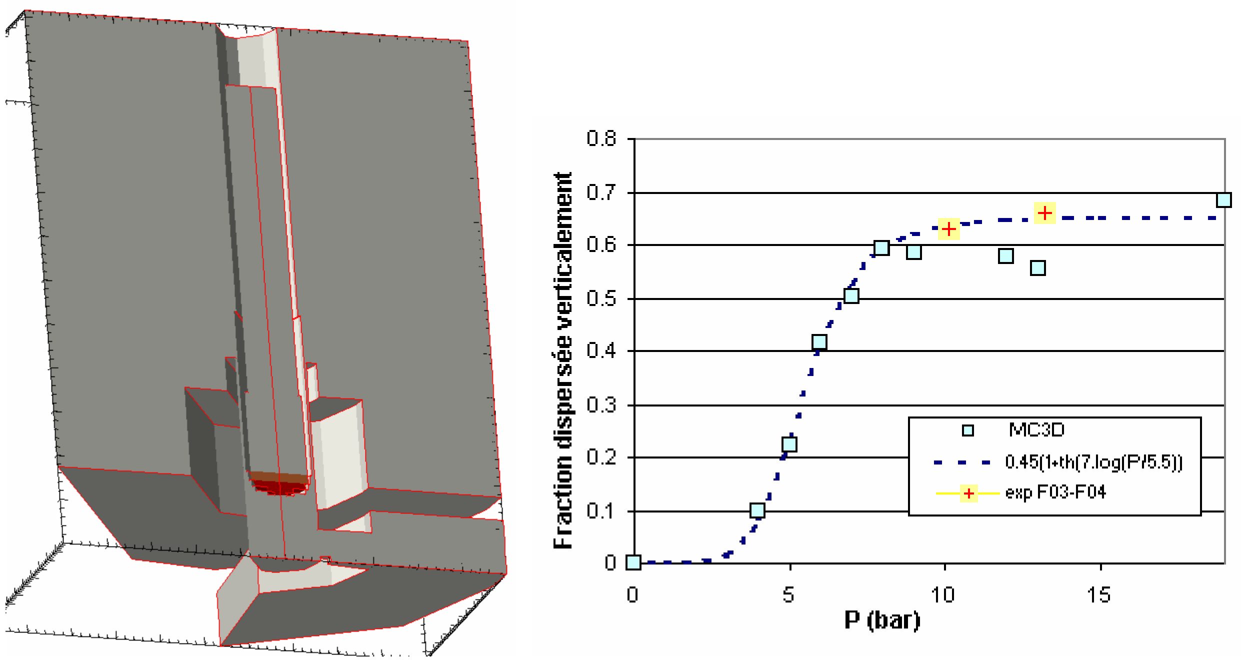


Figure 4.1-4: 3D geometry from MC3D analysis of P'4 DISCO tests, and assessment of the fraction dispersed towards the top of the pit (water as simulator and break diameter of 60 mm)

4.1.4 SUMMARY AND OUTLOOK

The experimental database shows that the consequences of DCH are essentially related to cavity geometry, and to the pathways between it and other containment areas. In particular, it is now widely accepted that the consequences of DCH are limited in reactors with no direct pathway between the cavity and the containment dome. In contrast, the situation is less clear for reactors which do have a direct pathway between the cavity and the containment; tests show that substantial fractions of corium may be dispersed into the containment in such cases if pressure in the reactor coolant system is elevated at the time of DCH.

Combustion of the hydrogen produced by oxidation as well as the hydrogen initially present appears to be the dominant phenomenon for containment pressurisation. It should be noted that DCH oxidation is difficult to extrapolate to reactor scenarios based on tests with simulant materials, which are much less reactive than corium components, especially zirconium. It is also difficult to draw conclusions as to how hydrogen combustion affects the loads generated by DCH. The effects of scale may also be important.

As to DCH modelling, it has proven complicated; the complexity and diversity of the phenomena involved do not lend themselves to simplified modelling. Multiphase simulation code appears to be the most promising option, but existing solutions are not yet able to satisfactorily predict outcomes on a reactor scale, due to problems modelling oxidation and combustion in particular.

The impact of water in the cavity has not really been characterised either; a better understanding is needed. For this point as well, simulation code such as MC3D may provide new insight.

In conclusion, the IRSN believes that current programmes of analysis should be pursued to bridge the gaps in knowledge. In particular, dedicated tests for studying hydrogen behaviour during the DCH phase are planned. At the same time, the IRSN recommends using numerical simulation codes (MC3D and AFDM) to compensate for the lack of data and to allow extrapolation to reactor scenarios.

4.2 HYDROGEN RISK AND MITIGATION STRATEGIES

4.2.1 HYDROGEN RISK — BACKGROUND

For the purposes of PWR severe accident studies, hydrogen risk is defined as the possibility of losing containment integrity in a reactor, or its safety systems, as a result of hydrogen combustion. The hydrogen is mainly produced by oxidation of cladding zirconium and fuel element structures during the core degradation phase, and oxidation of metals present in the corium pool or in the basemat during the molten corium-concrete interaction phase. This hydrogen is transferred into the containment (and transported therein) by convection loops arising essentially from condensation of steam released via the RCS break or during corium-concrete interaction. Depending on mixing in the containment atmosphere, the distribution of hydrogen is more or less homogeneous. If considerable hydrogen stratification exists, local concentrations of hydrogen may become substantial, exceeding the lower flammability limit for the gas mixture. The distribution and concentration of hydrogen in the containment building may also be modified by its spray systems. Spraying does homogenise the distribution of hydrogen in the containment and lead to “de-inertisation” of the mixture through the condensation of steam on water droplets. Systems such as recombiners and igniters can be installed in the containment building to avoid accumulation of hydrogen in part or all of the structure.

4.2.2 OVERALL PHENOMENOLOGY AND PROBLEM AREAS

Once released into the containment atmosphere during a severe accident, hydrogen is initially composed of air and steam. At that point in time, there is convective motion in the atmosphere as a result of the previously injected steam and its condensation on cold surfaces. The hydrogen will play a role in reinforcing the effects of gravity, due to its low density, and in altering condensation on the walls by increasing resistance to steam diffusion. Convection may thus be altered, and the key question is whether the entire contained volume is set in motion as a result. If so, since the characteristic mixing time for hydrogen and air is brief, atmospheric homogeneity is likely other than in zones close to the point of injection and the walls. If not, only part of the contained volume — in all likelihood, the upper part — undergoes mixing, and homogenisation with air first occurs there. If limited in size, this zone may contain a mixture relatively rich in hydrogen. The hydrogen will then undergo migration, for which the characteristic time is longer (several hours for the geometric dimensions postulated) towards the dead zones most likely situated in the lower regions. Here, the gas mixture will incorporate hydrogen, but hydrogen levels will never exceed those in the homogeneous zone.

The flammability of the containment gas mixture depends on its temperature, pressure and composition, as well as the ignition mode. However, in practice, the point representing the mixture's composition (hydrogen, air, steam) on the Shapiro diagram (see Figure 4.2-1) is used to determine whether the mixture is flammable. In this diagram, the zones of ignition and detonation are respectively delimited by the exterior and interior curves. These limits are dependent on temperature and pressure. The detonation limit is not an intrinsic characteristic of the gas mixture; it is only valid for the geometry in which it is obtained.

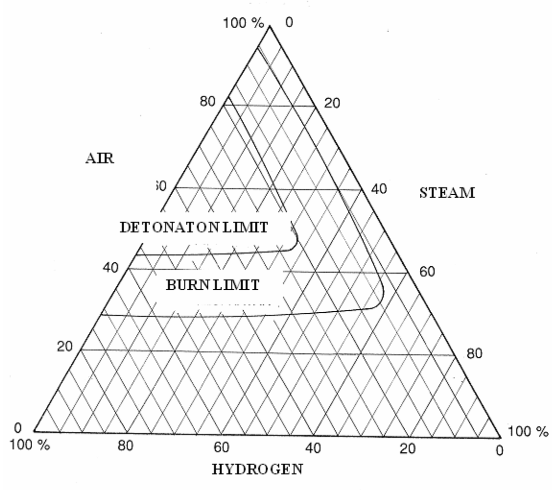


Figure 4.2-1: Shapiro diagram for hydrogen-air-steam mixtures

In a mixture known to be flammable, combustion may be initiated by an energy source of a few millijoules. Consequently, in the presence of electrical power sources, if there is actuator switching and hot points develop, it appears probable that ignition would occur rapidly once the combustion zone is entered. In contrast, a much more powerful energy source (at least 100 kJ) is required to trigger a stable detonation. This explains why direct detonation can be ruled out for practical purposes; the only mechanism considered likely to provoke detonation is flame acceleration and the deflagration-to-detonation transition. In fact, under the effect of hydrodynamic instabilities and turbulence (caused primarily by obstacles in the flame's path), an initially laminar deflagration (with a flame velocity around 1 m/s) may accelerate. Fast combustion regimes may also develop, involving rapid deflagration (a few hundred m/s), deflagration-to-detonation transition (DDT) and detonation (over 1000 m/s). These explosive phenomena pose the biggest threat to the mechanical integrity of the containment walls, because they can produce very large dynamic loads locally.

Based on their understanding of the mechanisms involved, researchers have developed prerequisite criteria, i.e. conditions required for the various combustion modes. There are two types of criteria:

- The "σ" criterion related to flame acceleration. The σ quantity is the mixture's expansion factor, a ratio of fresh and burnt gas densities at constant pressure. It is an intrinsic property of the mixture in question. The critical value σ^*, beyond which flame acceleration is possible, depends on initial gas temperature and flame stability. It is based on the results of numerous experiments at various scales and in various geometries.
- Similarly, prerequisite conditions have been defined for characterising the transition between deflagration and detonation regimes (DDT). They are based on comparing a characteristic dimension of the geometry with detonation cell size. This comparison, notated as λ, characterises mixture sensitivity.

These criteria were defined for homogeneous gas mixtures, and understanding the effect of concentration gradients is one of the objectives of the ENACCEF programme. Nonetheless, these criteria, together with the analysis of hydrogen distribution in the containment building and the facility's geometry, can be used to identify potentially dangerous situations (i.e. combustion possible; associated loads must be assessed). To apply these criteria, however, the codes used to calculate hydrogen distribution in the containment must be validated based on situations representative of severe accident conditions; this has been the aim of experimental programmes on hydrogen distribution in recent years.

4.2.3 EXPERIMENTAL PROGRAMMES

4.2.3.1 Hydrogen distribution

Hydrogen distribution in the containment building is controlled by the coupling of complex physical phenomena, such as:

- Flows at the point of gas injection and transport within the containment of chemical species such as hydrogen and steam
- Natural convection resulting from temperature differences between the atmosphere and the walls, and density differences between the various gas species present
- Steam condensation on the walls and internal structures of the containment building
- Mass and thermal stratification of gases
- Flow diffusion and turbulence
- Dynamic effect of spray droplets on flow; condensation on spray droplets

There are numerous analytical experiments focusing on these phenomena in isolation. Regarding condensation, for example, the experiments of Dehbi on natural convection, and those of Tagami, Uchida and Huhtiniemi on forced convection, have enabled the development of global models. However, the resulting correlations are more or less dependent on test conditions and geometry. There have been large-scale global experiments in addition to the analytical tests because the various phenomena governing hydrogen distribution are strongly coupled. A state-of-the-art report on containment thermal hydraulics and hydrogen distribution was completed in 1999 under the auspices of the OECD by a group of international experts (including IRSN experts) ([4.2_1]). It describes all the experiments (HEDL, HDR, BMC, NUPEC) conducted since the early 1980s. In most cases, these were large-scale global experiments with limited control of boundary conditions and limited instrumentation. They are not designed to validate multidimensional computer codes and can only be used to validate 0D codes.

To correct this shortcoming, better-equipped facilities have been constructed in recent years. These dedicated structures can be used to validate multidimensional, multicompartment computational tools. PANDA, ThAI, TOSQAN and MISTRA are examples of such facilities.

4.2.3.1.1 PANDA programme

The PANDA facility, located at the Paul Scherrer Institute in Switzerland, was initially designed for analysing containment thermal hydraulics in boiling-water reactors. It consists of four interconnected compartments with a total volume of 460 m^3. Recently, as part of the OECD SETH project, tests were conducted (most without condensation) on two compartments — a total volume of 180 m^3 — using instrumentation specially designed to capture structural differences in flow with a view to validating multidimensional codes ([4.2_13]). The test grid for this project addresses flows resulting from a lateral or central injection of steam and/or helium, plume interaction with the wall, the impact of an opening on gas distribution, etc. The PANDA facility makes it possible to study complex flows. However, the lack of control over wall temperature makes it difficult characterise condensation in detail.

A new experimental programme, SETH-II, is under development with support from the OECD. Involving both the PANDA and MISTRA facilities, this programme is aimed at collecting additional data on transient flows having various physical mechanisms and the potential to erode stratified environments.

4.2.3.1.2 ThAI programme

The ThAI facility, situated in Germany, is dedicated to analysing phenomena associated with hydrogen risk, iodine chemistry, and the transport and deposition of aerosols in PWR containments. It has a containment of 60 m^3 and an internal structure with multiple compartments. The outside walls of the containment are thermally insulated. With regard to hydrogen risk, the ThAI facility is designed for the study of hydrogen distribution and combustion, and the characterisation of catalytic recombiners. From 1998—2002, the facility was used to conduct thermal-hydraulic tests, one of which was used for ISP-47 ([4.2_3]). In that test, stratification of the gas mixture (air, helium, steam) was achieved by injecting helium and steam in the upper part of the structure. The lateral, low-momentum plume of steam then injected into the lower section was unable to bring about mixing in all areas of the compartmentalised atmosphere and thus did not homogenise the gas mixture.

A series of tests on distribution and characterisation of recombiners is currently in the design phase and may be carried out as an OECD project.

4.2.3.1.3 TOSQAN programme

The TOSQAN facility, designed and operated by the IRSN, consists of a cylindrical steel vessel with an internal volume of 7 m^3 (not including the sump). Instrumentation for the gas volume includes devices for measuring pressure, temperature, concentration of gas species (by mass spectrometry and spontaneous Raman scattering) and velocity (laser Doppler velocimetry, LDV, and particle image velocimetry, PIV). Characterisation of water droplets dispersed by the spray system focuses on size (interferometric laser imaging for droplet sizing), velocity (laser velocimetry) and temperature (global rainbow refractometry). The test programme is centred on the themes of condensation, spraying, interaction between the sump and the atmosphere within the containment, and wash out of aerosols ([4.2_11]).

Condensation tests, one of which was used for ISP-47, have been completed. They investigated steady-state conditions with and without helium. In the test used for ISP-47, helium added to the steam injection is initially distributed homogenously in the upper part of the structure (above the injection point). This is where the main convection loop is situated. An instability then develops (fluid heated by walls in lower areas), accelerating the usually slow phase of lower containment enrichment by creating movement throughout the atmosphere and thus a homogeneous mixture. Under steady-state conditions, the atmosphere is homogeneous.

With regard to spraying, the tests with off-centre spray nozzles have been completed and those with centred nozzles are underway. These tests are part of an international benchmarking effort organised by the SARNET network. Test campaigns addressing the atmosphere-sump interaction and aerosol wash out are currently in the design phase.

4.2.3.1.4 MISTRA programme

The main objective of the CEA's MISTRA programme is to study condensation on the walls and the water droplets (from spraying) in a geometry larger than that of TOSQAN and with the possibility of compartments ([4.2_12]). The MISTRA facility includes a containment of 100 m^3 (diameter: 4.25 m, height: 7 m), constructed of stainless steel, thermally isolated, and equipped with three internal condensing surfaces which are thermo-regulated. The instrumentation includes devices for measuring pressure, temperature, gas concentration and velocity (laser Doppler anemometry). These instruments can be used to qualify multicompartment codes, multidimensional codes and their coupling.

Condensation tests, one of which was used for ISP-47, were carried out for steady-state conditions with centred and off-centre injections, in open and compartmentalised configurations. With regard to the test used for ISP-47, in which helium was added to the main steam flow, the results are similar to those for the TOSQAN test. Initially there is a homogeneous atmosphere in the containment area affected by the main convection loop. Below the point of injection, helium concentration increases slowly until complete homogenisation of the containment is achieved (around three hours). The mixing observed throughout the atmosphere in TOSQAN does not occur because the lower containment is colder than the other areas (stable configuration). The impact of spraying on concentration distribution is currently being studied, using the same geometrical configurations as for the tests on condensation.

To develop computer code use in the reactor building, the IRSN and the CEA are currently analysing scale effects for TOSQAN and MISTRA. This analysis will verify whether models developed and validated from small-scale tests can be used to predict hydrogen distribution in reactor containments.

PANDA/MISTRA is a collaborative test programme on transient phenomena which homogenise initially stratified containments. It has been presented to the OECD.

4.2.3.2 Hydrogen combustion

As is the case for hydrogen distribution, several experimental programmes have focused on propagation of premixed hydrogen flames. The objective of these tests is twofold: 1) characterise the transition between slow and fast regimes, and between deflagration and detonation; and 2) produce a database to validate computer codes. There are two types of test campaigns:

- Analytical tests using spherical bombs, for characterising laminar flames and building a database for establishing prerequisite criteria for flame regimes.

- Dedicated tests for studying turbulent flames; can be used to validate computer codes and establish prerequisite criteria for characterising flame regimes.

As is the case for hydrogen distribution, a state-of-the-art report on flame acceleration and deflagration-to-detonation transition was produced in 2000 by a group of international experts (including IRSN experts) with support from the OECD ([4.2_6]). This report provides a description of the major experiments (BMC, NUPEC, VIEW, HTCF, FLAME, RUT, etc.) focusing on flame acceleration and detonation. The transition criteria for the various combustion regimes are based on tests conducted in these facilities. These criteria have been further developed through the European project HYCOM and the ENACCEF programme. This report also surveys the state of the art in combustion models.

4.2.3.2.1 RUT experimental programme

The RUT facility, located in Russia, has a total volume of 480 m^3 and a total length of 62 m. Initially designed for military applications, the facility consists of three parts: one channel which is completely rectilinear, a second channel which is curved at one end and a "canyon" or cavity in the intermediate area. All three zones have a rectangular cross-section and may eventually be blocked by obstacles. This geometry can be used to study monodirectional flame acceleration in the channels as well as more complex 3D effects or interactions in the canyon. To the best of our knowledge, it is the only facility of its size used to study turbulent hydrogen combustion and thus the only one subjected to pressure loads that can be transposed to reactor scenarios. There is an effort at RUT to use mixtures representative of severe accident conditions in the containment building. These mixtures contain hydrogen, air and sometimes a diluent (steam). The instruments used are well suited to validating CFD codes ([4.2_7]). The various test campaigns at RUT have investigated the following combustion regimes:

- Slow deflagration: Characterised by flame velocities below the speed of sound in fresh gases, and by pressure levels below the adiabatic isochoric complete combustion (AICC) pressure.
- Fast deflagration: Characterised by flame velocities close to the speed of sound in burnt gases, and by pressure levels above the AICC pressure; the term "choked flame" is used in this case.
- "Critical" regimes: In cases of deflagration-to-detonation transition (DDT), but where the resulting detonation does not spread or is not directly transmitted to the entire mixture.
- Stable detonation: Forms after DDT in one area of the facility; characterised by velocities and pressure peaks close to Chapman-Jouguet values (CJ) and propagated to the rest of the fuel mixture.

One of the objectives of the various programmes at the RUT facility has been to establish and validate the σ and λ criteria (defined in Section 4.2.2). The IRSN and the FZK have helped define and finance all tests performed at RUT.

4.2.3.2.2 HYCOM, a European programme

The European programme HYCOM was a continuation of the first RUT tests. It was aimed at studying flame acceleration in hydrogen-air mixtures and particularly at validating the σ criterion ([4.2_10]). The effect of burnt gas expansion (piston effect) and the impact of compartmentalisation were studied using the RUT facility; the impact of venting was studied using the DRIVER and TORPEDO facilities, which each consist of a cylindrical tube respectively 174 mm and 520 mm in diameter and 12.2 m and 12.4 m in length.

The IRSN actively participated in this programme with EDF's support. HYCOM also entailed an analytical component (full-scale benchmark exercise based on EPR geometry).

4.2.3.2.3 ENACCEF programme

The ENACCEF programme (flame acceleration) was carried out by the CNRS for the IRSN and, during its initial years, EDF. Its primary goal was to validate the σ criterion using tests conducted on a vertical structure representing an SG bunker opening into the dome ([4.2_2]). The bottom half of the ENACCEF facility consists of an acceleration tube. This tube is cylindrical with a diameter of 168.3 mm and a height of 3.2 m. It may be equipped with obstacles having various blockage ratios and forms, and may contain an obstacle simulating an SG with a volume of 11.12 litres. The upper half consists of an adjustable dome whose volume can vary from 780.9 litres to 957.8 litres. ENACCEF instrumentation includes photomultiplier and pressure sensors to measure the progression of the flame front and the pressure generated throughout the test facility. Gas sampling points are situated at various levels to measure the composition of the gas mixture in the facility. LDV and PIV techniques are also used to measure the flow velocity field before the flame passes. The ENACCEF facility is therefore highly instrumented and particularly well equipped to validate CFD codes. The effects of dilution, ignition point location as well as the bulk and non-uniformity of the mixture have been studied, allowing refinement of the σ criterion developed through the RUT and HYCOM programmes and providing data for the validation of CFD codes. The impact of water droplets on flame propagation is also being investigated at this time.

4.2.3.3 Mitigation systems

Catalytic recombiners serve to reduce hydrogen levels in the containment during an accident. They are usually constructed using catalytic materials (platinum and palladium on alumina) and housed in a metallic structure whose purpose is to optimise the circulation of gases in contact with the catalyser (bed of beads or row of vertical plates). The behaviour of recombiners in situations representative of severe accidents has been the object of several test programmes, conducted primarily by manufacturers (Siemens, AECL, etc.) ([4.2_5]).

The H2PAR programme, conducted by the IRSN with support from EDF, aimed mainly to verify that catalytic hydrogen recombiners continue to function in an atmosphere representative of severe accidents and containing several chemical compounds in aerosol form (risk of catalyser poisoning) ([4.2_4]). This programme also investigated the risk of recombiner-initiated combustion and determined thresholds for the specific recombiner studied. The impact of various parameters on recombination performance was also analysed, including geometric parameters (number of catalytic plates, height of stack location), physical parameters (molar fraction of hydrogen) and chemical parameters (several catalytic plates replaced by neutral chemical plates) ([4.2_14], [4.2_15]).

The KALIH2 test programme, conducted by the CEA with support from EDF, had similar objectives. It evaluated the effects of the following on recombiner performance: humidity, smoke from cable fires, and carbon monoxide ([4.2_8], [4.2_9]). Unlike H2PAR, KALIH2 made it possible to study the impact of overpressure on recombiner efficiency.

4.2.4 MODELLING AND SIMULATION CODES

The codes used to predict hydrogen distribution in the containment building have traditionally been based on a multicompartment approach. Amongst others, CONTAIN, MAAP, GOTHIC, MELCOR, COCOSYS, the CPA module of the ASTEC code and the multicompartment module of the TONUS code are all based on this approach. These codes have demonstrated their capacity to calculate hydrogen distribution, as well as their limits, in small- and large-scale experiments, with or without the use of spray system. They do however require a special dataset to be able to predict all the potential flows in a reactor, particularly in volumes where concentration gradients may exist (stratification, jets, etc.). Codes based on a multidimensional approach, like the multidimensional module of TONUS or the GASFLOW code developed by the FZK, correct this shortcoming. However, use of these codes is limited by the geometrical complexity of containments and by the costs involved, which may be prohibitive due to current computational tools. As a result, coupling the two approaches, as the TONUS code does, is currently the best compromise between precision and cost.

The recent computer code exercises (ECORA and ISP-47), based on the experimental results of the four programmes mentioned above, have led to the following conclusions.

The ECORA exercise (gas injection transient without condensation), which only used CFD tools, showed that the main limitation was the computing resources required for analysing large-scale transients. The prohibitive costs are an obstacle to best practices (mesh convergence and sensitivity, impact of time step and time order of the numerical scheme, etc.). However, the models used in ECORA made it possible to accurately predict steam transport between the compartments, which was one of the key points of this exercise.

Regarding ISP-47, this exercise focused on multicompartment, multidimensional tools. Moreover, since several institutes use the same tools, there was better assessment of the user effect. The final conclusions are in the process of being drafted, but the following points can be mentioned:

- The CFD tools, rarely used for this type of exercise in the past, did not show any advantages compared with multicompartment tools. For the TOSQAN and MISTRA tests used in ISP-47, this was due to the relatively simple flow structures.
- The results obtained with the multicompartment tools showed strong user-dependent variability. Best practices should thus be implemented (this recommendation is valid for CFD codes that use correlations [condensation, recombiners]).
- Concerning simulation of stratification, there was only one contribution in which a multicompartment tool successfully predicted that stratification would be maintained in the ThAI test during steam injection into the lower area, and this tool belonged to the user who designed

the experiment. Furthermore, questions were raised as to the exact specifications of both steam injection and the lower containment structures, which might play a role in steam jet deflection.

- The “blind” exercises, important for assessing the predictive aspect of the codes, produced a wide array of results.
- Modelling of wall condensation requires complementary studies, especially regarding the effect of helium on spatial distribution.
- Scale effects remain an open issue.
- Finally, the need for experimental data adapted to multidimensional code validation was apparent for transient flows. The PANDA-MISTRA SETH-II project was developed with this goal in mind.

Additionally, loads generated by hydrogen combustion can be calculated using computer codes based on multicompartment or multidimensional approaches. In general, multicompartment codes are limited to slow flame analysis for which pressure loads can be considered as static. Calculation of dynamic pressure loads usually requires using CFD codes based on multidimensional approaches. For example, the HYCOM project produced very complete results with regard to hydrogen combustion in reactor containments and the modelling of this phenomenon. In particular, this project demonstrated the following:

- Global values, such as maximal pressure, are relatively well reproduced by CFD codes and multicompartment codes. However, CFD codes give better results for fast flames, whereas multicompartment codes are better suited to slow flames.
- Differences exist between the various codes for “dynamic” values, such as flame velocity or pressure increases.
- Certain phenomena observed experimentally, such as flame quenching, are not correctly reproduced by the computer codes.
- Modelling heat loss is important and must be enhanced.

But above all, the HYCOM project highlighted the difficulties of modelling hydrogen combustion in cases of non-homogeneous mixtures, especially when this non-uniformity is accompanied by a change in the combustion regime. Although these conditions are the closest to real scenarios, they have not yet been adequately modelled and require additional experimental data suited to code validation. This is one of the objectives of the ENACCEF programme.

4.2.5 SUMMARY AND OUTLOOK

Research and development on hydrogen risk have produced a number of outcomes reinforcing the decision to install specific mitigation systems in all French nuclear power plants. Studies of representative accident sequences indicate that despite the installation of recombiners, it is difficult to prevent, at all times and locations, the formation of a combustible mixture potentially leading to local flame acceleration. The RUT and HYCOM programmes have produced criteria which, if respected, would allow the risks of flame acceleration and DDT to be eliminated. The ENACCEF programme outcomes will further develop these criteria.

In order to better quantify the possible consequences of hydrogen combustion, complementary research and development must be carried out in the near future. The necessary initiatives are as follows:

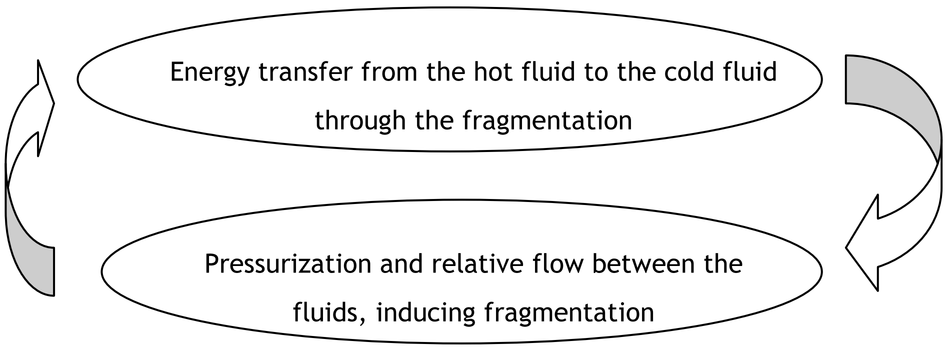
- For hydrogen distribution, the study of transient flow regimes and those with stratification. This area can be addressed by existing programmes, especially TOSQAN, PANDA and MISTRA.
- For combustion, studies of how water droplets affect hydrogen flame acceleration. This theme will be addressed by the ENACCEF programme.
- For recombiners, studies of how recombiner location affects the rate of recombination as well as flame ignition by the recombiners. These two themes are addressed by SARNET activities; the first is the focus of a numerical study and the second, an experimental study based on REKO tests.
- For hydrogen distribution in the computer codes, as demonstrated by the preliminary conclusions of ISP-47, further development is needed to better represent condensation in the presence of incondensable gases and stratification, and to describe the mechanisms of destratification. The spray system benchmarking organised by the IRSN has demonstrated the limitations of existing codes.
- For computer codes and hydrogen combustion, as demonstrated by the HYCOM outcomes, enhanced and validated combustion models are needed to simulate flame propagation in non-uniform environments, especially those with hydrogen gradients enabling transition between flame regimes.

In conclusion, R&D efforts to date have already significantly enhanced understanding of the phenomena governing the distribution of gas mixtures and their potential combustion. In particular, establishing criteria based on experimental data has led to identification of potentially high-risk situations. Regarding computational tools, although they have clearly reached a degree of maturity, their predictive capacity must be reinforced by enhanced modelling (multicompartment calculations, for which the results depend heavily on user expertise, for example) and/or overcoming computing limitations, which currently make it impossible to respect all the rules of proper use (case of multidimensional codes).

4.3 STEAM EXPLOSIONS

4.3.1 DEFINITION, OVERALL PHENOMENOLOGY

The global phenomenology of steam explosions has been relatively well understood since the 1970s. When there is contact between two fluids, and one (the fuel or "corium" from core meltdown; henceforth these two terms will be used interchangeably) is at a temperature higher than the boiling point of the other (the coolant), an explosive interaction may be triggered. This phenomenon involves the unstable coupling of the two mechanisms illustrated below:



The ultra fine fragmentation of corium (sub-millimetric fragments) induces a transfer of energy to the coolant and its vaporization, with a characteristic time smaller than the pressure relief one. This in turn leads to high overpressure, followed by an explosive expansion which may damage surrounding structures (pressures up to 1000 bar have been measured in the alumina KROTOS tests [4.3_1]).

The prerequisite condition for a steam explosion is simple contact between two fluids, but the most high-energy situations are those in which the two fluids undergo coarse mixing prior to fine fragmentation (hence the term "premixing"). In PWR reactor systems, such mixtures may form after core meltdown, as corium relocates to the lower head (in-vessel explosion), and potentially in the flooded cavity following vessel failure (ex-vessel explosion).

These conditions are not sufficient, and in many cases, systems in which fluids are in contact or undergo mixing do not produce explosions. In these cases, there is only a coarse fragmentation (millimetric or centimetric fragments) with a relatively slow energy transfer between corium and the coolant, resulting in slow pressurisation of the system (TMI-2 scenario). In order for an explosion to occur, there must be an "internal" triggering event (producing a so-called spontaneous explosion) or an "external" event (shock) initiating fine fragmentation somewhere in the premixture, which is then propagated throughout the rest of it. Such spontaneous or artificially triggered explosions have been produced experimentally with the molten materials comprising PWR corium (Zr, ZrO_2, Zr-ZrO_2, UO_2-ZrO_2, UO_2-ZrO_2-Zr).

In its most extreme form, the phenomenon is analogous to that of detonation; there is shockwave propagation and the chemical processes are replaced by heat and mass transfer processes. But the analogy is limited, and more or less realistic approximations are needed to construct analytical models of detonation, which have very little potential for practical application. This explains the current use of complex multiphase, multidimensional models to handle this problem.

The recent OECD SERENA programme (Steam Explosion Resolution for Nuclear Applications, 2001—2005), brought together the leading steam explosion specialists. They evaluated current understanding of the phenomenon and assessed computational capacities of the main dedicated codes solutions ([4.3_4]).

4.3.2 PHYSICAL PHENOMENA

In a PWR severe accident in which corium comes into contact with water in the lower head or the cavity, the phenomenon appears to amount to a process of dispersion and fragmentation in two steps, as shown in Figure 4.3-1. The first step involves premixing and creates the initial conditions of the actual explosion (second step), which is characterised by three key mechanisms: fine fuel fragmentation, energy transfer and the resulting pressurisation. The first mechanism is very dependent on premixing conditions at the time of explosion, namely mixture composition, distribution of the various phases (corium, liquid, gas) and the corium interface, including its temperature and its state of solidification. Since it determines the initial conditions of the explosion, this step must be described in detail.

Between the premixing and explosion steps, there is fine fragmentation, commonly referred to as the trigger. These three phenomena are discussed in detail below.

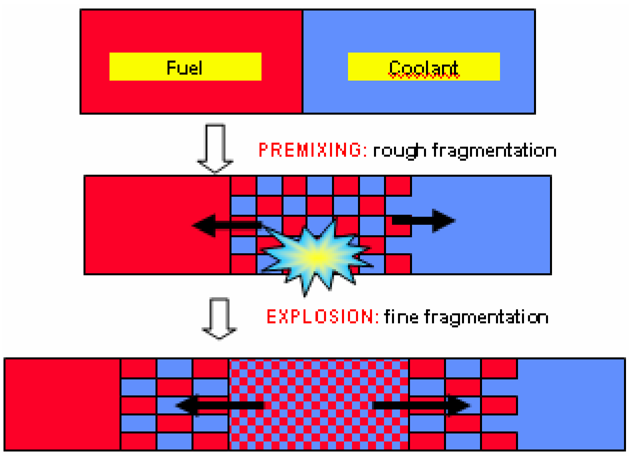


Figure 4.3-1: Schematic diagram of the phases in a steam explosion

4.3.2.1 Premixing

The importance of premixing, which determines the initial conditions of the explosion, was clearly demonstrated in the KROTOS experiments. Premixtures observed for alumina were very different than those observed for corium (see Figure 4.3-2), producing widely varying explosion intensities (10 times more energy released with alumina than with corium) ([4.3_1]). Since these tests, studies have mostly focused on this phase. It is also likely that the capacity for spontaneous triggering depends on the premixing configuration. In all tests conducted to date, the data on the premixing of materials with high melting points (> 2000 °C) have been mostly qualitative. This data is not sufficient to explain and qualify the differences in behaviour observed. The second phase of the SERENA programme should provide more detailed information (see section 4.3.3.2.2).

For analytical purposes, the premixture is assessed using multidimensional, multiphase thermal-hydraulic codes (see Section 4.3.3). There are so many dynamic and thermal interactions in this phase that it cannot be characterised in a simple manner.

Considerable R&D efforts are focusing on two key points, namely corium fragmentation and generation of a void fraction, as presented below.

Figure 4.3-2: At left, alumina/water premixture (KROTOS-57). At right, corium/water premixture (KROTOS-58). Video camera, 24 frames/s. 10x20 cm window. Taken from [4.3_1].

4.3.2.1.1 Corium fragmentation

There are two stages of premixing; primary fragmentation, involving the continuous phase (typically a corium jet), produces a first generation of droplets which will then undergo a secondary fragmentation. This second stage continues until the size of the droplets is such that the instabilities causing their division are sufficiently reduced.

Numerous studies have concentrated on primary fragmentation, including two relatively recent publications in France ([4.3_2], [4.3_3]). The models described therein focus on fine fragmentation (atomisation) acting directly on the jets. It is known that the jets may be fragmented by other mechanisms, implying larger-scale instabilities and broader dispersion of corium as a result (Figure 4.3-3). These mechanisms are believed to cause the behaviour observed during tests with alumina in the KROTOS facility ([4.3_1]), where the fragments filled the entire cross-section of the experimental tube (see Figure 4.3-2). Secondary fragmentation was studied extensively into the 1980s, bringing to light certain trends and fundamental characteristics. Nonetheless, applying this knowledge to premixing only allows qualitative characterisation of the phenomenon. A complementary parametric approach therefore seems reasonable.

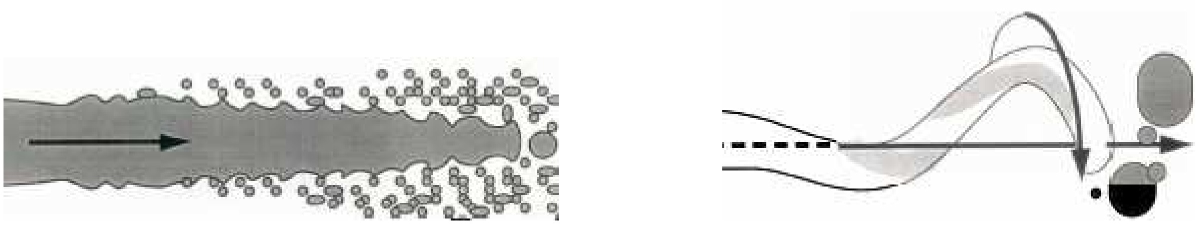


Figure 4.3-3: Instability/fragmentation via atomisation (left) and on a larger scale (right)

4.3.2.1.2 Void fraction

There is currently substantial uncertainty concerning the precise effect of the void fraction (gas phase fraction) on the explosion. However, beyond a certain threshold, there is less of a chance that an explosion will be triggered and escalate. Moreover, the void fraction must be correctly assessed to understand the overall significance of the premixing calculations. This issue is much more complex than it seems, mainly because of the extreme temperature conditions. The steam generation processes of film boiling and radiation are still poorly understood. The flow configurations used, based on studies of isothermal two-phase flows, are somewhat arbitrary. The presence of incondensable gas from oxidation, or eventually the simple dissociation of steam, adds to the difficulty.

This results in a certain degree of disparity across the models, and this disparity is largely blamed for the uneven computational results obtained during phase 1 of SERENA. In the absence of detailed experimental data, the validity of the various models cannot be established with sufficient certainty. Correctly evaluating the void fractions and their distribution is thus a critical challenge and one of the major objectives of the SERENA phase 2 proposal, as described in Section 4.3.3.2.2.

4.3.2.2 Triggering

The phase during which the explosion is triggered is undoubtedly the most complicated to study. As there is currently no reliable model for predicting the time and location of this phenomenon, quantification of its probability based on physical parameters is also unavailable. Existing knowledge is based mainly on experimental observations. For corium and similar fuels, it has been experimentally observed that spontaneous explosions occur most likely when the fuel makes contact with the base of the test section). However, there is nothing to indicate that a later explosion could not take place.

Theoretically, it is known that a droplet of hot fuel may explode as the result of a minor disturbance, on the order of a few bars, but despite extensive theoretical researches, this process, known as “thermal fragmentation”, remains quite poorly understood. It is believed that the disturbance causes destabilisation of the steam film around the droplet. Through poorly defined processes involving corium-coolant contact, this destabilisation appears to produce local pressurisation which further destabilises the fuel droplet. This process likely occurs during the triggering and escalation phase, but its real importance is not yet known.

Consequently, it is not currently possible to predict whether an explosion will occur in a given accident situation, beyond certain obvious qualitative aspects. Researchers must therefore assume that an explosion is always possible for a given scenario and investigate its consequences. This leads to systematically incorporating a trigger to determine the steam explosion response of a given system. In studies reinforcing safety analysis, researchers currently postulate that an explosion occurs at a given time and location by imposing destabilising conditions locally. However, the triggering event imposed through a disturbance does not necessarily lead to a strong interaction if the conditions are not satisfied (for the model). In theory, this minimises the need for in-depth understanding of what causes triggering, provided that escalation — the system's response to minor disturbances — is correctly described.

4.3.2.3 Explosion

The explosion phase results from very intense heat transfer between corium and the coolant. For violent explosions, such as those obtained in the KROTOS facility using alumina (500—1000 bar), the process may be described as follows:

- Detonation with intense heat transfer (isochoric configuration)
- General expansion in the mixing zone

Because of the obvious equipment limitations, experimental studies on the fine mechanisms of explosion are extremely complicated. Researchers only have characteristic values to describe the main physical phenomena.

But paradoxically, this phase is simpler to model from a numerical perspective than the premixing phase, provided that there are adequate approximations for fine fragmentation as well as heat and mass transfers, which produce the pressure increase. For one thing, these two phenomena are clearly dominant mechanisms; for another, several aspects can be simplified or even ignored because of the timescale (a few milliseconds). However, further efforts are needed to better understand the various processes themselves.

4.3.2.3.1 Fine fragmentation

Fine fragmentation of a fuel droplet subjected to a shock wave is illustrated in Figure 4.3-4.

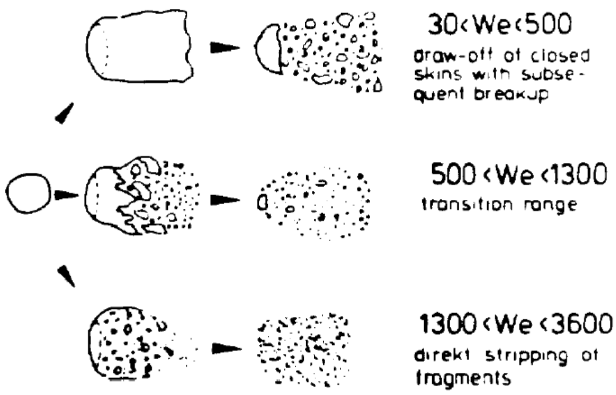


Figure 4.3-4: Types of fragmentation based on Weber number ($\rho V^2 D/\sigma$) as observed by Bürger et al. ([4.3_5])

During this phenomenon, characteristic droplet size is reduced by one or two orders of magnitude in around one millisecond. Models with varying levels of detail exist, but so far they have not led to a consensus as to the phenomenology itself. The main challenge lies in predicting fragment size, which has a strong impact on explosion intensity. As of this writing, researchers tend to use a highly parametric approach.

4.3.2.3.2 Pressurisation mechanisms

The first stationary models of steam explosion ([4.3_6]) were based on the hypothesis of a instantaneous equilibrium between the fragments produced and the coolant. In the models, the heating of the water produces the pressurisation during the initial phases. This became known as the "micro-interaction" approach through the work of Theofanous ([4.3_7]). Physically, however, it is relatively clear that the fragments are surrounded by a thin steam film. A second series of models are thus based on the hypothesis proposed by Berthoud ([4.3_8]), that pressurisation is obtained via the vaporisation accompanying fragment-related heat transfers ("disequilibrium" model). The latter model, although more mechanistic overall, does require understanding of the mass transfers associated with the heat exchange between the fragments (100 µm on average) and the coolant. It is important to remember that these transfers occur during a pressure transient of a few hundred bar, lasting a few milliseconds. At present, most codes are based on one concept or the other.

4.3.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

4.3.3.1 Past experimental programmes

Table 4.3-1 lists the main programmes that have focused on premixing or explosion, using corium or simulant jets weighing several kilograms. The data from these programmes has served as a base for developing and qualifying computer models. The reference database is that of the FARO experiments. To the table entries must be added those programmes whose results have not been extensively used, either because the phenomena studied were not modelled in the codes, or because the results were not available until very recently. The unlisted programmes include ANL's ZREX tests ([4.3_9]), which focused on the increased explosion intensity caused by concurrent oxidation of zirconium in the Zr-ZrO_2 and Zr-steel mixtures. There are also the premixing tests conducted at JAERI ([4.3_10]) using 10 litres of steel and Al_2O_3-ZrO_2 mixtures, for which results were not made available until 2005.

In the high-temperature domain, there are also important analytical tests such as TREPAM (CEA/IRSN), which enabled characterisation of fragment-related heat transfers in highly representative conditions (pressures up to 240 bar, velocities up to 46 m/s, temperatures on the order of 2500 K). There are also the BILLEAU tests (CEA/IRSN [4.3_11]) and the QUEOS tests (FZK [4.3_12]), which used solid spheres at temperatures up to 2200 °C to study premixing.

In 1999, the FARO/KROTOS programme at JRC-Ispra was terminated, despite numerous uncertainties concerning the behaviour of corium-water systems under steam explosion conditions. This prompted the CEA to purchase the KROTOS facility from the European Commission. It also played a role in the launch of the international programme SERENA, on the heels of the OECD SESAR/FAP report which called for continued investigation of corium-water interaction ([4.3_13]).

Table 4.3-1: Experimental programmes on steam explosion using a fuel jet configuration

Programme	Laboratory	Type of test	Material	Quantity	Main parameters
CCM [4.3_14]	ANL (United States)	Premixing	UO_2-ZrO_2-steel mixtures at 2800°C	2-12 kg	- Water mass - Jet diam. (2-5 cm) - Water temp. - Jet velocity
FARO [4.3_15] [4.3_16]	CCR Ispra (European Commission)	Premixing Explosion	UO_2-ZrO_2 at 2800°C	18-178 kg	- Jet diam. (5 and 10 cm) - Water level (1-2 m) - Water temp. - Pressure (2-50 bar)
KROTOS [4.3_1]	CCR Ispra (European Commission)	Premixing Explosion	Sn at 1000°C Al_2O_3 at 2300-2800°C UO_2-ZrO_2 at 2800°C	1 litre	- Molten material - Material temp. - Water temp.
PREMIX [4.3_17]	FZ-Karlsruhe (Germany)	Premixing	Al_2O_3 at 2300°C	16-60 kg	- Pressure (1-5 bar) - Al_2O_3 mass - Al_2O_3 ejection velocity
ECO [4.3_18]	FZ-Karlsruhe (Germany)	Explosion	Al_2O_3 at 2300°C	6-18 kg	- Pressure (3-19 bar) - Al_2O_3 mass - Al_2O_3 ejection velocity
MIRA [4.3_19]	KTH (Sweden)	Premixing	Various binary oxides at 1150-1400°C	5-34 kg	- Oxide temp. - Water temp. - Water level (1 and 2 m)

4.3.3.2 Experimental programmes currently underway

4.3.3.2.1 National programmes

Aside from the analytical programmes DROPS (IKE [4.3_22]) and MISTEE (KTH [4.3_23]), which use simulant materials at relatively low temperatures to improve physical models, there are currently two experimental facilities where global reactor-oriented tests are carried out using corium: KROTOS, transferred to Cadarache, and TROI in South Korea ([4.3_24]).

4.3.3.2.2 SERENA, an international programme

Phase 1 of SERENA was purely analytical. Its goal was to evaluate the capacity of computer codes to analyse reactor scenarios and provide credible results. Disparities in the findings indicate the difficulty of reducing uncertainty to a level allowing assessment of dynamic loads. However, since in-vessel calculation results were below the limits for lower head rupture (data on ageing not accounted for), priority was given to the ex-vessel scenario for which the calculations gave dynamic load values liable to challenge cavity walls and the containment building. From a phenomenological perspective, there are no fundamental differences between in-vessel and ex-vessel scenarios. The main differences are related to the nature of the corium, its mode of transfer in water, the system pressure and the water temperature. These conditions must therefore be addressed first and foremost.

It was established that the primary uncertainties and disparities were due to a lack of detailed data on the premixing zone and the general behaviour of corium compared with simulants such as alumina (material effects). Until now, the lower energy level of corium compared with alumina was only demonstrated in a limited number of compositions, which were not necessarily representative of ex-vessel corium.

A second phase of the SERENA programme, aimed at filling the data gaps on the premixing phase as well as material and geometry effects, has been proposed to the CSNI by the CEA, the IRSN and two of their South Korean counterparts, the KAERI and the KINS. It is based on the KROTOS and TROI installations (Figure 4.3-5):

- KROTOS is essentially one-dimensional and can accommodate up to 5 kg of fuel. The tests conducted at Ispra are of prime importance for code qualification and global understanding of the phenomena. In particular, the new installation will include a radioscopy device allowing researchers to obtain detailed data on premixing.
- TROI can inject up to 30 kg of corium into a larger-sized tank. The range of possibilities for corium injection will allow researchers to determine the impact of the various configurations on the basic explosion intensities established in KROTOS, and will allow verification of the codes' “extrapolability” to complex situations.

The experimental programme will include a rigorous analytical component structured in the same way as for phase 1. The SERENA phase 2 proposal is also firmly reactor-oriented. Calling for various materials typical of accident situations, its primary objective is to:

- Provide data on the distribution of premixture components for code validation.
- Study material effects, particularly the low-energy behaviour of corium observed thus far, to determine if this behaviour can be generalised.
- Determine whether the codes can address unconventional geometries.
- Establish “envelope” conditions for pressure loads.

The technical proposal is in the final stages and a draft agreement will soon be proposed to the members of the CSNI.

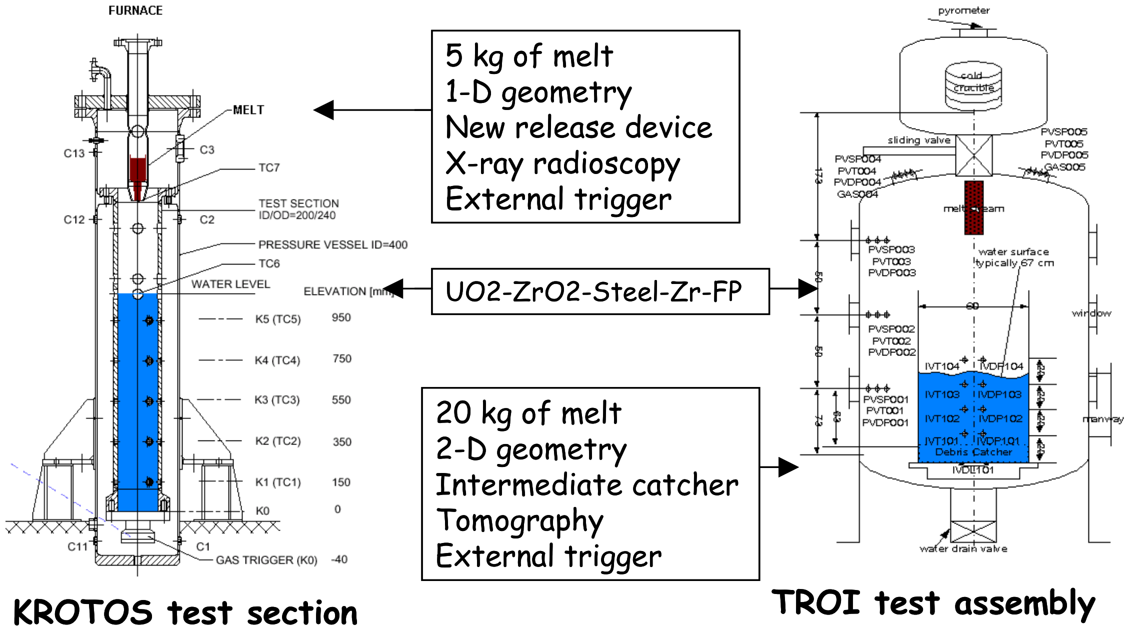


Figure 4.3-5: KROTOS (CEA Cadarache) and TROI (KAERI, South Korea) experimental installations. Key characteristics and instrumentation

4.3.3.3 Codes

The IRSN and the CEA continue to develop the MC3D code, particularly for the numerical simulation of the premixing and explosion phases ([4.3_20], [4.3_21]). EDF discontinued its support in 2002. The SERENA programme has positioned MC3D amongst a small group of advanced tools leading the way in steam explosion simulation (with the likes of PM-ALPHA [USA], IKEMIX/IDEMO [Germany, GRS/IKE] and JASMINE [Japan]) ([4.3_21]).

Codes for analysing steam explosions must be able to handle the numerous interactions between phases. This implies very complex modelling with detailed numerical schemes, particularly to ensure robustness. Furthermore, certain codes, such as MC3D, are used for both assessing safety in nuclear installations (practical approach) and enhancing understanding of phenomenology (R&D approach). This duality implies constraints which are often difficult to manage.

In the PREMELANGE (premixing) application of MC3D, there are three fields for the fuel:

- A continuous field (jet), modelled using a volume-tracking method (VOF-PLIC, see Figure 4.3-6).
- A droplet field modelled using an Eulerian method. The droplets are produced by fragmentation of the continuous field.
- An optional fine fragment field.

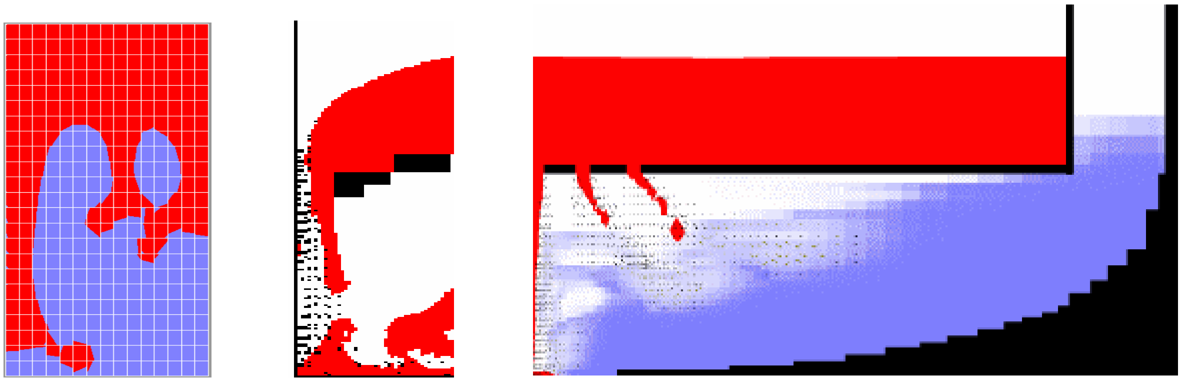


Figure 4.3-6: Examples of the volume-tracking method used in MC3D (red: continuous fuel, black points: droplets). From left to right: Rayleigh-Taylor instabilities, ejection of fuel from vessel under high-pressure conditions (transition towards two-phase flow at break), analysis of in-vessel interaction with several jets.

There are fewer differences between the codes regarding explosion modelling, and most of them use two fields for the fuel, namely a droplet field and a fragment field.

Some codes, such as PM-ALPHA and IDEMO, use what is known as a “micro-interaction” approach, in which the heating of a fraction of the water produces the pressurisation. The MC3D and JASMINE models assume that pressurisation is due to vaporisation around the fragments generated by the explosion. The consequences of these modelling choices are currently being assessed.

Although the predictive capacities of these tools are still somewhat limited at this time, they have significantly enhanced overall understanding of the phenomena and the multiple interactions. With regard to safety studies that are relatively complete, MC3D has only been operational for about three years. The code remains difficult to use, requiring a high level of expertise. However, the multiple partnerships in place around this code should accelerate maturity.

4.3.4 SUMMARY AND OUTLOOK

The research outcomes on steam explosion may appear modest in light of the problems left to solve. In any case, it is clear that adequate modelling can only be achieved through the development of sophisticated simulation tools. Considering the progress made in this area in the last decade, particularly in Europe and Japan, it is easier to evaluate the work yet to be done. Moreover, there is currently an international pooling of efforts which is particularly evident in the SERENA programme. This sharing of resources is essential and creates common ground between visions which are sometimes divergent. Encompassing an ambitious analytical programme, the phase 2 proposal for SERENA is aligned with this approach. With cutting-edge, innovative instrumentation and controlled experimental conditions, the groundwork is in place to considerably build knowledge and extend the database for qualifying code.

Regarding the impact on safety, there has been renewed international interest in a more detailed assessment of the ex-vessel situation, as much for PWRs (excluding EPRs) as for BWRs, which is reflected in the SERENA-2 proposal. This is primarily due to questions about the mechanical strength of concrete structures, considering the loads produced by steam explosions. This problem is all the more important given that preventative reflooding of the cavity is one of the mitigating actions widely implemented or postulated in the event of a severe accident in an operational reactor (VVERs, PWRs and BWRs).

Concerning in-vessel steam explosions (alpha mode), there is agreement in the international community that the consequences of this phenomenon are of a residual nature. In the short term, the IRSN plans to verify these conclusions against recent R&D work (MC3D and Europlexus calculations).

CHAPTER 4 TABLES

CHAPTER 4 FIGURES

CHAPTER 4 REFERENCES

[4.1_1] Cénérino G., Chevallier C., "Echauffement direct de l'enceinte – Etat de l'art", IPSN/DPEA/SEAC/98-064, 1998 – Non-public reference.

[4.1_2] Binder J.L., McUmber L.M., Spencer B.W., "Direct Containment Heating Integral Effects Tests at 1/40 Scale in Zion Nuclear Power Plant Geometry", NUREG/CR-6168, ANL-94/18, 1994.

[4.1_3] Blanchat T.K., Pilch M.M., Allen M.D., "Experiments to Investigate Direct Containment Heating Phenomena with Scaled Models of the Calvert Cliffs Nuclear Power Plant", NUREG/CR-6469, SAND96-2289, 1997.

[4.1_4] Chaumont B., "Debris dispersal experiment for the French reactor cavity (PWR 900 MWe NPP) during a pressure melt ejection accident", NT IRSN/DPEA/SEAC/03-018, 2003 – Non-public reference.

[4.1_5] Meyer L., Albrecht G., Kirstahler M., Schwall M., Wachter E., Wörner G., "Melt Dispersion and Direct Containment Heating (DCH) Experiments in the DISCO-H Test Facility", FZKA 6988, 2004.

[4.1_6] Meyer L., Gargallo M., Kirstahler M., Schwall M., Wachter E., Wörner G., "Low Pressure Corium Dispersion Experiments in the DISCO Test Facility with Cold Simulant Fluids", FZKA 6591, 2002.

[4.1_7] Meyer L., Plassart D., "DCH test campaign in the modified DISCO-C facility using water as corium simulant – Summary report", NT IRSN/DSR/SAGR/05-059, 2005 – Non-public reference.

[4.1_8] Cranga M., Giordano P., Passalacqua R., Caroli C., Walle L., "ASTEC RUPUICUV Code – Version 1.2 – Ex-vessel corium discharge and corium entrainment to containment", ASTEC-V1/DOC/03-19, 2003 – Non-public reference.

[4.1_9] Murata K.K., Williams D.C., Tills J., Griffith R.O., Gido R.G., Tagios L.G., Davis F.J., Martinez G.M., Washington K.E., "Code Manual for CONTAIN 2.0: A Computer Code for Nuclear Reactor Containment Analysis", NUREG/CR-6533, SAND97-1735, 1997.

[4.1_10] Wilhelm D., "Analysis of Direct Containment Heating Experiments with Thermite in P'4 Geometry", Note Technique DSR/SAGR/05-58, 2005 – Non-public reference.

[4.2_1] D. KARWAT & al, "State-of-the-Art Report on containment thermalhydraulics and Hydrogen Distribution", rapport NEA/CSNI/R(99)16.

[4.2_2] C. E. PAILLARD & all, "Note technique sur l'enceinte d'accélération de flamme", Rapport CNRS- LCSR October 2001 – Non-public reference.

[4.2_3] H-J. ALLELEIN & al, "Proposal for an international standard problem exercise on containment thermal hydraulics", March 2001 – Non-public reference.

[4.2_4] D. LETEINTURIER & al, "Essais H2PAR : période mi-98 à fin 2000 synthèse des essais conclusions du programme", Rapport IRSN/DPEA/DIR/02/01 – Non-public reference.

[4.2_5] J. LOESEL-SITAR & al, "Environmental Qualification of Hydrogen Recombiners, Test Report", Rapport AECL 00-68460-TR-001 – Non-public reference.

[4.2_6] W. BREITUNG & al, "OECD State-of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation Transition In Nuclear Safety", OECD/CSNI, October 1999.

[4.2_7] S. DOROFEEV & al , "Large scale experiments for validation of hydrogen combustion models and criteria", article présenté au Jahrestatung Kerntechnick, Stuttgart, May 2002.

[4.2_8] O. BRAILLARD & al, "generic model tests of passive autocatalytic recombiners (PARs) for combustible Gas control in nuclear power plants: Volume 2- test data for NIS PARs", note CEA 96/023A – Non-public reference.

[4.2_9] O. BRAILLARD & al, "generic model tests of passive autocatalytic recombiners (PARs) for combustible Gas control in nuclear power plants", Volume 3- test data for SIEMENS PARs, note CEA 96/023B – Non-public reference.

[4.2_10] W. Breitung & all, "Integral large scale experiments on hydrogen combustion for severe accident code validation HYCOM", Nuclear Engineering and Design, volume 235, Issues 2-4, pp 253-270.

[4.2_11] P. ZAVALETA & all, "Expérience TOSQAN Description du programme expérimental 1ère partie : condensation en paroi", rapport d'étude SERAC/LPMC/99-03 – Non-public reference.

[4.2_12] D. Leteinturier & al, "rapport de convergence des programmes MISTRA de la DRN et TOSQAN de l'IPSN", Note technique DPEA/99-04 – Non-public reference.

[4.2_13] O.AUBAN & al, "OECD/SETH Large scale investigation of gas mixing and stratification, PANDA Test Facility Description and geometrical data", rapport PSI TM-42-05-06 – Non-public reference.

[4.2_14] P. RONGIER, T. BONHOMME, C. PEREZ, "1ère Synthèse H2PAR – Résultats expérimentaux : Fiches d'expériences, expériences E1 à E19 et PHEB 02" - Rapport DPRE/SERE n° 98/014 (I) – Non-public reference.

[4.2_15] P. RONGIER, T. BONHOMME, C.PEREZ "2ème Synthèse expérimentale H2PAR - Détermination des conditions d'inflammation d'un mélange Air/H2/H2O par les recombineurs Siemens et A.E.C.L – Fiches d'expériences" – Rapport IPSN/DPRE/SERLAB n°99/004 (I) – Non-public reference.

[4.3_1] I. Huhtiniemi and D. Magallon and H. Hohmann, "Insight into steam explosions with corium melts in KROTOS", Nuclear Engineering and Design, Volume 204 (2001) pages 391-400.

[4.3_2] Meignen, R., Berthoud, G., 1997. "Fragmentation of molten fuel jets", Proceedings of the International Seminar of Vapor Explosions and Explosive Eruptions, pp. 83–89.

[4.3_3] J. Namiech , G. Berthoud and N. Coutris, "Fragmentation of a molten corium jet falling into water", Nuclear Engineering and Design, Volume 229, Issues 2-3, April 2004, Pages 265-287.

[4.3_4] Final Report of Phase 1 of International OECD programme SERENA. Draft, 1 June 2006. To be published end of year as CSNI report.

[4.3_5] M. Bürger, S.H. Cho, E.V. Berg, A. Schatz, "Modelling of drop fragmentation in thermal detonation waves and experimental verification", Specialist's Meeting on FCI, Santa Barbara, California, USA, Jan 5-8, 1993.

[4.3_6] Board S.J., Hall, R.W. and Hall, R.S., "Detonation of fuel coolant explosions". Nature 254 3, 1975, pp. 319–321.

[4.3_7] W. W. Yuen and T. G. Theofanous, "On the existence of multiphase thermal detonations", International Journal of Multiphase Flow, Volume 25, Issues 6-7, 11 September 1999, Pages 1505-151.

[4.3_8] G. Berthoud, "L'interaction CORIUM-EAU : synthèse et analyse des résultats expérimentaux", Note CEA STT/LPML/87/28/C, October 1987 – Non-public reference.

[4.3_9] D.H. Cho, D.R. Armstrong and W.H. Gunther, "Experiments on Interactions Between Zirconium-Containing Melt and Water", NUREG/CR-5372, 1998.

[4.3_10] K. Moriyama, Y. Maruyama, T. Usami ands H. Nakamura, "Coarse break-up of a stream of oxide and steel melt in a water pool", JAERI-Research 2005-017, 2005 – Non-public reference.

[4.3_11] F. Duplat, G. Berthoud and M. Hamon: "Recent Results on the BILLEAU Premixing Experiment using Cold and Hot Spheres (2200°C) and Comparisons with MC3D calculations", Proc. 2nd Int. Conf. on Advance Reactor Safety, ARS'97, Orlando, Florida, 1-4 June 1997.

[4.3_12] L. Meyer, "QUEOS, An Experimental Investigation of Pre-mixing Phase with hot spheres", Proc. OECD/CSNI Specialist Meeting on FCI, 19-21 May 1997, Tokai-Mura, Japan, P155-166.

[4.3_13] Nuclear Safety Research in OECD Countries, "Major Facilities and Programmes at Risk", 2001, NEA#03145, ISBN: 92-64-18468-6.

[4.3_14] S.K. Wang, C.A. Blomquist, B.W. Spencer, L.M. Mc Umber and J.P. Schneider, "Experimental Study of the Fragmentation and Quench Behaviuor of Corium Melts in Water", Proc. 5th Nuclear Thermal Hydraulics, San Francisco, P 120-135, 1989.

[4.3_15] D. Magallon, H. Hohmann and I. Huhtiniemi, "Lessons Learnt from FARO/TERMOS Corium Melt Quenching experiments", Nuclear Engineering and Design, Volume 189 (1999) pages 223-238.

[4.3_16] D. Magallon and I. Huhtiniemi, "Energetic event in fuel-coolant interaction test FARO L-33", ICONE-9 Conference, Nice, France, 8-12 April 2001.

[4.3_17] A. Kaiser, W. Schutz and H. Will, "Melt Water Interaction Tests (PREMIX tests PM10 and PM11)", Proc. OECD/CSNI Specialist Meeting on FCI, 19-21 May 1997, Tokai-Mura, Japan, P646-655.

[4.3_18] W. Cherdron, H. Jacobs, A. Kaiser and W. Schütz, "Measurement of Steam Explosion Loads: the ECO experiments", NURETH-11, Avignon, France, 2-6 October 2005.

[4.3_19] "Characterization of Molten-Fuel Coolant Interaction Processes", EUR 19567 EN, 1999,4th EC Framework Program.

[4.3_20] R. Meignen, "Status of the Qualification Program of the Multiphase Flow Code MC3D", Proceedings of ICAPP '05, Seoul, KOREA, May 15-19, 2005, Paper 5081.

[4.3_21] R. Meignen et al., "Comparative Review of FCI Computer Models Used in the OECD-SERENA Program", Proceedings of ICAPP '05, Seoul, KOREA, May 15-19, 2005, Paper 5081.

[4.3_22] Bürger M., "Comparison and Theoretical Interpretation of Experiments on Hydrodynamic Drop Fragmentation". IKE 2-FB-16, (CEC-RCAMFCI Project, Report INV-MFC(98)-D014), July 1998, Universität Stuttgart - Non-public reference.

[4.3_23] Park, H.S., Hansson, R.C., Sehgal, B.R., "Fine fragmentation of molten droplet in highly subcooled water due to vapor explosion observed by X-ray radiography", Experimental Thermal and Fluid Science 29 (3), pp. 351-361.

[4.3_24] J. H. Song, I. K. Park, Y. S. Shin, J. H. Kim, S. W. Hong, B. T. Min and H. D. Kim, "Fuel coolant interaction experiments in TROI using a UO_2/ZrO_2 mixture", Nuclear Engineering and Design, Volume 222, Issue 1, May 2003, Pages 1-15.

CHAPTER 5 : PHENOMENA THAT COULD LEAD TO A LATE CONTAINMENT FAILURE

5.1 MOLTEN CORIUM-CONCRETE INTERACTION (MCCI)

5.1.1 BACKGROUND TO MCCI AND THE OBJECTIVES OF THE R&D PROGRAMME

In the event of a failure of the reactor vessel, the corium consisting of the molten core and internal structures will spread over the concrete basemat at the bottom of the reactor pit. The contact between the corium and the concrete leads to an event generally referred to as the molten corium-concrete interaction or MCCI. This interaction causes the gradual ablation of the basemat and the walls of the reactor pit. The failure of the basemat would lead to the leakage of radioactive products outside the containment building. A breach in the side walls of the reactor pit may also result in the corium coming into contact with water present nearby. The vaporisation of this water would lead in turn to an increase in the pressure within the containment building. The pressure within the containment building is also increased by the release of gasses produced during the ablation of the concrete. The kinetics of this process is similar to those of the ablation of the basemat, i.e. from one to several days. Finally, it should also be noted that the production of aerosols during the MCCI causes changes in the behaviour of the aerosols in the containment building and hence in the source term.

The R&D carried out in relation to MCCI aims to provide the understanding needed in order to make an acceptable prediction of the kinetics of the axial and radial ablation of the basemat and reactor pit walls, together with the changes in the sources of gasses and aerosols produced during an MCCI.

5.1.2 PHENOMENOLOGY OF MCCI AND THE RESULTING PROBLEMS

Given the high temperature of the molten material coming from the reactor vessel ($T_{oxides} > 2500$ K) which is maintained at that temperature by the residual power, and the melting point of concrete (around 1600 K for silica concretes), the main process occurring during an MCCI is the melting of the walls of the reactor pit. As the concrete consists mainly of SiO_2, $CaCO_3$ and H_2O, it releases both condensed (SiO_2, CaO) and gaseous phases (H_2O, CO_2) into the pool. The corium pool therefore contains heavy oxides from the core (UO_2, ZrO_2), light oxides from the concrete (mainly SiO_2 and CaO) and a range of metals (Fe, Cr, Ni, Zr). All these constituents are stirred together by the gasses resulting from the decomposition of the concrete. The possible contact between metals and condensed or gaseous oxides could give rise to possibly exothermic oxidation reactions with reaction products including new gasses such as H_2, CO and SiO(g). Finally, contact between the high temperature corium and cooler concrete may cause the corium to solidify with the possibility of debris being suspended in the liquid. The corium pool therefore contains many constituents in a variety of phases (liquid, solid and gas). The composition and physical properties are constantly changing during the MCCI due to the decomposition of the concrete and the chemical reactions taking place.

As the rate of ablation of the vertical and horizontal concrete walls depends on the ratio of the heat flux received by the walls and the energy per unit volume necessary for their ablation, any determination of the ablation kinetics of these walls requires the calculation of the distribution of heat flux at the boundaries of the corium pool. It is generally considered that the gas-induced stirring results in the liquid phase or phases within the corium-concrete pool being essentially homogeneous, and very steep temperature and concentration gradients could exist at the interfaces (see Figure 5.1-1). Using this approach, it can be seen that the heat flux at the boundary of the pool may be expressed in terms of a coefficient of convective heat exchange and the temperature at the interface between the corium pool and the material separating the concrete from the corium. This interface temperature depends on the structure of the interface between the corium and the concrete (i.e. whether or not a stable crust is formed). Depending on the rate of release of gasses from the concrete and the densities of the oxide and metal phases (which are only partially miscible) the pool may consist of a single layer of mixed oxide and metal, or of several layers of oxide and metal stratified according to their relative densities. In the latter case (see Figure 5.1-2), a determination of both the interface temperature and the heat exchange coefficient is therefore needed, not only at the interface between the layers and the concrete, but also at the interface between the liquid oxide and liquid metal.

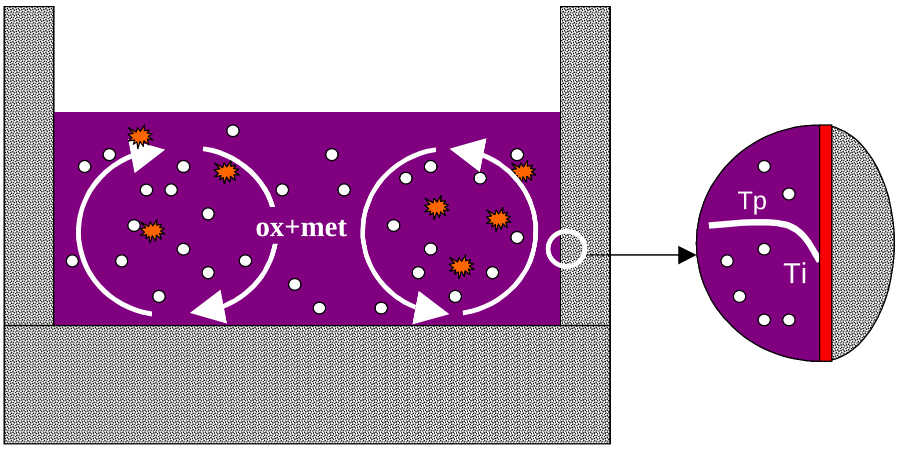


Figure 5.1-1: Corium pool in a mixed configuration - Close-up of the corium-concrete interface

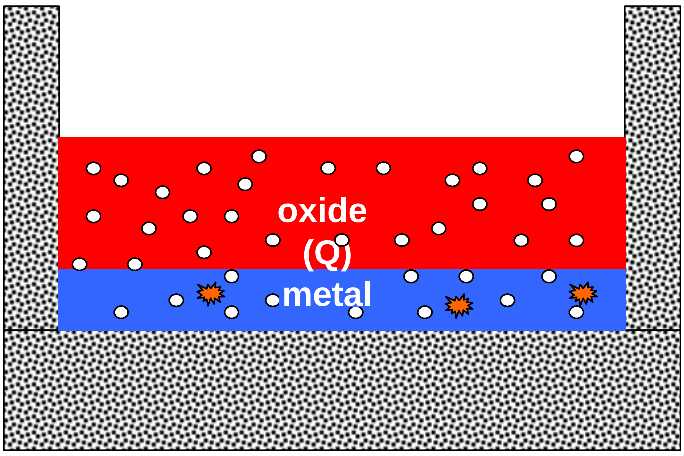


Figure 5.1-2: Corium pool in a stratified configuration

5.1.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

Studies of the molten corium-concrete interaction are based on both experimentation and modelling. The aim of the studies is to improve understanding of the phenomena involved (heat transfer, solidification, mixing, etc.), and to study the molten corium-concrete interaction as a whole using simulation code based on models derived from experimental data.

5.1.3.1 Experimental work

Two types of test are used in the investigation of the molten corium-concrete interaction:

- Analytical tests, based on model systems, used to measure some of the characteristic variables associated with the phenomena such as the interface temperature, heat exchange coefficient, etc.
- Global tests to measure ablation rates during a molten corium-concrete interaction using prototype materials or simulants.

5.1.3.1.1 Analytical tests

Heat exchange coefficients

Numerous analytical tests [5.1_1], [5.1_2], [5.1_3] seeking to determine the heat exchange coefficient between a liquid pool and a porous wall allowing the passage of gasses have been carried out during the past thirty years. An analysis of the results of these tests shows that the physical properties of the liquids used are often close to those of water, and that the available data mainly relates to horizontal walls. In the case of water, the measurements carried out as part of the various experimental programmes all give very similar results for a given superficial gas velocity. Data relating to viscous liquids (comparable with concrete-enriched corium) and vertical walls is relatively rare [5.1_4]. It would, however, appear that the heat exchange coefficients for vertical and horizontal walls are broadly similar when water is used as the test liquid.

Results relating to the heat exchange coefficients between two immiscible liquids through which a gas is passing (the same configuration as that in a stratified reactor) are even rarer (Greene [5.1_5], Werle [5.1_6]). Analysis of the results of these tests reveals a wide variation in the measurements (by a factor of between 5 and 10), a lack of relevance to the systems being studied (absence of solidification at the interface) and experimental procedures giving rise to a degree of caution. These factors make it difficult to apply these results to reactor studies.

Interface temperature

The ARTEMIS programme [5.1_7] is the only programme so far to have determined the temperature at the boundaries of a corium pool. The programme made use of simulants, including salts (LiCl and $BaCl_2$) with phase diagrams similar to those of reactor constituents, to investigate the relationships between physico-chemistry and thermalhydraulics. Tests carried out using one-dimensional configurations (horizontal corium-concrete interface) have confirmed that, under test conditions believed to be representative of a reactor, the interface temperature was close to the liquidus temperature of the pool, and that the pool temperature decreased in line with the decrease in liquidus temperature as the concrete content of the pool increased. The second phase of the programme (ARTEMIS 2D) will expand the study to include multi-dimensional configurations, and a third phase will concentrate on stratified oxide/metal configurations, including a determination of the boundary conditions at the interface between the liquid layers.

Mixing and separation of immiscible liquids through which a gas is passing

A study of the mixing and separation of immiscible liquids through which a gas is passing has been carried out with the aim of predicting the configurations (mixed or stratified) of a corium pool during an MCCI. Most of the experimental work on this topic has been carried out using simulants [5.1_8], [5.1_9], [5.1_10] and was restricted to hydrodynamic considerations only (no effects of phase changes). This work aimed to determine the mixing and separation thresholds in terms of the superficial gas velocity (or void ratio) as a function of the difference in density between the two liquids. A review of the data reported in [5.1_10] indicates a degree of variance, which can be marked, between the results of the various tests. This is partly due to the different physical properties of the liquids.

Physical properties of the materials

Highly analytical tests have also been carried out to investigate the thermophysical (viscosity) and thermochemical properties of the corium needed in order to model the ex-vessel behaviour of the corium.

5.1.3.1.2 Global tests

The global tests were used to investigate the entire interaction, with all the phenomena involved in an MCCI included in a coupled manner. These tests are difficult to carry out, given the technological difficulties to be overcome, including very high temperatures and heating processes. Interpretation of the tests results is also difficult due to the limited number and imprecise nature of the measurements, the delicate estimation of thermal losses, and the presence of phenomena whose influence is often difficult to quantify (ejection of material, anchoring of the crust, etc.). However, despite these difficulties, these tests are still essential as they often reveal phenomena that has not hitherto been identified and which may be significant (e.g., the ejection of MCCI corium under water during the MACE tests). A list of these tests is given in Table 5.1-1.

Form a chronological point of view, the 1D oxide tests (ACE, MACE and SURC) have been performed ten or more years ago. The results of these tests have been used to develop and confirm the hypotheses forming the basis of the corium behavioural models (decrease in the temperature of the pool to close to that of the liquidus), and to partially validate the simulation codes. The 2D tests on prototype oxide materials (OECD-MCCI and VULCANO) were begun much more recently and provide data on the 2D distribution of heat flux during an MCCI. The results from these tests tend to show that the ablation takes place preferentially in the radial direction in the case of silica concrete, and that it is relatively homogeneous in the case of lime-silica concrete [5.1_11]. The analysis and interpretation of the results of these tests is in progress. Of particular interest will be the relative importance of the effects due to physical phenomena (e.g. the type of concrete) and those arising from the specific characteristics of the experimental installations (e.g. type of heating) in explaining these differences in behaviour.

Table 5.1-1: Summary of the global MCCI tests

Programme	Characteristics	Mass of corium	Geometry	Parameters
SURC (1D)	reactor materials + fission products	200 kg	cylinder diameter: 0.4 m	concrete composition, power
ACE (1D)	reactor materials + fission products	250/450 kg	parallelepiped 0.5mx0.5mx0.4m	composition of the concrete, power
MACE (1D)	reactor materials water injection	100/1800 kg	parallelepiped (0.5 to 1.2m)x(0.5 to 1.2m) x 0.4m	composition of the concrete, power, water flow rate
BETA (2D)	alumina-iron thermite stratified oxide-metal	450 kg	truncated cone diameter: 0.4 m	composition of the concrete, power
COMET-L (2D)	alumina-iron thermite stratified oxide-metal	920 kg	cylinder diameter: 0.6 m	composition of the concrete, power
OECD-MCCI (2D)	reactor materials	350 kg	parallelepiped 0.5mx0.5mx0.6m	composition of the concrete, power
ARTEMIS 2D	simulant salts	110 kg	cylinder diameter: 0.3m x height: 0.6m	power, gas flow rate
VULCANO-ICB (2D)	reactor materials	40 kg	half cylinder diameter: 0.3m height: 0.3m	composition, concrete, power

There are no results available from tests involving prototype materials for stratified oxide/metal configurations in which the heating is representative of reactor conditions. The BETA and COMET tests used simulant materials and the use of induction heating implies that power was injected into the metal phase rather than the oxide phase as would be the case in the reactor case. The BETA test showed preferred ablation in the axial direction. This behaviour cannot be extrapolated to a reactor as it was probably due to method of heating used [5.1_12]. The VULCANO tests should provide much of the data that is still missing at the present time.

5.1.3.2 Modelling

Corium behaviour and the interface temperature

A first approach, the earliest to be adopted, is to assume that the temperature of the pool lies between the liquidus temperature and the solidus temperature. The pool has the consistency of a slurry, between the solid phase and the liquid phase. The crust that forms at the boundaries has the same composition as the corium pool and does not alter the internal solid fraction. The interface between the pool (with or without crust) and the concrete consists of a film of molten concrete known as the "slag layer". The solid/liquid equilibrium temperature is therefore equal to the solidus temperature and the entire semi-molten mass is convective. However, this hypothesis is not in agreement with the changes in temperature measured during some of the ACE and OECD-MCCI tests.

More recently, the CEA has developed a new model known as the "phase segregation model". In the MCCI case, this model is derived from the model coupling thermal-hydraulic and physico-chemistry used to describe the in-vessel behaviour of the corium [5.1_13]. In this approach, it is assumed that the pool is liquid and that the crust, consisting entirely of refractory materials (UO_2 and ZrO_2), is deposited at the interface between the pool and the concrete. Although this crust may be unstable, it is possible in the reactor case to consider the situation averaged over space and time and it is sufficient to model the crust behaviour under steady-state conditions. In the same way, a succession of steady-state hypotheses is used to model the coupling between thermalhydraulics and physico-chemistry: at the pool-crust interface, the liquid pool and solid crust are in thermodynamic equilibrium, and the interface temperature is equal to the liquidus temperature of the pool. This model has been partly validated by the ARTEMIS 1D tests in respect of the interactions at a horizontal wall and the ablation and gas superficial velocity corresponding to the long term MCCI phase of the reactor case. This model is less applicable as it assumes equilibrium during the initial transient phase of the interaction and the error may be important in the case of short term MCCI tests. The model also probably fails if the concentration of silica in the pool is high [5.1_14] during the long term phase as this effect was not simulated by ARTEMIS. If there is chemical equilibrium is not reached, the interface temperature of the pool may be lower than the liquidus temperature and the segregation may be less pronounced or even absent.

Heat exchange coefficient

Numerous models and correlations relating to heat transfer between a pool and a porous horizontal wall through which gas is injected are available in the literature [5.1_1], [5.1_2], [5.1_3], [5.1_4], and [5.1_5]. These have been obtained by the simple correlation of the experimental data referred to above (following a dimensional analysis) or by means of a more theoretical approach. They are expressed in the form Nu=f(Re, Pr) or Nu=f(Ra, Pr) according to whether the heat transfer is assumed to be controlled by forced convection due to the gasses from the decomposition of the concrete, or by natural convection. The limited range of physical properties of the fluids used in the tests using simulants (see 5.1.3.1) make it difficult to evaluate these models in spite of the fact that they produce results that are in relatively good quantitative agreement in these tests [5.1_15]. Although these models give identical results on the basis of available experimental data, there is a wide variation in the results of the correlations when they are used with parameters representative of the reactor case. This variation probably indicates that some of the models do not fully reflect all of the physical phenomena present and the relative weightings of the various parameters. For this reason, a model [5.1_16] based on a more phenomenological approach was suggested in 2005. This model gives satisfactory results for the available experimental data (horizontal and vertical walls) but requires additional validation for viscous liquids.

It is important to stress that, in addition to the absolute local value of the heat exchange coefficient between the pool and the crust, the variations in the value of this coefficient along the surface of the pool are also crucial in determining the variations in heat flux and local ablation rates under the quasi-steady-state conditions that characterise the major part of the MCCI phase.

There are also many models for the heat transfer between two liquids available in the literature [5.1_5], [5.1_17]. However, the variations in the few experimental results available makes their assessment in the context of reactor studies extremely difficult.

Mixing and separation of immiscible liquids through which a gas is passing

The results of tests using simulants have been used as a basis for the development of experimental correlations [5.1_10] in which the variation in the results is similar to that in the measurements. These correlations have not been validated by tests on real materials and their use under reactor conditions can lead to a range of results including configurations that are mainly mixed or mainly stratified during an MCCI.

5.1.3.3 Simulation codes

The codes developed for MCCI studies [5.1_18], [5.1_19], [5.1_20] is all based on the same set of assumptions:

- The corium pool consists of various layers (oxides or metals) each homogeneous in both temperature and composition.
- The corium pool may be either mixed or stratified.
- The interface structure is described by a thermal resistance model taking account of the possible formation of a solid crust and/or a zone containing concrete decomposition products (slag layer). The ablation rate is calculated from the discontinuity in the heat flux at the ablation front (Stefan equation) treating the ablation as a thermal decomposition process.

The differences between the codes mainly relate to the corium behaviour models considered and the sub-models or correlations used (void ratio, heat exchange correlation, etc.).

The TOLBIAC-ICB [5.1_18] code, developed by the CEA, is based on a phase segregation model which assumes that the pool is liquid and that a crust of refractory materials forms at the interface. The interface temperature is equal to the liquidus temperature (greater than 2200 K for most of the MCCI). It is calculated from the composition of the pool via a coupling with the GEMINI2 thermodynamic code.

The CORCON code [5.1_20], on the other hand, assumes that the pool is a mixture with solid debris held in suspension in a liquid. The model also assumes the existence of a slurry zone if the temperature of the pool lies between the liquidus and solidus temperatures. The “slag layer” model is used to describe the interface. In this approach, the interface temperature is the melting point of the concrete (close to 1600 K for silica concrete).

The MEDICIS code [5.1_19], developed at the IRSN, uses more general modelling to describe the behaviour of the pool interfaces. It assumes the existence of a mushy zone at the interface between the liquid pool and the concrete consisting of a convective zone and a conductive zone, the extents of which are still to be defined. In this approach, the interface temperature used for the convective transfers from the pool to the interface is then the threshold temperature at the point where the convective zone meets the conductive zone within the mushy zone (see Figure 5.1-3). As no model currently determines this interface temperature, also referred to as the solidification temperature, in a satisfactory manner in all cases, it is calculated for the present by linear interpolation using a user parameter γ between the liquidus temperature (γ=0) and the solidus temperature (γ=1). The extreme values of γ=0 and γ=1 correspond to the interface modelling in the TOLBIAC and CORCON packages respectively. The best choice of γ, derived from the current experimental validation of MEDICIS by means of MCCI experiments, is in the range (0 ; 0.3). The segregation of phases within the pool towards the crust is ignored as the crust at the corium/concrete interface is usually thin. However, the liquidus and solidus temperatures, together with the relationship between the molten fraction of the corium, its composition and its temperature, are all calculated prior to the MCCI simulation using a GEMINI2 simulation. A film of molten concrete (slag layer) is also assumed to exist between the concrete and the pool (with or without crust).

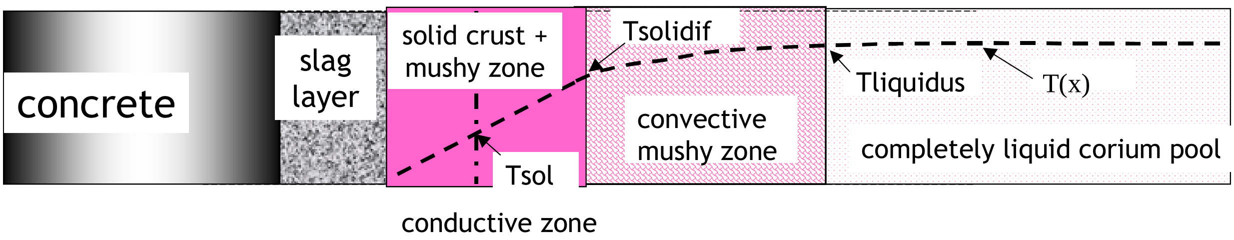


Figure 5.1-3: Modelling of the corium/concrete interface in MEDICIS

The main stages in an MCCI simulation are as follows:

- Calculation of the physical properties of the pool.
- Calculation of the solid fraction (if applicable).
- Determination of the interface temperature.
- Calculation of the heat transfer coefficients.
- Estimation of the ablation rate.
- Mass balance taking account of the chemical reactions and energy balance including surface radiation in order to obtain the composition and temperature of each zone in the pool.

- Calculation of the thicknesses of the crust and the ablations.
- Updating the shape of the cavity and the heat exchange surfaces.

Given the uncertainties in the heat transfer models and the characteristics of the interfaces, it is essential that the code tool has sufficient flexibility to enable sensitivity studies to be carried out and changes to be made easily. This is the case with TOLBIAC-ICB and MEDICIS, but not with CORCON.

The simulation code is validated against the results of the tests described above. This validation process is complex in the case of the global tests due to transient effects and the large number of phenomena observed during these tests that are not necessarily modelled in the code including the ejection of corium and deposits of material on the walls of the test sections.

5.1.4 SUMMURY AND OUTLOOK

5.1.4.1 Knowledge requirements and uncertainties

Recent experimental results and reactor simulations carried out using the MEDICIS and TOLBIAC-ICB code have revealed the following main uncertainties [5.1_21]:

- The nature and properties of the corium pool interfaces, including the solidification temperature, and the transfer of heat and mass across the corium/concrete interfaces.
- The 2D distribution (axial and radial) of the heat flux in a homogeneous pool.
- The transfer of heat and mass between the metal and oxide layers in a stratified configuration.
- Changes in the configuration of the pool (stratification).

In terms of the interface structure, an analysis of the available 1D MCCI experimental results relating to a homogeneous pool with a horizontal interface shows that the results are close to those derived from the phase segregation model, including the liquid pool, solidification temperature close to the liquidus temperature, high two-phase heat transfer coefficient, and formation of refractory crusts at the corium/concrete interface, albeit with a less refractory composition that derived from the phase diagrams. However, solidification tests in the absence of gas show a tendency for the phenomenon of accumulation of refractory components at the interface to disappear at high silica concentrations. The ARTEMIS 2D tests using simulants and new instrumentation dedicated to the measurement of the interface temperatures at horizontal and vertical walls should provide an additional understanding of the physical basis of the interface phenomena.

Considerable uncertainties still surround the last three points in the above list, i.e. the distribution of the heat flux at the pool interfaces, the transfer of heat and mass between the metal and oxide layers in a stratified configuration, and changes to the configuration of the pool. In particular, an improved understanding of the heat transfer between the metal and oxide layers and of the type of pool configurations and their changes will enable the currently considerable uncertainties surrounding the axial and radial ablation kinetics in a stratified pool configuration to be reduced to acceptable levels [5.1_19].

5.1.4.2 Future work

The continuation of the experimental programmes currently in progress (ARTEMIS 2D and VULCANO MCCI), together with planned future experiments (follow-up of OECD-MCCI) and the associated analytical work (interpretation and benchmark exercises with codes) will improve understanding of the physical phenomena determining the structure of the interfaces and will add to understanding of the 2D distribution of heat flux in a homogeneous pool configuration. The VULCANO-MCCI and ARTEMIS tests relating to oxide-metal interfaces in a stratified pool should, among other expected results, contribute to a reduction in the remaining uncertainties and provide data relating to heat transfers between oxides and metals and changes in the configuration of the pool.

However, should the results obtained from these various tests prove to be insufficient, additional experimental programmes will be required. Two analytical test programmes using simulants are currently envisaged:

- The ABI programme concentrating on the study of heat exchanges between two immiscible liquids though which a gas is passing. This programme will obtain more precise data on the oxide/metal heat transfers which will have a major impact on the results of reactor simulations in the presence of a stratified configuration [5.1_21] ;
- The CLARA programme will provide a better understanding of 2D ablation in a homogeneous pool configuration by studying in more detail the heat transfers between a heated pool and a porous wall through which a flow of gas is passing in 2D configuration with simplified boundary conditions (without ablation).

A large-scale qualification test would also be required.

A programme of additional studies is currently in preparation with the aim of reducing the uncertainties in relation to the thermochemical properties of corium (determination of phase diagrams for certain selected oxide and metal mixtures under conditions where the thermochemical data is insufficient or uncertain). An analysis of these experimental results will complete the NUCLEA thermochemical database used by the thermodynamic simulation codes (e.g. GEMINI2 [5.1_22]).

TABLES IN CHAPTER 5

FIGURES IN CHAPTER 5

REFERENCES FOR CHAPTER 5

[5.1_1] Kutateladze S.S., "Boiling and bubbling heat transfer under free and forced convection of liquid", Int. J. Heat and Mass Transfer, 22, 1979.

[5.1_2] Duignan M.R., Greene G.A., Irvine Jr T.F., "Heat transfer from a horizontal bubbling surface to an overlying water pool", Chem. Eng. Comm., 87, 1990.

[5.1_3] Bonnet J.M., "Thermalhydraulic phenomena in corium pools for ex-vessel situations: the BALI experiments", 8th Int. Conf. On Nuclear Engineering, 2000.

[5.1_4] Felde D.K., Kim H.S., Abdel-Khalik S.I., "Convective heat transfer correlations for molten core debris pools growing in concrete", Nucl. Eng. Des., 58, 1980.

[5.1_5] Greene G.A., "Heat, mass and momentum transfer in a multifluid bubbling pool", Advances in Heat Transfer, 21, 1991.

[5.1_6] Werle H., "Enhancement of heat transfer between two horizontal liquid layers by gas injection at the bottom", Nucl. Technology, 59, 1982.

[5.1_7] Veteau J.M., "ARTEMIS program: investigations of MCCI by means of simulating materials experiments", ICAPP'06, 2006.

[5.1_8] Epstein M., Petrie D.J., Linehan J.H., Lambert G.A., Cho D.H., "Incipiant stratification and mixing in aerated liquid-liquid or liquid-solid mixture", Chem. Eng. Science, 36, 1981.

[5.1_9] Casas J.L., Corradini M.L., "Study of void fractions and mixing of immiscible liquids in a pool configuration by an upward gas flow", Nucl. Technology, 99, 1992.

[5.1_10] Tourniaire B., Bonnet J.M., "Study of the mixing of immiscible liquids by sparging gas: results of the BALISE experiments", 10th NURETH, 2003.

[5.1_11] Farmer M.T., Lomperski S., Basu S., "Results of reactor material experiments investigating 2D core-concrete interaction and debris coolability", ICAPP'04, 2004.

[5.1_12] Alsmeyer H., "BETA experiment in verification to the WECHSL code: experimental result on the pool concrete interaction", Nucl. Eng. Des., 103, 1987.

[5.1_13] Seiler J.M., Froment K., "Material effects on multiphase phenomena in late phases of severe accidents of nuclear reactor", Multiphase Science and Technology, 12, 2000.

[5.1_14] Journeau Ch., Jegou C., Monerris J., Piluso P., Frolov K., "Phase macrosegregation during the slow solidification of prototypic corium", 10th NURETH, 2003.

[5.1_15] Bradley D.R., "Modelling of heat transfer between core debris and concrete", ANS Proceedings of the 1988 Nat. Heat Tansfer Conf., 1988.

[5.1_16] Tourniaire B., "A heat transfer correlation based on a surface renewal model for molten core concrete interaction study", Nucl. Eng. Des. 236, 2006.

[5.1_17] Blottner F.G., "Hydrodynamics and heat transfer characteristics of liquid pools with bubbles agitation", NUREG Report CR-0944, 1979.

[5.1_18] Spindler B., Tourniaire B., Seiler J.M., Atkhen K., "MCCI analysis and applications with the TOLBIAC-ICB code based on phase segregation model", ICAPP'5, 2005.

[5.1_19] Cranga M., Fabianelli R., Jacq F., Barrachin M., Duval F., "The MEDICIS code, a versatile tool for MCCI modelling", ICAPP'5, 2005.

[5.1_20] Bradley D.R., Gardner D.R., Brockmann J.E., Griffith R.O., "CORCON-MOD3: an integrated computer model for analysis of molten core concrete interactions", NUREG Report CR-5843, 1993.

[5.1_21] Tourniaire B., Spindler B., Coulon V., Augé A.C., Cranga M., "Transfert de chaleur entre le bain de corium et le béton en cours d'interaction corium/béton – Analyse du groupe de travail CLARA", Rapport CEA/DEN/DTN/SE2T/LPTM/05-105 – 2005 – Non-public reference.

[5.1_22] Cheynet B., Chevalier P.Y., Fischer E., "Thermo suite", Calphad, 26, n°2, pp. 167-174 - 2002.

CHAPTER 6 : IN-VESSEL AND EX-VESSEL RETENTION AND COOLING OF CORIUM

6.1 IN-VESSEL RETENTION

6.1.1 DEFINITION, PHENOMENOLOGY AND SAFETY CHALLENGES

In-Vessel Retention (IVR) requires that the integrity of the second containment barrier, the reactor vessel, is maintained in the event of a severe accident with a meltdown of the reactor core and corium possible relocation to the bottom of the vessel.

In-vessel retention may be achieved by reflooding the reactor cooling system to halt the progress of the core meltdown, or by flooding the reactor pit in order to remove heat from the vessel once corium has migrated to the lower plenum, and hence prevent a rupture of the vessel.

The possibility of in-vessel retention (IVR) first became apparent following the reactor accident at Three Mile Island 2 (TMI 2) in 1979. Post-accident examination of the reactor after 1986 [6.1_1] revealed the remains of part of the core (around 20 tonnes of corium) at the bottom of the vessel which had remained intact. The ability of the vessel to withstand the thermal load resulting from the residual power released by the corium has been attributed [6.1_2] to the fact that the core relocation took place under water (reactor vessel flooded and under a pressure of around 100 bar) and that water was able to circulate through a space that had formed between the corium and the walls of the bottom of the vessel (gap cooling).

In relation to the reflooding of the reactor cooling system, it must be stressed that the version of the Severe Accident Management Guide (SAMG) currently in force in France recommends restrictions on the injection of water in situations where the Safety Injection flow rate is considered insufficient to remove the residual power. The operator is recommended to wait for an hour and a half after the SAMG entry criteria have been reached before releasing the water at a low flow rate. These restrictions are due to the various safety risks that may occur during reflooding, including:

- Massive generation of steam associated with the production of hydrogen and pressurisation.
- Risk of a steam explosion when water comes into contact with molten material.
- Continuation of the meltdown despite the injection of water.
- Release of fission products.

In more concrete terms, the reflooding may occur with the core in any condition (fuel rods intact, slightly degraded with ballooning or flows of molten material, layers of debris, corium pool, etc.).

Studies of the TMI2 accident have resulted, fifteen years later, in a number of proposed reactor designs using the concept of retention by additionally submerging the reactor pit in order to transfer heat from the corium outside of the vessel as a safety mechanism. These include, in particular:

- the Westinghouse AP600 and AP1000 [6.1_3];
- the KAERI APR1400 (South Korea);
- the European ESBWR.

6.1.2 IN-VESSEL RETENTION BY FLODING THE PRIMARY CIRCUIT

6.1.2.1 Physical phenomena involved and the current state of knowledge

6.1.2.1.1 Situations in which the fuel rods are intact or only slightly degraded

In cases where the fuel rods are intact or only slightly degraded (temperatures below around 1800 °C), a considerable production of hydrogen is very likely as has been demonstrated by the available test results (QUENCH). The oxidation kinetics are determined by the flow rate of the steam passing through the core, which is in turn determined by the rate of advancement of the water front. Currently available codes can satisfactorily model the progression of the water front for intact geometries up to the point at which the first deformations or molten material flows begin to appear. The main unknowns are the geometry of the fuel rods at the time of reflooding and the associated heat exchange relationships. In this type of configuration, it is probable that the progression of the degradation of the core will be halted, provided that the flow of water is sufficient, and that the reflooding itself does not result in thermal shock causing mechanical destruction and the collapse of a major proportion of the fuel. The characteristics of the resulting debris bed could be such as to prevent any further cooling. The criteria for a collapse of the fuel and the scale of the resulting debris remain unknown.

6.1.2.1.2 Situations with debris beds

Once the fuel rods have collapsed, the configuration of the core becomes much more complex than in the situation in which the fuel rods are intact or only slightly degraded. In particular, it becomes a porous environment which greatly increases the pressure differentials and makes it much harder for water to penetrate the collapsed areas. If the water cannot reach certain areas of the debris bed, it is generally assumed that these areas continue to rise in temperature until their melting point is reached and a molten pool is formed. It should also be noted that a debris bed can form in the lower head when the molten corium flows into water. The maximum power that can be extracted from a debris bed by water before it dries out and melts is known as the critical flux. It is expressed in terms of the flux through 1 m^2 of the upper surface of the debris bed. At the present time, the phenomenology of reflooding layers of debris is understood to a satisfactory degree. However, the many models that have been developed since the 1980s are mainly based on 1D experimental studies. A number of uncertainties therefore remain in relation to the extension of these results to multi-dimensional geometries (EDF SILFIDE [6.1_23]). In particular, a number of simulations and experimental observations, regrettably incomplete, suggest that the power capable of being extracted from a debris bed is higher in a multi-dimensional configuration (by a factor of two) and that, even after drying, the rate of flow of steam through the debris bed is still sufficient to maintain part of the debris below its melting point. The uncertainties surrounding these phenomena remain, however, considerable. In the case of TMI2, there is still no explanation of why the debris melted in the TMI2 core despite reflooding at high pressure (the critical flux is approximately proportional to the square root of the pressure). The reflooding of a highly degraded core or a debris bed remains the most poorly modelled phenomenon in the currently available codes.

Multi-dimensional models are currently available to reproduce the phenomenology in a debris bed. The main existing models are incorporated in the ICARE/CATHARE (IRSN), WABE(IKE/GRS) and MC3D (CEA/IRSN) codes. The multi-dimensional effects are still to be confirmed experimentally at a large enough scale to enable reliable measurements to be made of the local temperatures and of the production of steam during reflooding. A preliminary project is currently in preparation at the IRSN with the aim of assessing the relevance and feasibility of carrying out this type of test.

6.1.2.1.3 Situation with the presence of a liquid corium pool

During the TMI2 accident the reactor vessel was filled with water at the moment when the molten corium began to flow. Around 10 tonnes of oxidic corium (~1 m^3) migrated to a compact mass in the lower head and around 10 further tonnes of debris was deposited in a layer around and above the compact corium mass. An analysis of samples taken from the reactor vessel has shown that the temperature of the inner surface of the vessel in contact with the corium mass rose to around 1100 °C while the external surface reached around 800 °C. The internal pressure was around 100 bar. The vessel then cooled very slowly. All the thermal simulations indicate that, assuming a perfect contact between the corium mass and the vessel, this temperature should have continued to rise until the vessel ruptured. The suggested explanation is that a gap formed between the corium and the vessel. The formation of this gap was due to two phenomena:

- water present in the porous steel boiled preventing contact between the corium and the steel;
- the solidifying corium and the vessel that was becoming hotter expanded at different rates.

The infiltration and recirculation of water within this gap would have cooled the vessel sufficiently to prevent it rupturing.

A number of experimental studies have been carried out in order to investigate this phenomenon:

- Small-scale tests of the flow of molten material into water at the base of a reactor vessel in order to reproduce and analyse the mechanism experimentally. Tests of this type were carried out by Fauske *et al.* (FAI tests) [6.1_11], in Japan (JAERI) and in Korea (KAERI) [6.1_12]. All these tests were carried out using alumina thermite to simulate the corium.
- Tests to analyse the maximum power that could be evacuated by water in a gap where the limiting mechanism is the critical flux were carried out by IBRAE (Russia) [6.1_13], Siemens (Germany) [6.1_14] and KAERI (Korea) [6.1_15].

The conclusion at the present time is that the gap could only be formed by boiling water and that the differential expansion mechanism is not a likely source. One interpretation made by the CEA [6.1_4], shows that the possibility of power being evacuated by water in the gap is also unlikely at the low pressure used as the reference case for severe accidents in a PWR. For example, with a gap of 3 mm and a pressure of 1 bar, the critical flux is around 0.02 MW/m^2 compared with the 0.5 MW/m^2 which would be have to be evacuated assuming that half the mass of the core relocates in the lower head. The present conclusion is that there are too many intrinsic limitations (requirement for the permanent presence of water in the lower head, low critical heat flux, closure of the gap by melting of the vessel, etc.) to accept that the mechanism of cooling by the formation of a gap is credible in the vast majority of cases at low pressure.

In the light of current knowledge, it would appear difficult to show that the reflooding of the reactor cooling system would avoid a breach in the reactor vessel at the moment when a debris bed forms in the core, if the primary circuit is depressurised and in the absence of reflooding of the reactor pit.

6.1.2.2 Experimental programmes

A brief description of the main planned, in progress or recently carried out experiments is given below:

- LOFT-FP: This project, completed in 1985, was carried out by INEL (USA) on a cluster of 121 UO_2 fuel rods with neutron heating (in-pile). The test investigated the degradation of the components and the release of fission products at local temperatures of up to 2400 K. Cooling was initially by means of steam followed by reflooding with water.
- PBF-SFD: This project, completed in 1985, was carried out by INEL (USA) on a cluster of 32 non-irradiated UO2 fuel rods with neutron heating (in-pile). This test was also designed to study the degradation of the components and the release of fission products, but in this case the local temperatures reached between 2600 K and 3100 K. Cooling was initially by means of steam followed by reflooding with water in some of the tests.
- CORA: This project, completed in 1993, was carried out by FZK (Germany) on a cluster of 25 non-irradiated UO2 fuel rods with electric heating (out-of-pile). These were degradation tests at local temperatures of up to 2200 K. Cooling was initially by means of steam followed by reflooding with water in some of the tests ([3.1_13], [3.1_14]).
- QUENCH: This project was completed in 2005. A further two tests are intended to be carried out but the funding is not guaranteed. These tests were carried out by FZK (Germany) on a cluster of 25 non-irradiated ZrO_2 fuel rods with electric heating (out-of-pile). These were degradation tests at local temperatures of over 2000 K. Cooling was initially by means of steam followed by reflooding with water or steam.
- ISTC 1648 (QUENCH): This ISTC project was carried out by NIIAR (Russia). The objective of the test was to study reflooding under post-LOCA conditions and the tests consisted of three tasks: Degradation and reflooding tests on a section of irradiated VVER fuel, reflooding tests on a new VVER assembly of 31 fuel rods, and the development of a reflooding module for the SVECHA codes by the IBRAE (the Nuclear Safety Institute of the Russian Academy of Sciences).
- PARAMETER: This ISTC project, operated by LUCH (Russia), will investigate the degradation of VVER assemblies of 19 non-irradiated prototypical fuel rods (similar to QUENCH, but with UO_2 pellets). The test is planned to simulate reflooding from above and from below at temperatures of up to 2300 K. Two tests are planned. One of these is a degradation test with reflooding from above and slow kinetics. The second will use reflooding from both above and below.

6.1.2.3 Review and R&D requirements

Three documents ([6.1_22], [6.1_24] and [6.1_25]) summarise the current status of knowledge relative to the various risks for safety arising from the reflooding of the core of a PWR reactor. The documents identify the main remaining uncertainties and the requirements for R&D.

In terms of thermalhydraulics and fuel, the main requirements are a realistic assessment of the cause and effect relationships between phenomena, and the development of a simulation tool capable of reproducing them. This in turn requires more accurate and detailed models of transient situations, in particular the two key transients, degraded core → molten pool, followed by pool in the core → pool in the lower head. Recent models used in current codes are based on a multi-dimensional description in order to provide a better simulation of the transients affecting the materials inside the reactor vessel. However, there is a marked lack of experimental results capable of characterising these phenomena, mainly due to the increasing problems of representativity associated with scaling effects.

At the present time, the three main points still requiring studies are:

1. The changes that occur in a heavily degraded core when it is reflooded (is cooling possible?). There is also a need for tests to characterise the progression of the water front as a function of the geometric configurations and the parameters characterising the debris formed from irradiated fuel. The distribution of the size of elements of the debris would be an important result of these tests, and this result could be obtained from out-of-pile tests using irradiated fuel rods (ISTC QUENCH project).
2. The changes taking place in a debris bed as it transforms into a molten pool (if cooling is not possible). This requirement is particularly concerned with the phenomena of dissolution and oxidation and their impact on the stratification of the pool. This area of study is one of the objectives of the CORTRAN project.
3. The arrival of corium in the lower head, especially when it is filled with water. This requirement is concerned with the fragmentation, oxidation and cooling of the corium as it comes into contact with the water, and its spread into the lower head. This point is currently being investigated as part of the programmes dealing with steam explosions.

The interest of in-pile tests, albeit necessarily carried out on a small or medium scale, will be with the representativeness of the specific power and the use of real fuel, especially with a significant burnup, with the presence of the fission products in their true physical and chemical state just prior to an accident. It should be noted that the release of fission products during reflooding is discussed in Chapter 7.

6.1.3 IN-VESSEL RETENTION BY FLOODING THE RACTOR PIT

6.1.3.1 General approach: Orders of magnitude

In-vessel retention is currently understood to mean the possibility of retaining the corium in the reactor vessel while maintaining the integrity of the reactor vessel (i.e. maintaining the second containment barrier). It may, however, be possible to extend this concept to include the retention of most of the mass of corium, with a controlled discharge of part of the corium into the reactor pit. In this document, the discussion is limited to the concept of maintaining the integrity of the reactor vessel.

In order to demonstrate that the integrity of the reactor vessel would be maintained in the event of a core meltdown and the migration of all or part of the core to the lower head, a number of other points also have to be demonstrated. These include:

- A demonstration that the reactor vessel does not melt and remains intact at all points, especially in areas of high thermal stress.
- A demonstration that the reactor vessel is mechanically strong enough to withstand a steam explosion inside the reactor vessel.

An order of magnitude calculation shows that the volume of the core in a PWR 900 MWe when compacted would occupy close to the full hemisphere of the lower head. Assuming a residual power of 20 MW uniformly distributed over the surface, the heat flux would be around 0.8 MW/m^2. This heat flux is extremely high and could only be evacuated by efficient convection at the external surface of the corium pool or the reactor vessel. Even if this could be achieved, part of the reactor vessel wall would melt leaving a residual thickness of just a few centimetres. A simple calculation also shows that, if this flux cannot be evacuated efficiently (e.g. due to the external dry-out), the reactor vessel would melt-through within only a few minutes. A system must therefore be provided to extract the heat from the corium pool at all points on the reactor vessel. However, this condition alone is not sufficient. It is also necessary to ensure that the residual thickness of the reactor vessel wall is sufficient to maintain its mechanical integrity. The low residual thickness of the steel reactor vessel would not be capable of withstanding a high pressure in the reactor cooling system and the primary circuit would therefore have to be depressurised. The demonstration of mechanical integrity must therefore be carried out at the final pressure, after depressurisation, taking full account of the thermo-mechanical stresses induced by the power released from the corium. This problem also arises in the case of a pressure peak resulting, for example, from a steam explosion due to water from the reactor cooling system coming into contact with a corium pool that has migrated to the lower head [6.1_10].

6.1.3.2 Analytical approach and corium configurations

The approach initially adopted was to try and demonstrate that the integrity of the reactor vessel would be maintained under the worst case conditions, i.e. with no internal reflooding and with the entire mass of corium migrated to the lower head. Of all the possible conditions, these would produce the highest heat flux transmitted to the vessel. The aim was to demonstrate that the integrity of the vessel would be maintained even under the most pessimistic scenarios.

The configuration of the corium in the lower head determines the distribution of heat flux over the walls of the vessel. An attempt was made to define these configurations by distributing the core inventory (oxides, zirconium and steel) in molten layers according to their respective densities. However, a number of different configurations were still possible depending on the assumptions made. The main assumptions were as follows:

- The degree of oxidation of the zirconium (between around 25% and 80% depending on the scenario).
- The mass of the molten steel (between a few tonnes and several tens of tonnes).
- The distribution of solid layers (debris and crust) during the remelting of the corium in the lower head.

One of the most critical configurations for the reactor vessel is that shown in Figure 6.1-1. In this configuration, the molten metals (density ~6500 kg/m^3) float on top of the molten corium oxide (density ~ 8000 kg/m^3). This is the limiting situation that has been most studied and most of the problems of the boundary conditions and heat transfers within the pool have been resolved. This is also the configuration that was used in the external vessel cooling studies (CHF). This will be referred to in the remainder of this discussion as the reference configuration.

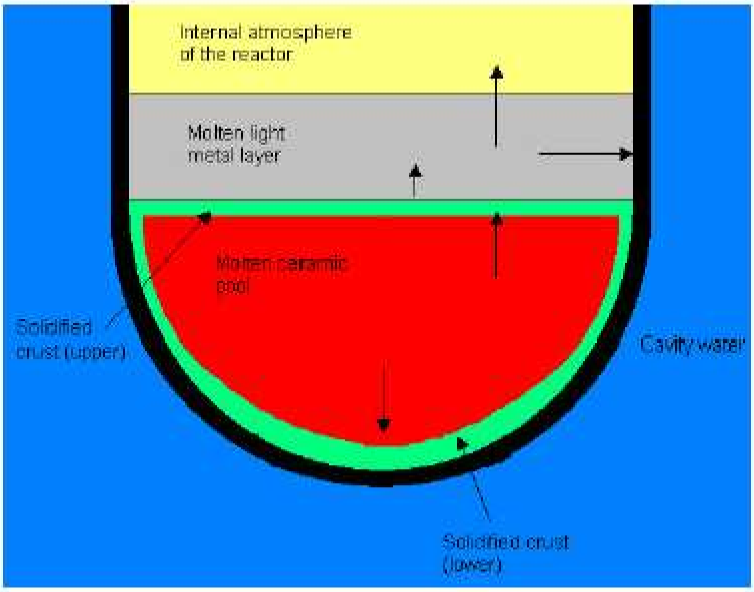


Figure 6.1-1: The configuration of the corium pool in the lower head with external cooling

6.1.3.3 Analysis of the worst case configurations

6.1.3.3.1 Distribution of heat flux and cooling for the worst case configurations

For a given configuration of the corium, the distribution of the heat flux depends on the conditions at the boundary between the molten mass and the solid wall (crust or steel wall of the reactor vessel), and on the natural convection transfer coefficients within the liquid mass. A number of recent studies have attempted to define the temperature conditions at the boundary of a corium pool. A summary of these studies is given in [6.1_5]. The main difficulty is that the molten material consists of a mixture of oxides and/or metals. These mixtures are not eutectic, i.e. they do not have a single melting point. Melting takes place within a temperature interval which may be relatively wide and which depends on the composition of the mixture. It was originally thought that the existence of such a melting interval would result in the presence of a mushy zone between the liquid pool and the solid crust surrounding it. This mushy layer could affect heat transfers in some unknown manner that could not be studied by means of the thermalhydraulic tests on pure substances such as water that are usually used to analyse heat transfer within a pool. The OCDE RASPLAV programme was launched with the aim of resolving this issue. The CEA has shown [6.1_5] that, under steady-state thermal hydraulic conditions (i.e. once the heat flux has stabilised), such a mushy zone could not exist. Once the heat flux has stabilised, and when the external conditions allow, the thickness of the solid crust is constant (i.e. the velocity of the advance of the solidification front is zero). In this case, the interface temperature between the liquid and the solid tends towards the liquidus temperature corresponding to the composition of the liquid mixture. The boundary between the solid and the liquid is clearly defined. This conclusion has been confirmed experimentally in a number of tests (PHYTHER at the CEA (described in [6.1_5]), then RASPLAV [6.1_16] and SIMECO (KTH Sweden). The solidification transient has been studied at the IRSN [6.1_17]. This interface condition also applies to a metal layer. If the liquidus temperature corresponding to the composition of the metal layer (consisting mainly of steel and zirconium) is lower that the melting point of the steel, the steel may be dissolved by the molten metal. The temperature at the interface with the solid steel of the reactor vessel will therefore tend towards the liquidus temperature corresponding to the composition of the liquid metal layer. It is clear that, depending on the composition of the liquid metal layer, the temperature at the inner surface of the reactor vessel wall could be well below the melting point of the steel. The interface temperatures between the liquid and the solid are calculated using thermodynamic codes (such as GEMINI) as a function of the composition of the liquid layer under consideration. The corollary of this behaviour at the interface is that the pool is completely liquid and that the heat transfer equations derived from tests on simulant materials (pure substances such as water) are transposable to real materials.

Heat transfer correlations have been taken from tests on simulant materials (BALI, COPO, ACOPO, RASPLAV Salt, etc.) for a range of different geometric configurations [6.1_9]. Work has also been carried out to validate the CFD simulation codes for natural convection.

6.1.3.3.2 Orders of magnitude of heat flux and flux concentration (or focusing effect)

Given the worst case configuration shown in Figure 6.1-1 and assuming that the entire oxidic mass of the core is relocated in the lower head, the residual power is approximately distributed as follows:

- Around 50% of the residual power dissipated in the oxidic pool is transmitted to the base of the reactor vessel.
- The remaining 50% is transmitted from the oxidic pool to the liquid metal layer.

In the absence of water inside the vessel, the metal layer will transmit most of the power (arising from the oxidic pool and from its own dissipation, i.e. around 60% of the residual power) to the steel wall of the reactor vessel in contact with the liquid metal. The metal layer effectively concentrates the heat flux on the surface of the reactor vessel wall in contact with the metal layer (hence the term ‘focussing effect’). To a first approximation, the heat flux in this region is inversely proportional to the thickness of the metal layer. The thickness of the layer must be greater than around 50 cm (equivalent to 50 tonnes of steel) in order for the heat flux to fall below around 1.5 MW/m^2. The integrity of the vessel can only therefore be guaranteed if the flux transmitted to the vessel can be extracted by natural two-phase convection by means of cooling water boiling outside the vessel. This naturally raises the question of the critical flux at the external surface of the reactor vessel wall.

6.1.3.3.3 Critical flux in the external natural water circulation

The limiting factor is therefore the critical flux associated with the external cooling of the reactor vessel, particularly in the vicinity of the metal layer. It is for this reason that considerable work has been done to determine this critical flux, and to attempt to increase it. A number of tests have been carried out using a variety of approaches. Amongst these, the most interesting at large scale are ULPU (UCSB) [6.1_19], SULTAN (CEA) [6.1_18] and KAIST (Korea).

The first factor affecting the critical flux is the water recirculation system in the reactor pit in which the water is recirculated by natural convection. Simply reflooding the reactor pit is not sufficient. The water must be made to recirculate in such a way as to maximise the flow of liquid along the walls of the reactor vessel. This implies the existence and geometric optimisation of an ascending hot leg (the reactor vessel) and a cold leg maximising the difference in the weights of the water columns in order to form a strong motor for recirculation, while minimising pressure losses. The maximum critical flux is obtained when the steam quality remains negative in the heated zone, i.e. when the flow is sufficient to restrict boiling to the wall in the heated zone. However, it is also necessary for the boiling to become saturated (positive quality) above the heated zone in order for the void effect to develop. There exists an optimum flow rate above which the aspirant effect of the void will be insufficient and below which saturated boiling will occur in the heated zone and he critical heat flux will decrease. The existence of this optimum flow limits the maximum critical flux to a value of around 1.5 MW/m^2.

An analysis of the results of the tests cited above shows a wide variation in the values of the critical flux. ULPU shows values close to 2 MW/m^2 (but with a wide variation in the experimental results), while the SULTAN and KAIST tests give values for the critical flux at a vertical wall of between 1.2 and 1.5 MW/m^2.

A number of methods have been proposed for increasing this critical flux. Many of these are concerned with effects associated with the treatment of the external surface of the reactor vessel. According to some authors [6.1_6], a deposit of porous metal (shoopage) on the external surface of the reactor vessel would result in a significant increase in the critical flux (by a factor of up to two times). There is, however, a lack of agreement as to the effectiveness of this approach and experimental verification at large scale would be necessary.

6.1.3.3.4 Limitation associated with the mechanical strength of the residual thickness of the vessel

For a heat flux of 1.5 MW/m^2, the residual thickness of the reactor vessel that has to withstand the mechanical stress (i.e. the thickness within which the temperature is less than 600 °C) is no more than 1 cm. This thickness is sufficient to withstand pressures of several tens of bar. An increase in the critical flux would lead to a reduction in this thickness (they are inversely proportional) with a consequent reduction in the maximum pressure that the vessel can withstand. These considerations make the work seeking to demonstrate flux values of more than 2 or even 3 MW/m^2 far less interesting.

6.1.3.3.5 Limitation associated with the minimum mass of molten steel

In the light of the preceding discussion, one of the key points in the approach is the mass of steel available to form a metal pool on the surface of the corium. At values of critical heat flux between 1.3 and 1.5 MW/m^2, the minimum depth of molten steel needed to avoid the focusing effect would be around 50 to 60 cm for a 1000 MWe PWR. Given the geometry of this type of reactor, such a depth corresponds to a mass of molten steel of around 50 to 60 tonnes. Studies on the AP600 and AP1000 have shown that a meltdown would release a sufficient quantity of steel from the following structures:

- Internal structures in the lower head.
- The reactor vessel walls.
- Structures in the lower section of the core.

Work carried out as part of the OECD MASCA programme [6.1_20] has shown that the complex physical phenomena occurring could reduce the available mass of metal and this has thrown doubt on the figures for the AP600 and AP1000. These phenomena include:

- Trapping of some of the liquid metal (from the lower internal structures, for example) in the solid oxidic debris.
- Migration of part of the metal in the lower head as a result of physical and chemical effects. For a fixed metal inventory, this would naturally lead to a reduction in the depth of the upper metal layer (see Figure 6.1-2).

The physical and chemical effects arise from the presence of unoxidised zirconium in the metal phase. This zirconium is likely to react with the uranium dioxide present in the oxidic phase forming a metallic uranium phase. If this mixes with the molten steel, it could result in a liquid metal layer that is denser than the oxidic pool causing it to stratify at the bottom of the lower head. It is now possible to take account of this phenomenon in in-vessel retention studies. The use of thermodynamic codes (GEMINI2-Thermodata) makes it possible to calculate the compositions of complex mixtures of oxides and metals in equilibrium at a range of temperatures. The estimated densities of the phases resulting from these calculations can be used to estimate the mass of metal that would stratify below the oxide pool. Subtracting this from the original steel inventory gives the mass of metal present in the upper layer. This model has been developed by the CEA and IRSN with the aim of quantifying the masses of metal needed to ensure that the flux transmitted from the metal layer to the vessel does not exceed the critical flux [6.1_7]. The method has been applied to a number of reactor types including existing French PWRs, the AP600 and AP1000 reactors, and the proposed APR1400 reactor in Korea. The results of these studies show that a critical parameter is the fraction of unoxidised zirconium present in the pool. The greater this fraction, the greater the mass of metallic uranium produced and the greater the proportion of the total mass of metal that migrates to the bottom of the lower head. The complexity of the demonstration of in-vessel corium retention therefore increases with the mass of available metallic zirconium. These studies also show that the results of the simulations are strongly affected by the choice of databases to be used in the thermodynamic calculations and the values of the critical flux at the outer surface of the reactor vessel. In the specific case of the AP1000, the calculations show that, in some configurations, the mass of metal needed for in-vessel retention is greater than the mass of available steel assumed in the American studies used to support the concept.

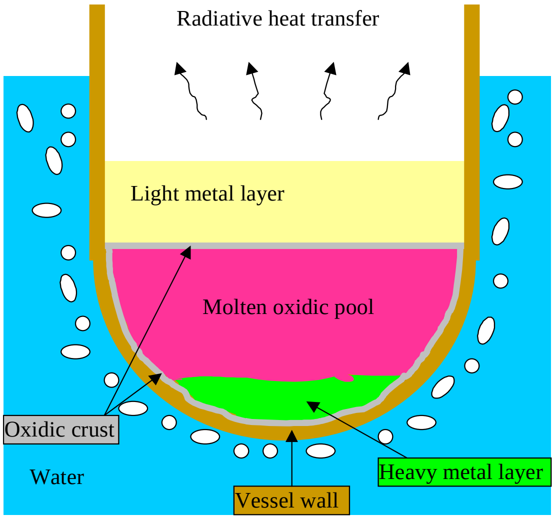


Figure 6.1-2: In-vessel recuperation configuration with inverted metal stratification

These worst-case studies were carried out under steady-state conditions with a fixed pool configuration. The formation of the metal layers and oxide pool implies transients in the growth of the metal layer and an increase in the power from the oxide pool, both of which imply that the critical flux could be reached during these transients.

6.1.3.4 Developments in the in-vessel retention approach in the event of flooding of the reactor pit

The use of worst-case situations does not always demonstrate that the integrity of the reactor vessel will be maintained in all cases. This approach needs to be expanded to take account of the following:

- more realistic corium configurations,
- the possibility of external flooding of the reactor pit and internal reflooding within the reactor vessel,
- situations that may lead to a partial discharge of the corium in the reactor pit, provided that it can be shown that these situations are controllable,
- a range of reactor situations (e.g. when the reactor is shut down and the residual power is lower).

Such an approach must be linked to a series of probabilistic considerations as part of a probabilistic safety assessment.

An injection of water onto a corium pool would eliminate the focusing effect. It has been shown (ANAIS tests carried out at CEA [6.1_8]) that this results in the solidification of the surface metal layer and that a considerable portion of the residual power is transmitted to the reflooding water. The ANAIS tests have also shown that the risk of a steam explosion is limited, under these conditions, to the area where the jet of water impacts the liquid corium. A major explosion following an accumulation of water is avoided as the surface of the corium pool normally solidifies very quickly before a large quantity of water has accumulated in the lower head.

The description of the phase during which the corium migrates to the lower head and remelts needs improvement (developments of ASTEC and DIVA in progress). These studies will have to rely heavily on modelling work as the experimental work is effectively limited to material effects and phenomenology (MASCA, CORTRAN). There are many representativeness problems associated with the effects of scale, power distribution and boundary conditions.

6.1.3.5 Possible technological improvements associated with in-vessel retention

6.1.3.5.1 Deposit of a ceramic material on the inner surface of the reactor vessel walls

The deposition of an internal ZrO_2 ceramic coating with a thickness of a few millimetres has been considered in some projects, including the APR 1400 [6.1_21]. The aim of this ceramic coating is to achieve the following:

- To avoid any breach in the vessel wall due to the impact of a jet of superheated corium (the blowtorch effect).
- To form a refractory barrier to the corium.

The protection against the blowtorch effect may be effective, although it has been shown that it would not necessarily be useful as a crust of solid corium would form at the point of impact of a corium oxidic jet, and a molten metal jet would not, in principle, continue for a long enough period of time to break through the reactor vessel wall.

The refractory barrier effect does not exist. This is because:

- A crust of solid oxide forms anyway between the corium oxide pool and the vessel once steady-state thermal conditions are reached.
- The thickness of the ceramic is not stable when the underlying steel wall melts (under steady-state conditions, the residual thickness of the reactor vessel depends on the heat flux coming from the corium pool).

6.1.3.5.2 Installation of a core catcher under the reactor vessel (or a secondary vessel)

The design of a secondary vessel or a core catcher underneath the VVER-1000 reactor currently being built by the Russians at Tian Wan in China (see Figure 6.1-3) makes use of sacrificial materials consisting mainly of steel, an iron oxide ceramic (Fe_2O_3) and alumina (Al_2O_3). The effect of this material, when in place in the core catcher in sufficient quantity, is as follows:

- It oxidises the remaining zirconium without generating hydrogen (by transforming Fe2O_3 and Zr into ZrO_2 and Fe).
- It reduces the density of the oxidic phase (consisting of a mixture of UO_2, ZrO_2, FeO and Al_2O_3) to a value less than that of the steel. Under these conditions, all the steel sinks below the oxide pool and the flux concentration or focusing effect is eliminated.
- It increases the volume of material and hence reduces the mean heat flux transmitted to the vessel.

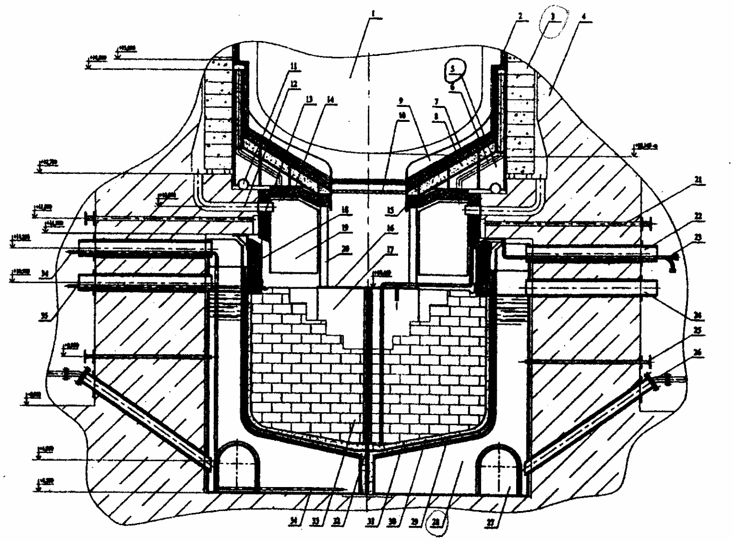


Figure 6.1-3: The Tian Wan core catcher

6.1.3.5.3 Specific mechanisms to increase the critical flux

The CEA has developed a range of technological proposals with the aim of increasing the critical flux at the external surface of the reactor vessel by a potential factor of between two and five times while maintaining the mechanical integrity of the vessel. Such systems may be envisaged for future high power reactors, but this implies modifications to the structures within the reactor pit.

6.2 <u>COOLING OF THE CORIUM UNDER WATER DURING THE MOLTEN CORIUM-CONCRETE INTERACTION</u>

6.2.1 DEFINITION AND OVERALL PHENOMENOLOGY

One strategy that has been considered with the aim of speeding up the cooling of a corium pool and stabilising it during a Molten Corium-Concrete Interaction (MCCI) is to inject water to cover the corium in the reactor pit.

The high solidification temperature of corium (around 2400 K in the case of corium with a low concrete content) results in a crust forming at the corium surface due to the transfer of heat from the corium to the reactor pit by radiation. This crust will naturally be thicker if the corium is covered with water, but this thicker crust will also have the effect of insulating the corium pool from the water and limiting the flow of heat from the corium to the coolant. Order of magnitude calculations have shown that, if heat transfer between the corium pool and the water is only by conduction through the crust, any slowing of the concrete ablation due to the injection of water is minimal. In order for the corium to be cooled effectively under the water, other heat transfer mechanisms in addition to conduction must be present. The R&D work carried out on this topic (experimentation and modelling) aims to identify and quantify the efficiency of these mechanisms.

6.2.2 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

6.2.2.1 Experimental programmes

The main experimental programmes on this topic are those carried out at the Argonne National Laboratory (ANL) since 1989 using real materials (MACE [6.2_1], MSET [6.2_2], and OECD-MCCI [6.2_3]. This latter programme is itself divided into three sub-programmes, SSWICS, MET and CCI. The CEA PERCOLA programme [6.2_4] is also targeted in this area, but uses simulant materials. The ANL programmes include both global experiments and more analytical tests.

6.2.2.1.1 MACE and CCI tests

These global tests aimed to study the coolability of corium during an MCCI using prototypical materials with water injected at the surface of the pool. Three tests were carried out in 1D configurations with ablation of the bottom of the vessel only (M1b, M3b and M4) and four tests were carried out in 2D configurations with ablation of the bottom and sides of the vessel (M0, CCI-1, CCI-2 and CCI-3). The following procedure was used in all the tests: The corium pool was formed by direct heating or by a thermite reaction (melting a corium sample by means of an exothermic chemical reaction), the ablation of the concrete took place via an MCCI, first with no water and then following injection of water when a specific criterion was reached. The effectiveness of the water injection was assessed by comparing the speed of ablation of the concrete before and after the water injection, and by measuring the heat flux extracted at the surface of the corium pool indirectly via the quantity of steam generated. The measured temperature of the pool also provided an indication of the effect of the water injection. Care must be exercised when transposing these results to the reactor case as a number of aspects of the tests were non-representative. In particular, the corium was heated by power dissipated in the liquid corium by the Joule effect (power applied to the liquid) while in the reactor case, the residual power is shared between the liquid and the solid.

These tests have revealed several mechanisms by which the water can cool the corium.

- A number of tests (M0, M3B and CCI2) showed that some of the corium pool was ejected above the crust, entrained by gasses produced from the decomposition of the concrete. This formed a debris bed with a granulometry of around one centimetre. The analytical tests also showed that it is possible to cool debris of this size with water.
- The temperature gradients and thermo-mechanical stresses cause the crust to crack allowing water to penetrate. This phenomenon, known as 'water ingression', is likely to propagate, cooling the entire mass of corium. However, recently developed models [6.2_5], [6.2_6] show that the cracks resulting from the temperature gradient are too small for a corium pool to be cooled effectively by this mechanism alone. Nevertheless, it does play an important part in the thermo-mechanical behaviour of the crust as the cracks reduce its mechanical strength.
- Finally, the surface of the corium is cooled by direct contact between the water and the liquid corium. This phenomenon is highly transient, but it could occur during sudden mechanical failure of the crust, or at the moment when the water is first injected onto the corium pool. This effect is likely to be favoured by the scale of the experimental configuration.

However, it is not currently possible to arrive at a firm conclusion as to the effectiveness of cooling a corium pool during an MCCI by injecting water on top of the pool in a reactor situation, although it does appear that the various phenomena described above could slow down the rate of ablation of the concrete. These tests encountered a number of technological problems which limited the range of the experiments and the study of the phenomena involved.

- The small scale of the tests resulted, in most cases, in the formation of a crust on the top of the pool which anchored to the sidewalls of the test vessel. As the concrete ablation front progressed, a gap opened up between the liquid corium and the crust. This limited the effectiveness of the ejection phenomenon. Given the diameter of the reactor pit in a real reactor, it is likely that the crust would remain in contact with the liquid corium.
- The method of heating the corium by passing a current through it meant that it was not possible to heat the solid crusts. The solidification observed during the tests is not therefore representative of that which would occur in a real reactor.

6.2.2.1.2 MSET test

The purpose of the MSET test, carried out in 2001, was to study the phenomenon of corium ejection through the crust that was observed during the MACE tests. This test was carried out using prototypical materials without ablation of the concrete and with water injection to the top of the pool. The release of gas was simulated by passing a gas at a controlled flow rate through a porous material at the bottom of the corium pool.

The MSET test resulted in the formation of a debris bed, but showed that no ejection took place at superficial gas velocities below 10 cm/s. This naturally raises questions regarding the effectiveness of this mechanism in reactor conditions where the long term superficial velocity is below 5 cm/s. An analysis of the results revealed that the probable causes of this behaviour were as follows:

- Anchoring of the crust, with the formation of a gap between the pool and the crust.
- The likelihood of a large solid fraction as the temperature of the pool was well below the liquidus temperature.

6.2.2.1.3 SSWICS tests

The SSWICS tests [6.2_7] were carried out using prototypical materials and were aimed at a study of mechanism whereby water penetrates the crust via thermo-mechanically induced cracks during an interaction under water ("water ingression"). When this mechanism is occurring, it is assumed that cracks form in the upper crust and increase its thickness by increasing the solidified mass.

These separated effects tests were carried out under transient conditions without heating the corium pool. The pool was produced by a thermite reaction and lay over an inert basemat (no ablation of the basemat). Water was gradually injected onto the corium pool and the cooling kinetics was studied by measuring the vaporisation kinetics. The efficiency of the water penetration was assessed by comparing the flux extracted during the tests with that obtained by transient conduction alone. The permeability of the crust was measured after completion of the test and this data was used to evaluate the critical heat flux using dedicated models.

The series of seven tests provided a quantitative measurement of the influence of the type of concrete (siliceous or LCS), the composition of the corium pool (between 4% and 25% of concrete by weight), and the pressure (1 to 4 bar). The corium slugs obtained following the experiment were sectioned and tested for their mechanical strength.

The main results of the SSWICS tests were as follows:

- The tests confirm that an effect due to the cracking of the crust and the penetration of water into these cracks do exist. This effect was only significant low concrete concentrations below 15% by weight. This would require the prompt injection of water immediately after the relocation of the corium into the reactor pit in the event of an accident. This conclusion is also subject to reservations as the tests were carried out without any injection of gas. A counter-flow of gas would limit the penetration of water into the cracks and hence the effectiveness of the mechanism. The superficial gas velocity is also greatest at the start of the interaction when the phenomenon would appear to be at its most effective. It should also be noted that these tests took no account of the power dissipated in the pool.
- Measurements of the breaking strain of the crust would appear to indicate that, in the reactor case, it is very unlikely that the crust would form a single mass which would be attached to the sidewalls of the reactor pit.

6.2.2.1.4 *The PERCOLA programme*

This experimental programme was carried out by the CEA between 1999 and 2002. Following results from the MACE tests and theoretical work at the CEA [6.2_9] showing that a corium pool could be cooled if it could be converted into a debris bed, this programme aimed to study the phenomenon of corium ejection above a crust by gasses resulting from the ablation of the concrete. This analytical programme made use of simulant materials (water and oil) and revealed a number of ejection mechanisms. The tests also provided a quantitative measurement of the influence of a number of parameters associated with the phenomenon [6.2_4], including:

- The viscosity of the fluid (a measure of the enrichment of the corium with concrete during an MCCI).
- The superficial gas velocity (a measure of the type of concrete and the rate of decrease of gas flow during an MCCI).
- The density of holes in the crust (currently unknown).
- The diameter of these holes (currently unknown).
- The thicknesses of the crust and the debris bed (a measure of the growth in the crust and debris bed during an MCCI following ejection).

6.2.2.2 Modelling

The majority of the modelling work has focussed on the phenomena of corium ejection and water penetration.

An analytical model of corium ejection[1] has been developed as part of the PERCOLA programme [6.2_10]. This model includes physical parameters specific to an MCCI (superficial gas velocity, pool viscosity, etc.) that were not taken into account in the earlier Ricou and Spalding model [6.2_11]. Applying the PERCOLA model to reactor scenarios indicates that a coolable debris bed would form rapidly if the ejection phenomenon was effective [6.2_12]. This model has been validated by the results of the PERCOLA tests but complementary validation is needed using more representative tests including ablation of the concrete and prototypical materials. Some of the input parameters to the model are still unknown, including the density and size of holes in the crust permitting the passage of the corium. Dedicated models for these parameters are needed or, failing this, reliable experimental data. A number of models have been proposed by Farmer with the aim of estimating these parameters [6.2_13].

The current model needs to be expanded to include a description of the debris bed and its influence on the corium ejection, particularly following the growth in the thickness of the debris bed[2].

[1] This model estimates the entrainment rate, the ratio between the volume of liquid ejected per unit time and the volume of gas released per unit time during an MCCI.

[2] The PERCOLA model only takes account of the increase in thickness of the debris bed. It assumes that the gas and corium escape along vertical channels (chimneys) that form within the debris bed.

In relation to the water ingression phenomenon, it should be noted that a critical heat flux correlation has been developed as part of the SSWICS programme [6.2_8]. This was derived from a model of the crust cracking process during water penetration. This correlation has been fitted in the light of the results of these tests without the injection of gas during the solidification of the corium.

6.2.2.3 Simulation codes

As the problem of coolability under water is a sub-set of the larger problem of the molten corium-concrete interaction, it would appear at first glance that there is no need for separate specific simulation codes. The TOLBIAC-ICB code [6.2_14], for example, contains the corium ejection models developed following the PERCOLA programme. It should also be noted that most of the models developed as part of coolability studies have been implemented in the ANL CORQUENCH code [6.2_15] in order to test the coupling between MCCI phenomena and those associated with the presence of water on the surface of the corium pool using 1D ablation simulations. The simplified models for water penetration and corium ejection used in the first release of the CORQUENCH code now have been included in the latest release of MEDICIS [6.2_16]. More detailed models, of the corium ejection in particular, will be included later once they have been validated by sufficient experimental data on coolability phenomena.

6.2.3 SUMMARY AND OUTLOOK

As has been shown in this summary, the currently available tests results (general 1D and 2D tests, corium ejection tests, and water penetration tests) are not sufficient for it to be possible to arrive at a firm conclusion as to the effectiveness of cooling a corium pool during an MCCI by injecting water on top of the pool.

Progress in this area has been held back by the technological difficulties involved when carrying out tests using real materials (scaling effects, attachment of the crust, representativeness of the heating method used, etc.) and by the difficulty to perform large scale experiments.

Given these results and the difficulties in obtaining them, tests on a range of alternative mechanisms for stabilising the corium are planned. These mechanisms will be defined during the MCCI-2 programme, while studies will continue into the phenomena associated with corium cooling by aspersion with water.

6.3 CORIUM SPREADING (EPR)

6.3.1 DEFINITION AND OVERALL PHENOMENOLOGY

The development of a core catcher for the EPR has required a corium spreading R&D programme on a European scale. The objective of corium spreading is to avoid any breach in the basemat by facilitating corium cooling. The reduction in depth minimises the heat flux from the decay heat to be evacuated at the surface.

Studies into spreading should include an assessment of the ability of the corium to flow over a substrate of given geometry and composition, with the conditions under which the corium is injected onto the spreading surface being determined by the sequence of the accident. The key parameters are the composition of the corium and the substrate, the initial temperature and corium injection rate, as well as the geometry of the spreading surface. A review of work on this topic may be found in references [6.3_13] and [6.3_29].

6.3.2 PHYSICAL PHENOMENA INVOLVED

The spread of the corium is determined by the outcome of a competition between the hydrodynamic forces, which favour the progression and thinning of the melt, and the progressive solidification of the corium which, especially within the solidification interval, results in an increase in apparent viscosity and the appearance of crusts in contact with the substrate and on the surface of the pool.

The hydrodynamics of spreading have been studied by many authors [6.3_1], [6.3_2], [6.3_3], mainly in the field of volcanology. Numerical models and semi-analytical solutions have been developed for fluids whose properties remain constant during the flow. Spreading over a horizontal surface is a free surface flow in which the driving force is a function of the free surface slope. The flow during a severe accident are determined by gravity and inertia (at high speeds) or viscous friction (at lower speeds).

The rheology of the corium [6.3_26], [6.3_27] changes dramatically as it cools, especially below the liquidus temperature when crystalline phases begin to appear. It depends partly on the viscosity of the suspending liquid phase (a mixture in which the silicate ions increase the viscosity by forming lattices) which has been described by Urbain [6.3_4] and others, and partly on the effect of crystals solidifying under the shear forces associated with the flow. This type of complex semi-solid fluid has been described by Flemmings [6.3_5] and others, and an empirical formula has been proposed for corium [6.3_26], [6.3_30]).

The corium cools by losing heat at the surface through radiation and by losing heat to the substrate by convection. Crusts may form at both these interfaces, slowing the flow of the melt. However, a significant contact thermal resistance has been observed at the corium-substrate interface [6.3_6]. The effect of the residual power is low, given the short duration of the spreading phenomenon.

Griffiths and Fink [6.3_7] have published a comparative study of the various spreading models for solidifying lava as a function of the dominant forces (gravity-inertia, gravity-viscosity, gravity-complex rheology, gravity-strength of crust, etc.).

6.3.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

6.3.3.1 Experimental programmes

The first test programmes concerned with corium spreading in a reactor situation were carried out at Brookhaven in the USA, [6.3_8] with the aim of studying corium spreading at the bottom of the reactor pit in Mark I boiling reactors. In Europe, experimental tests and simulations of spreading were also carried out during the development and analysis of the core catcher in the EPR. Most of this work was carried out as part of the European COMAS [6.3_10], CSC [6.3_9] and ECOSTAR [6.3_11] projects.

The experimental programmes were divided between analytical experiments used to study the effect of each phenomenon individually (e.g. the CORINE programme [6.3_12], [6.3_13] carried out by the CEA in Grenoble with financial support from the IRSN), semi-analytical experiments using simulant materials, and tests using prototypical materials[3]. Tables 6.3-1 and 6.3-2 give the characteristics of the main test programmes carried out with either simulant materials or with prototypical corium (figure 6.3-1) shows an example of a corium spreading test.

Figure 6.3-1: Spreading of prototypical corium over ceramic ridges (left) and concrete ridges (right) [VULCANO test VE-U7]

These experimental programmes (especially CORINE, VULCANO and KATS) cover most of the parameters accessible to experimentation, including the geometry, material properties, and boundary conditions.

[3] Melts with a chemical composition identical to that expected during a severe accident, but with a different isotopic composition (e.g. using depleted or natural uranium).

Table 6.3-1: Experimental programmes on the spreading of simulant materials

Programme	Laboratory	Material	Scale (volume of the melt)	Geometry	Parameters
CORINE [6.3_12] [6.3_13]	CEA (France)	Low-temperature simulants (water, glycerol, low melting point metal alloys)	~50 litres	19° sector	- Flow rate (0.5 - 3 l/s) - Material (viscosity, single substance - non-eutectic mixture) - Cooling from the top or bottom - Effect of a flow of gas from the substrate
Greene [6.3_8]	BNL (USA)	Lead	~ 1 litre	Square section	- Mass - Additional heating - Depth of water
S3E [6.3_14]	KTH (Sweden)	Low and intermediate temperature (1200°C) simulants	5-20 litres	Rectangular channels Open geometries	- Flow rate - Additional heating - Material - Substrate (effect of concrete) - Dry/ in a liquid, with or without boiling
SPREAD [6.3_15]	Hitachi Energy Res. Lab. (Japan)	Steel	1-15 litres	Rectangular channel Half-disc	- Mass spread - Additional heating - Flow rate - Inlet geometry - Substrate - Depth of water
KATS [6.3_16] [6.3_17] [6.3_18]	FZK (Germany)	Aluminium thermite (Al_2O_3 + Fe) up to 2000°C	Up to 850 litres	Rectangular channel Quadrant	- Mass spread - Flow rate - Substrate - Effect of the addition of sacrificial materials - Phase spread (oxide and/or metal) - Reflooding

Table 6.3-2: Experimental programmes on the spreading of prototypical materials

Programme	Laboratory	Material	Scale (volume of the melt)	Geometry	Parameters
COMAS [6.3_10]	Siempelkamp (Germany)	Corium-concrete-iron mixtures Liquidus around 1900°C	20-300 litres	Rectangular channels 45° sector	- High flow rates (>150 kg/s) - Effect of silica - Substrate (ceramic/metal/concrete)
FARO [6.3_19]	JRC Ispra (European Commission)	UO_2-ZrO_2 Liquidus around 2700°C	~20 litres	19° sector	- Presence or absence of a thin layer of water - Metal substrate
VULCANO [6.3_20]	CEA (France)	UO_2-ZrO_2 + concrete products Liquidus between 1900 and 2700°C	3 - 10 litres	19° sector	- Flow rate - Composition - Type of substrate

These corium spreading experiments show that when corium flows are beginning to solidify, the liquid and solid phases remain in association, i.e. there is no macro segregation as occurs during slower transients. The solid fraction is therefore continuously varying within the flow. Also, in the case of a corium with a wide interval between the solidus and liquidus temperatures, a viscous skin (solid/liquid) appears on the surface, at least initially, rather than a solid crust. Conversely, in the case of more refractory coriums in which the solidus and liquidus temperatures are close, a solid crust forms at the surface which then cracks to allow the liquid to pass through. In this case, the phenomenology is strongly dependent on the scale of the flow, and the available experimental results are insufficient to describe fully the behaviour of the crust as they derive from small scale experiments with masses of injected water 1000 times smaller than would be the case in a real reactor. Ablation of the substrate (in the case of concrete) during the flow remains low. Its effect on the spreading kinetics has been shown but it remains limited.

It is considered that the available knowledge relating to the dry spread of corium in the most likely case of an extended solidus-liquidus temperature interval resulting from a composition rich in low-refractory materials originating in sacrificial concrete is sufficient to validate the simulation codes and to extrapolate the results to a real reactor.

In the case of corium spreading under water, the CORINE tests using simulants and with a depth of water of around 10 cm indicate that a layer of corium would accumulate until it reached the same depth as the water layer.

6.3.3.2 Modelling and simulation codes

A number of mechanistic simulation codes have been developed in Europe to model corium spreading. Table 6.3-3 shows their main characteristics. These codes have been subject to an intensive validation programme. Comparison benchmark tests have been carried out as part of the VULCANO VE-U7 prototypical corium test [6.3_25], and ECOKATS-1 was a blind test using simulant materials [6.3_17]. These exercises showed a good estimate of the spread surface area, with an uncertainty of around 20%.

Table 6.3-3: The main corium spread simulation codes

Code	Origin	Geometry	Characteristics	Validation
MELTSPREAD [6.3_21]	ANL on behalf of EPRI (USA)	1D	- Ablation of the substrate and oxidation of the corium taken into account	- Mainly via the Greene's tests
THEMA [6.3_13]	CEA (France)	2D integrated along the vertical axis	- Solidification of the corium (in the mass/in the crust) and melting of the substrate taken into account - 3D solution of the heat equations in le substrate	- Analytical solutions - Analytical tests - Tests using simulants and prototypical corium
LAVA [6.3_22]	GRS (Germany)	2D integrated along the vertical axis	- Cellular automaton - Focus on the cooling of the corium and its rheology (Bingham)	ditto
CROCO [6.3_23]	IRSN (France)	2D horizontal + vertical	- Detailed modelling - Free surface calculated using an Eulerian grid in association with a Lagrangian model of the fluid.	ditto
CORFLOW [6.3_24]	FZK (Germany)	3D	- Detailed modelling - Free surface represented by a height function verifying the kinetic and continuity equations	ditto

In addition, KTH in Sweden has developed a simplified analytical model which has been validated satisfactorily (mean accuracy of around ± 50%) [6.3_14], [6.3_29].

In parallel with this corium spread modelling work, an R&D programme [6.3_26] has been studying the rheology of corium during solidification under shear forces in a spreading situation. This has enabled the viscosity models applicable to liquids with and without silica to be extended to corium. It has also been able to predict the viscosity of semi-solid corium as a function of its solid fraction by volume with sufficient accuracy to be used in spreading calculations [6.3_26], [6.3_28].

6.3.4 SUMMARY AND OUTLOOK

The R&D programme has answered the question of dry spreading. Among others, the VULCANO tests have shown that, even when the initial temperature of the corium-concrete mixture is 100 to 200 C° below the liquidus temperature, it will still spread satisfactorily provided that the flow rate of the melt is sufficient. The presence of a shallow layer of water (simulating condensation) or a concrete substrate (releasing steam and CO_2) only affects the spreading slightly.

The flow mechanisms under a significant depth of water are determined by the mechanical behaviour of the crusts on the surface and on the melt front, and by the fragmentation of the corium. These mechanisms require further investigation before validated models become available.

6.4 EX-VESSEL CORE CATCHER

6.4.1 TYPES OF EX-VESSEL CORE CATCHER

From the arrival of the very first nuclear reactors, very many studies have been carried out into schemes for catching corium. A summary of these is given in [6.4_1] and [6.4_2]. It is not possible to describe all these systems in this document. The discussion will be limited to a number of generic aspects with particular reference to the EPR design.

6.4.1.1 Main safety functions

From a reactor safety point of view, core catcher systems must fulfil a number of specific functions:

- Corium retention: The system must be able to hold the entire mass of corium coming from the reactor.
- Containment of fission products: The system must retain all the refractory fission products remaining in the corium.
- Minimal production of combustible gasses (H_2 and CO): The aim is to prevent the generation of hydrogen by oxidation of metals (residual zirconium and steel) which may come into contact with water.
- Long term stabilisation of the corium: The system must retain and cool the corium over the long term.
- Resistance to dynamic effects (e.g. from a steam or hydrogen explosion, or from structural collapse).

It is difficult to satisfy all these constraints simultaneously and the design of these systems is therefore relatively complex. As a result, demonstration of the correct operation of these systems is difficult, and considerable R&D effort is required.

6.4.1.2 The EPR core catcher

In the EPR design, the corium is first spread over a large area. It is then flooded and cooled by water from the IRWST (see Figure 6.4-1).

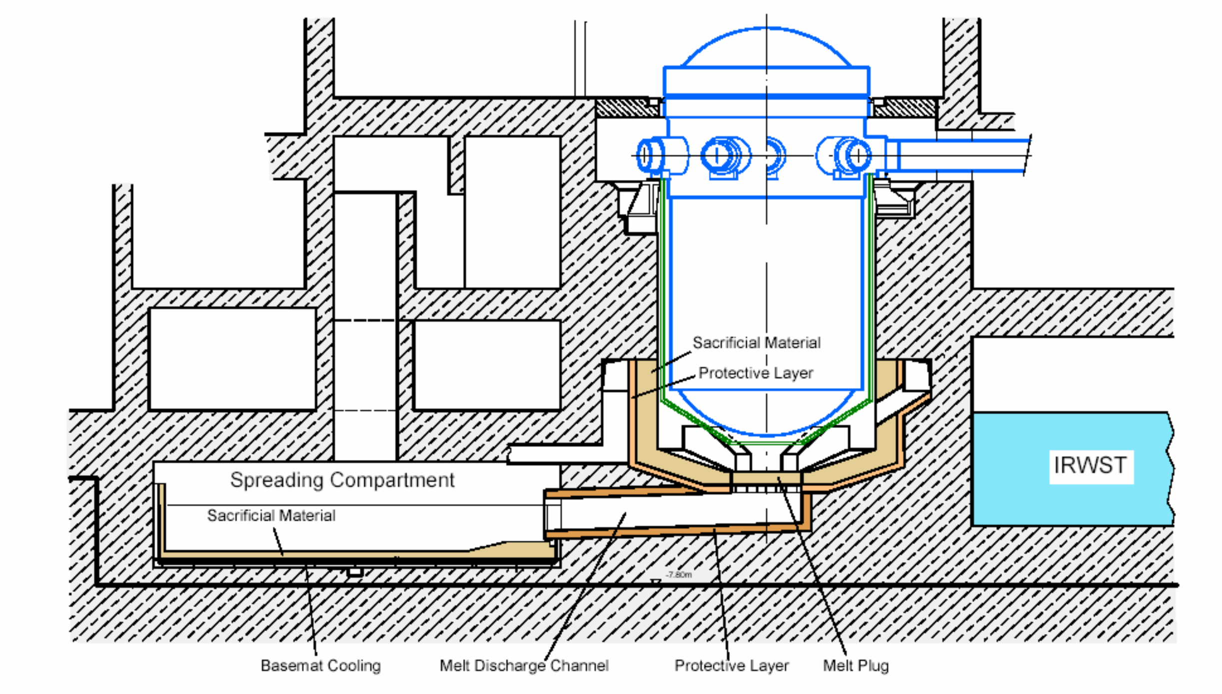


Figure 6.4-1: The main components of the EPR core catcher

The EPR design includes a preliminary phase of corium retention in the reactor pit in order to favour the spreading of the corium. During this phase, the corium must first erode through a 50 cm thick layer of sacrificial concrete before it can leave the reactor pit. This temporary retention and the addition of concrete to the corium increase the uniformity of the pool prior to the spreading phase. The uncertainties inherent in the formation of a corium pool within the vessel and those surrounding the breakdown of the vessel and the number of corium streams released are thus eliminated before the corium is spread and measures taken to stabilise it. A melt gate is also fitted at the bottom of the reactor pit below the sacrificial concrete. This gate provides access to a transfer channel linking the reactor pit to the spreading compartment. Away from this gate, the sacrificial concrete layer in the reactor pit is reinforced by a protective 20 cm thick layer of refractory material. The gate is intended to be a weak point, as it is the only area where the sacrificial concrete is not reinforced by a protective layer. It therefore breaks down rapidly in contact with the corium following ablation of the sacrificial concrete. It then provides an opening sufficiently large to enable the entire corium pool to pass rapidly into the spreading compartment.

The spreading compartment is a crucible with a surface area of around 170 m^2. The floor and side walls are made from a large number of separate cast iron elements. This structure is relatively immune to thermal expansion effects and distortion due to large temperature gradients. The lower cast iron sections include fins forming a number of rectangular horizontal cooling channels. The inner surface of the crucible structure is covered with a layer of sacrificial concrete. The arrival of corium in the compartment triggers the opening of valves allowing water to flow from the IRWST under gravity. This water begins to fill the horizontal channels under the spreading compartment. It then rises through the walls and eventually refloods the corium from above. This system is shown in Figure 6.4-2.

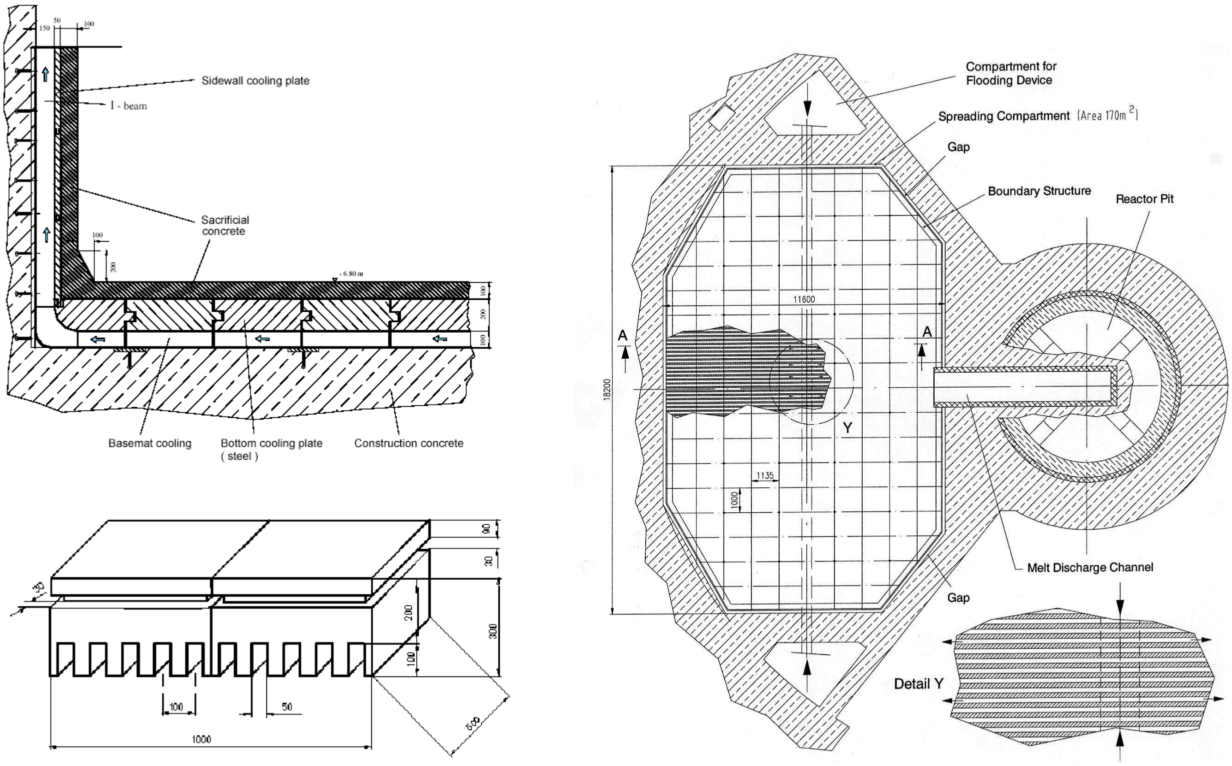


Figure 6.4-2: System of cooling fins in the EPR core catcher

6.4.2 PHYSICAL PHENOMENA INVOLVED

A successful demonstration of the correct operation of a core catcher design requires an understanding of a number of processes, together with the ability to predict their outcomes. These include:

- The interaction of the corium with the sacrificial concrete and the protective refractory ceramic material during the period of retention in the reactor pit.
- The opening of the melt gate (EPR).
- The spreading and flooding of the corium, together with control of the risk of a steam explosion.
- Long term cooling of the corium (in the spreading compartment in the case of the EPR).
- Long term containment of fission products.

6.4.2.1 Temporary melt retention in the reactor pit

The effect of the corium pouring into the reactor pit

An unknown quantity of corium pours into the reactor pit immediately following the breakdown of the reactor vessel. This implies a scenario in which a jet of corium hits the sacrificial concrete in the reactor pit, possibly resulting in significant ablation of this concrete.

Interaction between the corium and the sacrificial concrete

Once in the reactor pit, the corium begins to erode the sacrificial concrete. This ablation must proceed sufficiently slowly to allow all the corium likely to come from the core to accumulate before the melting of the gate allowing the corium to flow towards the spreading compartment. The spreading process must take place as a single event. The addition of concrete to the corium during this interaction must alter the properties of the corium in such a way as to enable it to spread satisfactorily, regardless of the initial conditions (enthalpy, solid fraction, viscosity, etc.). It is therefore essential to be able to predict accurately the phenomena associated with the molten corium-concrete interaction (see Chapter 5) and the thermophysical properties of the corium-concrete mixture in order to specify the correct type of concrete to be used and its thickness.

Interaction between the corium and the protective layer

In the case of the EPR, the protective layer is intended to protect the load-bearing concrete structure of the reactor pit in the event of a local breach in the sacrificial concrete. Other core catcher designs used in other reactor types also use refractory ceramic materials to contain the corium. The material in the protective layer must not be eroded, regardless of the composition of the corium or its configuration (stratification of the metal and oxide layers).

6.4.2.2 The opening of the melt gate (EPR).

In its initial state, the gate consists of a sacrificial concrete cover with the same thickness as the sacrificial concrete layer lining the reactor pit, together with a 4 cm thick steel plate. The corium first has to erode the sacrificial concrete layer, then melt the steel plate. Analysis has shown that, given a relatively homogeneous ablation front, the contact area between the corium and steel plate will be considerable and the plate should melt very quickly. However, in the event of a non-homogeneous ablation front (which cannot be excluded given the large uncertainties regarding MCCI phenomena (see Chapter 5)), there is a risk that there will be only a small area of contact between the corium and the steel plate, which could limit the size of the resulting breach through which the melt needs to flow. This analyse therefore indicates the need for further R&D work towards a new design for the melt gate.

6.4.2.3 Corium spreading and flooding

Once the gate is destroyed, the corium flows along the discharge channel and into the spreading compartment where its arrival triggers the release of water into the cooling system (see Section 6.4.1). All the corium must have spread before the water reaches the spreading compartment as the flow of corium under water is subject to considerable uncertainties and cannot be fully modelled at the present time (see Chapter 6.3). As the water is flowing through the system, the spread corium is interacting with the sacrificial concrete layer of the spreading compartment.

6.4.2.4 Long term cooling (EPR)

By the time that the interaction between the corium and the sacrificial material has stabilised on the cooled structure consisting of the cast iron elements traversed by around 360 rectangular horizontal channels, its surface has already been completely flooded. It is therefore cooled from above by direct contact with water, from the sides and from below via the cooling structure. It is essential to ensure that, under these conditions, the heat flux released by the corium may be evacuated and that there is no risk of heat damage to the structure (drying, followed by meltdown).

The cooling system must be passive, at least in the short term. This cooling is designed to be by natural convection. In order for the corium to be cooled effectively, the temperature must be kept below a critical value at all points on the cooled surface.

6.4.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

6.4.3.1 Experimental programmes, modelling and results

6.4.3.1.1 Temporary retention in the reactor pit: the effect of the corium pouring into the reactor pit

The consequences of the impact of jets of corium on a concrete surface have been studied as part of the KAJET tests [6.4_14]. These tests covered both oxide and metal coriums and various types of concrete. These experiments have shown that severe ablation can only occur with small diameter and high flow rate jets over a very long period of time. The impact of gravity driven superheated metal flows produced when a vessel ruptures under low pressure (1 to 2 bar) have also been investigated during the COMET tests [6.4_10]. These tests involved the pouring of superheated metals onto concrete from a height of several metres. These experiments showed that no significant ablation occurred.

6.4.3.1.2 Interaction between the corium and the sacrificial concrete

The problem of modelling the molten corium-concrete interaction has already been discussed in Chapter 5.

In the case of a core catcher system, the R&D work has shown that there are advantages in the use of a sacrificial material that can provide a number of additional functions, including:

- The oxidation of the corium metals without the production of any combustible gasses. Iron oxide is used to oxidise the residual Zr (producing ZrO_2 and Fe) without the generation of any hydrogen.
- The fluidisation of the corium by the addition of constituents that reduce the liquidus temperature and viscosity, thus facilitating the initial flow and spread, and the later cooling (e.g. concrete containing silica and iron oxide in the reactor pit of an EPR).
- Assisting containment as the use of silica (SiO_2) in the sacrificial material eventually forms a vitreous matrix capable of containing fission products.

6.4.3.1.3 Corium-ceramic interaction

This problem is discussed in a generic manner with particular reference to the EPR.

Considerations relating to the ability of a refractory material to withstand corium

In order to contain the corium, the simplest idea would appear to be to choose a refractory material capable of withstanding corium (temperature and chemical compatibility) and simply retain the corium in a ‘hot crucible’. The problem is that such a material does not exist. The ceramic oxides are dissolved by the corium oxide and are attacked by metals in the presence of oxygen. Refractory metal walls (tungsten, molybdenum, etc.) are attacked by metallic corium and by water. Carbides and nitrides also react with the corium.

Corium-ceramic interaction in the absence of concrete over the long term (> 1 week)

Zirconia (ZrO_2, melting point 2700°C) was suggested for use in the spreading compartment in the initial design of the EPR. The CEA carried out a study into the interaction between corium and zirconia. A model for the dissolution of zirconia by corium was proposed and validated by a series of interaction tests performed by NITI in Saint Petersburg [6.4_3] (Figure 6.4-3). The CEA has also shown that the presence of a metal layer between the corium oxide and the zirconia does not prevent the transfer of oxygen and the resulting ablation of the zirconia [6.4-19].

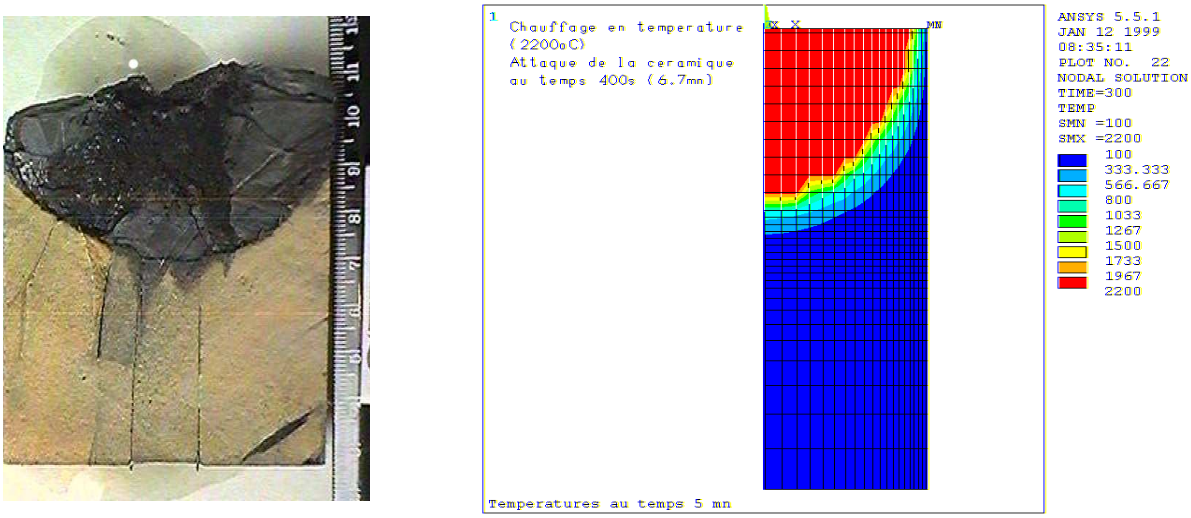


Figure 6.4-3: Corium-ceramic interaction: Left: Results of a test on a zirconia ceramic (light area) dissolved by corium oxide (dark area). Right: Reprocessed results from the same test.

These tests have shown that the zirconia could be dissolved by the corium over the long term (a week or more). Under these conditions, the interface temperature between the ceramic and the corium tends in the long term towards the liquidus temperature corresponding to the composition of the liquid corium oxide. This interface temperature would be the same even if the corium was contained within a solid crust. It has been concluded that there is no advantage in using costly refractory materials to contain the corium in the long term, and the solution is to contain the corium within its own solid crust (i.e. a 'self-crucible'). This has the advantage of reducing the corium solidification time to a minimum by reducing the mass accumulated in the core catcher. A low cost sacrificial material, such as high-alumina refractory concrete, is used to absorb thermal shocks. The sacrificial material is enclosed within a metal structure cooled from the outside. The ablation of the sacrificial material, which is less refractory than ceramic, results from structural disintegration due to the emission of gasses and thermal decomposition due to the melting of some of the constituents. This process takes place at a faster rate than the ablation of a ceramic limited by the dissolution kinetics.

Short term (<1 day) interaction between the corium and a ceramic in the presence of concrete (EPR reactor pit)

Small-scale tests [6.4_15] carried out recently by Framatome Erlangen have shown that, during a simultaneous interaction between corium, concrete and a zirconia ceramic, there is practically no ablation of the zirconia. This effect appears to be due to the fact that the corium-concrete interaction holds the temperature to a value close to or less than the liquidus temperature of the liquid corium oxide. This is in contrast to the long term interaction discussed above in which the temperature may exceed the liquidus temperature. At temperatures close to or below the liquidus temperature, the dissolution of the ceramic is no longer possible. The conclusion of these studies reveals that, during the corium-concrete interaction in an EPR reactor pit, the attack on the ceramic should be low and the melt gate should open before the protective zirconia layer is breached.

6.4.3.1.4 The opening of the gate (EPR).

The rapid and homogeneous melting of the steel plate cannot be guaranteed if the ablation of the sacrificial concrete proceeds in a non-homogeneous manner, as this implies that the contact surface area between the corium and steel plate would be relatively small.

The KAPOOL [6.4_17] series of experiments was carried out by FZK in Germany with the aim of studying the speed of ablation of a metal plate under these conditions. In these experiments, the oxide and metal phases were simulated by aluminium oxide and the iron produced in a thermite reaction. Three types of situation were considered: the interaction between the metal corium (iron) and a steel plate, the interaction between the corium oxide (aluminium oxide) and a steel plate, and the interaction between the corium oxide and an aluminium plate.

These experiments have shown that the steel plate was rapidly melted by the metal corium. However, when an oxide corium was used in the KAPOOL experiment, a crust formed at the interface between the corium and the plate which prevented the steel plate from melting. The aluminium plate did, however, melt due to its lower melting point. Studies carried out by the IRSN [6.4_16] have also shown that a 4 cm thick steel plate cannot be guaranteed to melt in all cases in such a way as to open a passage sufficiently large for all the corium to migrate to the spreading compartment before the arrival of the water.

In view of these results, it has been suggested that the steel plate should be replaced by an aluminium plate reinforced by a steel grid. The aluminium will ensure that a sufficiently large opening occurs almost immediately while the steel grid provides mechanical stability for the assembly. The cross-sectional area of the breach in the door will then no longer be limited by the metal door and will be determined solely by the size of the breach in the concrete.

6.4.3.1.5 Corium spreading and reflooding

The research work in relation to the spreading problem has been discussed in Section 6.3. This work has shown that the corium spreads satisfactorily over a dry surface providing that its temperature and flow rate are sufficiently high. The flow rate will be high enough if the size of the breach in the concrete above the door is greater than around 25 cm at the time that the melt is released. This initial breach size requirement is completely consistent with the spatial size of the instabilities that appear during corium-concrete ablation (M3B test using two tonnes of corium).

6.4.3.1.6 External cooling of the core catcher and long term stabilisation

The stability of a self-crucible is maintained by external cooling. The cooling system must be passive, at least in the short term (a few days). Spreading the corium in an EPR reduces the heat flux to be evacuated. This flux rapidly falls below around 0.1 MW/m^2. A heat flux of 0.1 MW/m^2 is relatively easy to evacuate by natural convection at a vertical or upward facing surface. At this type of surface, the critical flux is around 1 MW/m^2. The problem is to evacuate the heat flux from underneath the core catcher, especially when the surface area to be cooled is large as is the case in the EPR. In this configuration, steam can accumulate under the metal structure and prevent it cooling. In order to avoid this problem, the EPR makes use of specific design features to increase the heat transfer and to evacuate the steam. These are a system of fins similar to those used in the Vapotron [6.4_4]. In order to be effective, the fins must be sufficiently long and with a sufficiently high thermal conductivity to extract the heat flux from the area occupied by the steam, while maintaining the temperature low enough not to compromise its structural integrity (<~ 600°C). This system has been implemented in the EPR core catcher (see Figure 6.4-2). Demonstration tests have been carried out by AREVA using the Benson loop [6.4_5] and [6.4_6].

The VOLLEY test programme [6.4_18] was subsequently carried out by the University of Lappeenranta in Finland using two cast iron channels through which flowed either pure water or borated water. In all, ten tests were carried out, with the main parameters being varied including the inclination of the channels (horizontal or inclined at 1°), the flow rate of the water, and the heating power.

All these experiments confirmed the correct operation of the EPR cooling system, including under circumstances where there is partial flooding of the channels due to a steam-water counter-current inside them, i.e. when the cooling water is supplied from the pool of water formed above the corium.

6.4.3.1.7 Behaviour of fission products

A this stage of an accident, only the refractory fission products remain in the corium. The ACE tests [6.4_7] have shown that the release of fission products remains low, even when a gas is percolating through the corium. As soon as the corium is covered with water, it forms a solid crust at the surface which prevents the transfer of most of the fission products.

6.4.3.1.8 Steam explosion

The MACE and OECD CCI tests [6.4_9] have shown that reflooding the corium oxide with water does not result in a steam explosion. The ANAIS tests [6.4_8] have also shown no sign of a steam explosion when a layer of liquid steel is reflooded with water. A solid crust forms in less than a minute at the interface between the corium and the water, and this prevents a steam explosion. No steam explosion has occurred during the tests carried out to validate the design of the COMET core catcher in which water is injected from below into the corium under pressure following the melting of release mechanisms under the core catcher [6.4_10](Figure 2.2-1).

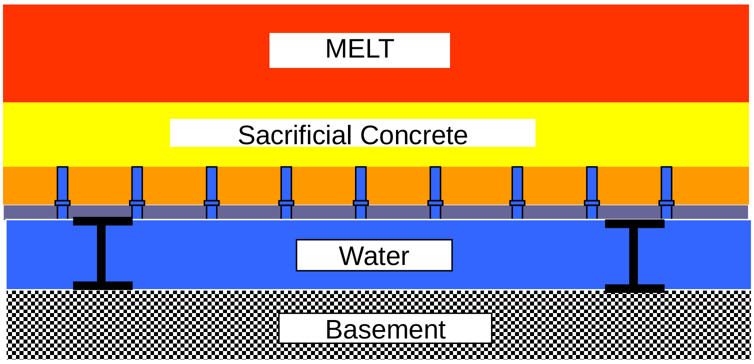


Figure 6.4-4: Operating principle of the FZK COMET core catcher: The ablation of sacrificial concrete by the corium pool causes the tips of the jets to melt injecting water into the corium.

6.4.3.1.9 Pressurisation of the containment

The reflooding of the corium with water releases steam which may increase the pressure within the reactor containment. If the power extracted during the transient is limited by a thermal resistance (e.g. when a crust forms at the surface of the corium in contact with the water, as in the case of the EPR core catcher), that rate of steam generation remains low and there is no significant increase in the pressure within the containment. However, the time taken for the corium to solidify completely is then longer (several days in the case of the EPR). The injection of water into the corium (as in the COMET design shown in Figure 6.4-4) causes the corium to solidify much more quickly (in less than an hour following the water injection) but it does increase the pressure in the containment. In this case, it has to be demonstrated that measures to control the pressure within the containment, such as aspersion, cooling or depressurisation through filters, allow to limit the increase in pressure to an acceptable value.

6.4.3.2 Computer codes

Given the wide variety of core catcher designs, there is no single code capable of describing all their operations. A number of tools are available to predict core catcher behaviour, including:

- Thermodynamic tools, used to predict the interactions between the corium and the constituent materials of the core catcher.
- Tools describing the external cooling of the core catcher walls (CATHARE, etc.).
- Tools describing the interaction between the corium and the water (MC3D).
- Tools linking a description of the corium to its interaction with the constituent materials of the core catcher. TOLBIAC [6.4_11] may be used to describe the interaction between corium and a ceramic. The distribution of the heat flux is essentially independent of the external cooling as the interface temperature between the liquid oxide corium (where most of the residual power is released) and the solid envelope is fixed (Tliquidus) [6.4_12]. This allows the ‘corium' heat flux distribution software to be used independently of the external cooling software. The current

release of the MEDICIS code [6.4_13] can be used to describe the corium-concrete interaction in which the concrete is eroded by thermal decomposition with the emission of gas, but it cannot be used to investigate the corium-ceramic interaction as this requires account to be taken of the dissolution of the ceramic and the conduction of heat into the ceramic substrate due to the slow speed of the interaction.

6.4.4 SUMMARY AND OUTLOOK

In general, a greater understanding of the following points is needed regardless of the design of the ex-vessel core catcher and its present state of development (industrial optimisation or assessment of the residual risks):

- Relocalisation, fragmentation and dispersion of the corium following a breach in the reactor vessel.
- Corium-ceramic interaction, or simultaneous corium-concrete-ceramic interaction.
- Corium-concrete interaction under water.
- Corium spreading under water.

TABLES IN CHAPTER 6

FIGURES IN CHAPTER 6

REFERENCES IN CHAPTER 6

[6.1_1] Nuclear Technology, "Edition spéciale consacrée à TMI2", 87, 1989.

[6.1_2] "Three Mile Island reactor pressure vessel investigation project", Proc. of an Open Forum Sponsored by OECD NEA and USNRC, 1993.

[6.1_3] NUREG/CR-6849, "Analysis of in-vessel retention and ex-vessel fuel coolant interaction for AP1000", 2004.

[6.1_4] Seiler J.M., "Analytical model for CHF in narrow gaps in vertical and hemispherical geometries", Accepted for publication in Special Issue of Nuc. Eng. Des. On Debris Coolability, 2006.

[6.1_5] Seiler J.M., Froment K., "Material effects on multiphase phenomena in late phase of severe accidents o f nuclear reactor", Multiphase Science and Technology, 12, 2000.

[6.1_6] Cheung , "Limiting factors for external vessel cooling", 10th Int. Topical Meeting on Nuclear Reactor Thermalhudraulics (NURETH 10), 2003.

[6.1_7] Seiler J.M., Tourniaire B., Defoort F., Froment K., "Consequences of physico-chemistry effects on in-vessel retention issue", 11th Topical Meeting on Nuclear Reactor Thermalhydraulics (NURETH 11), 2005.

[6.1_8] Amblard M., Froment K., Seiler J.M., Tourniaire B., "ANAIS experiment: consequence of water injection on a molten metal layer in the lower head", 10th Int. Topical Meeting on Nuclear Reactor Thermalhydraulics (NURETH 10), 2003.

[6.1_9] Bonnet J.M., Seiler J.M., "In-vessel corium pool thermalhydraulics for the bounding case", Salt Expert Group Meeting RASPLAV Seminar, 2000.

[6.1_10] Laporte T., "Synthèse des études réalisées au DMT sur les conséquences d'une explosion de vapeur en cuve ou hors cuve en cas d'accident grave de REP", CEA/DEN/DMT/LM2S/RT/00-039, 2000 - Non-public reference.

[6.1_11] Henry E.H., Hammersley R.J., 1997, "An experimental investigation of possible In-Vessel cooling mechanisms", FAI note I:\HPA\97-1 - Non-public reference.

[6.1_12] Kang K.H., Park R.J., Kim S.D., Kim H.D., 2005, "Simulant pool experiments on the coolability through external vessel cooling strategy ", ICAPP 05 conference Seoul, Korea, May 15-19.

[6.1_13] Asmolov V., Kobzar L., Nickulshin V., Strizhov V., 1998, "Experimental study of heat transfer in the slotted channel at CTF facility", OECD/CSNI Workshop on In-Vessel Core Debris Retention and Coolability, Garching, Germany, March 3-6.

[6.1_14] Köhler W., Schmidt H., Herbst O., Krätzer W., 1998 "Thermohydraulische Untersuchungen zur Debris/Wand-Wechselwirkung (DEBRIS)", Abschlussbericht Project N° 150 1017, November 1998 - Non-public reference.

[6.1_15] Jeong J.H., Park R.J., Kang K.H., Kim S.B., Kim H.D., 1998, "Experimental study on CHF in a Hemispherical Narrow Gap", OECD CSNI Workshop on In-Vessel Core Debris Retention and Coolability Garching, Germany, March 3-6 1998.

[6.1_16] AbalinS.S., Gnidoi I.P., Surenkov A.I., Strizhov V.F. 1998, "Data base for 3rd and 4th series of RASPLAV salt tests", OECD RASPLAV Report.

[6.1_17] Roux P., Fichot F., De Pierrepont S., Gobin D., Goyeau B., Quintard M., 2005, "Modelling of binary mixture phase change: assessment on RASPLAV Salt Experiments", NURETH 11, Avignon, 2-6 October 2005.

[6.1_18] Rougé S., 1995, "SULTAN test facility; Large scale vessel coolability in natural convection at low pressure", NURETH 7, Saratoga Springs, September 10-15, 1995.

[6.1_19] Dinh T.N., Salmassi T., Theofanous T.G. "Limits of coolability of the AP 1000 ULPU 2400 configuration V facility", CRSS-03/06, June 30, 2003.

[6.1_20] Seiler J.M., Fouquet A., Froment K.F., Defoort F., 2003, "Theoretical analysis for corium pool with miscibility gap", Nuclear Technology Vol 141, N°3, pp 233-243.

[6.1_21] Rempe J.L., Suh K.H., Cheung F.B. "Insights from investigations of In-Vessel retention for high powered reactors", NURETH 11, Avignon October 2-6 2005.

[6.1_22] Seiler J.M., Ducros G., 2005, "Reflooding of a PWR core assessment of knowledge and R&D needs. Potential contribution by PHEBUS", DEN/DTN/SE2T/LPTM/05-117 - Non-public reference.

[6.1_23] Atkhen K., Berthoud G., 2003, "Experimental and numerical investigations based on debris bed coolability in a multidimensional and homogeneous configuration with volumetric heat source", Nuclear Technology Vol 42, June 2003.

[6.1_24] Van Dorsselaere J.P. et al., 2000 "Programme Renoyage : Etat des connaissances sur le renoyage d'un cœur dégradé", NT SEMAR 2000/093, IRSN DPAM - Non-public reference.

[6.1_25] Van Dorsselaere J.P. et al., "Proposition d'un programme de R&D sur le renoyage d'un cœur, Rev 1", NT SEMAR 2002/047, IRSN DPAM - Non-public reference.

[6.2_1] Levy S., "Summary of coolability studies undertaken by ACE, MACE, ACEX", 2002 - Non-public reference.

[6.2_2] Farmer M.T., Kilsdonk D.J., Aeschliman R.W., "MSET-1 test data report", MACE-TR-D18, 2002 - Non-public reference.

[6.2_3] Farmer M.T., Lomperski S., Basu S., "Results of reactor material experiments investigating 2D core-concrete interaction and debris coolability", Proc. of 4th Int. Conf. on Advances in Nuclear Power Plants, 2004.

[6.2_4] Tourniaire B., Seiler J.M., Bonnet J.M., Amblard M., "Experimental study and modelling of liquid ejection through orifices by sparging gas", accepted for publication in Nucl. Eng. Des. Special Issue on Debris coolability, 2006.

[6.2_5] Epstein M., "Review of water ingression model", EPRI ACEX-TR-C31 Report, 1999 - Non-public reference.

[6.2_6] Seiler J.M., Cranga M., "Evaluation of CHF power extraction limits in corium cracks during a corium pool cooling", Rapport CEA/DEN/DTP/SETEX/LTEM/01-244, 2001 - Non-public reference.

[6.2_7] Farmer M.T., Lomperski S., Basu S, Status of Pool Coolability and Concrete Interaction (MCCI) Program at Argonne National Laboratory, Proceedings of ICAPP05, Seoul, Korea, May 15-19th 2005.

[6.2_8] Farmer M.T., Sienicki J.J., Kovtonyuk A., Basu S., "Water ingression modelling of the OECD/MCCI SSWICS tests", NRC/MCCI/2003-TR01, 2003 - Non-public reference.

[6.2_9] Bonnet J.M., Seiler J.M., "Coolability of corium spread onto concrete under water, the PERCOLA model", Proc. Of the 2nd OECD (NEA) CSNI specialist meeting on Core Debris-Concrete Interaction, 1992.

[6.2_10] Tourniaire B., Seiler J.M., "Modeling of viscous and inviscid fluid ejection through orifices by sparging gas", Proc. of 4th Int. Conf. on Advances in Nuclear Power Plants, 2004.

[6.2_11] Ricou F.P., Spalding D.B., "Measurements of entrainment by axisymetrical turbulent jets", Journal of Fluids Mechanics, 1961.

[6.2_12] Tourniaire B., "Application of the PERCOLA ejection model to reactor scenario of molten core concrete interaction under water", Proc. of 5th Int. Conf. on Advances in Nuclear Power Plants, 2005.

[6.2_13] Farmer M.T., "Status of MCCI data analysis and modeling activities", OECD/MCCI Program Review Meeting, Argonne National Laboratory, 2005 - Non-public reference.

[6.2_14] Spindler B., Tourniaire B., Seiler J.M., Atkhen K., "MCCI analysis and applications with the TOLBIAC-ICB code based on phase segregation model", Proc. of 5th Int. Conf. on Advances in Nuclear Power Plants, 2005.

[6.2_15] Farmer M.T., Spencer B.W., "Status of the CORQUENCH model for calculation of ex-vessel corium coolability by an overlying water layer", OECD Workshop on Ex-vessel Debris Coolability, 1999.

[6.2_16] Cranga M., Fabianelli R., Jacq F., Barrachin M., Duval F., "The MEDICIS code, a versatile tool for MCCI modelling", Proc. of 5th Int. Conf. on Advances in Nuclear Power Plants, 2005.

[6.3_1] Dragoni, M. , Bonafede, M, Boschi, E. , 1986, "Downslope flow models of a Bingham liquid: Implications for lava flows", J. Volcanol. Geotherm. Res., 30, 305-325.

[6.3_2] Huppert, H.E., 1982b, "The propagation of two-dimensional and axisymmetric viscous gravity currents over a rigid horizontal surface", J. Fluid Mech., 121, 43-58.

[6.3_3] Didden, N., Maxworthy, T., 1982, "The viscous spreading of plane and axisymmetric gravity currents", J. Fluid Mech. 121, 27-42.

[6.3_4] Urbain, G., 1987, "Viscosity estimation of slags", Steel Res. 58, 111-116.

[6.3_5] Flemmings, M.C., 1991, "Behavior of metal alloys in the semisolid state", Metall. Trans. 22B, 269-293.

[6.3_6] Berthoud, G., 2001, "Freezing of Pure Pool Flowing into Tubes", dans : Ehrhard, P., Riley, D.S., Steen, P.H., éditeurs, Interactive Dynamics of Convection and Solidification, Kluwer Acad. Publ., Dordrecht, NL, pp. 249-268.

[6.3_7] RW Griffiths, JH Fink, "Effects of Surface Cooling in the Spreading of lava flows and domes", J. Fluid Mech. 252, 1993, pp. 667-702.

[6.3_8] Greene, G.A., Finrock, C., Klages, J., Schwarz, C.E., Burton, S. B., 1988, "Experimental Studies on Pool Spreading, Bubbling Heat Transfer and Coolant Layer Boiling", Proc. 16th Water Reactor Safety Meeting, NUREG/CP-0097, 341-358.

[6.3_9] Cognet, G., Alsmeyer, A., Tromm, W., Magallon, D., Wittmaack, R., Sehgal, B.R., Widmann, W., De Cecco, L., Ocelli, R., Azarian, G., Pineau, D., Spindler, B., Fieg, G., Werle, H., Journeau, C., Cranga, M., Laffont, G., 2001, Corium spreading and coolability: CSC Project, Nucl. Eng. Des., 209, 127-138.

[6.3_10] Steinwarz, W., Alemberti, A., Häfner, W., Alkan, Z., Fischer, M., 2001, "Investigations on the phenomenology of ex-vessel core pool behaviour", Nucl. Eng. Des., 209, 139-146.

[6.3_11] Alsmeyer, H., Alberecht, G., Meyer, L., Häfner, W., Journeau, C., Fischer, M., Hellmann, S., Eddi., M., Allelein, H.-J., Bürger, M., Sehgal, B.R., Koch, M.K., Alkan, Z., Petrov, J.B., Gaune-Escart, M;, Altstadt, E., Bandini, G., 2005, "Ex-vessel core pool stabilization research (ECOSTAR) ", Nucl. Eng. Des., 235, 271-284.

[6.3_12] J.M. Veteau, R. Wittmaack, "CORINE experiments and theoretical modelling". In: G. Van Goetem, W. Balz, E. Della Loggia (Eds) FISA 95 EU Research on severe accidents, Office Official Publ. Europ. Communities, Luxembourg, 1996, pp. 271-285.

[6.3_13] Spindler, B., Veteau, J.M., 2004, "Simulation of spreading with solidification: assessment synthesis of THEMA code", Rapport CEA-R6053.

[6.3_14] Dinh, T.N., Konovalikhin, M.J., Sehgal, B.R., 2000, "Core Pool Spreading on a reactor Containment Floor", Progr. Nucl. Energ., 36, 4, 405-468.

[6.3_15] Suzuki, H., Matsumoto, T., Sakaki, I., Mitadera, T., Matsumoto, M., Zama, T., 1993, "Fundamental experiment and analysis for pool spreading on concrete floor", Proc. 2nd ASME/JSME Nucl. Eng. Conf., 1, 403-407.

[6.3_16] Engel, G., Fieg, G., Massier, H., Stiegmaier, U., Schütz, W., 1999, "KATS experiments to simulate corium spreading in the EPR code catcher concept", OECD Wkshp Ex-Vessel Debris Coolability, Karlsruhe, Allemagne, 15-18/11/1998.

[6.3_17] Spengler, C., Allelein, H.-J., Foit, J.-J., Alsmeyer, H., Spindler, B., Veteau, J.M., Artnik, J., Fischer, M., 2004, "Blind benchmark calculations for pool spreading in the ECOSTAR project", Proc. ICAPP '04 (Int. Conf. Advances in nuclear Power Plants), Pittsburg, PA, Communication n°4105.

[6.3_18] Alsmeyer, H., Cron, T., Messemer, G., Häfner, W., ECOKATS-2: "A Large Scale Experiment on Pool Spreading and Subsequent Cooling by Top Flooding", Proc. ICAPP ’04 (Int. Conf. Advances in nuclear Power Plants), Pittsburg, PA, Communication n°4134.

[6.3_19] Tromm, W., Foit, J. J., Magallon, D., 2000, "Dry and wet spreading experiments with prototypic materials at the FARO facility and theoretical analysis", Wiss. Ber. FZKA, 6475,178-188.

[6.3_20] Journeau, C., Boccaccio, E., Brayer, C., Cognet G., Haquet, J.-F., Jégou, C., Piluso, P., Monerris, J., 2003, "Ex-vessel corium spreading: results from the VULCANO spreading tests", Nucl. Eng. Des. 223, 75-102.

[6.3_21] Farmer, M.T., Sienicki, J.J., Chu, C.C., Spencer, B.W., 1993, "The MELTSPREAD-1 computer code for the analysis of transient spreading and cooling of high temperature melts", Rapport EPRI TR-103413 - Non-public reference.

[6.3_22] Allelein H.-J., Breest A., Spengler C., 2000, "Simulation of core pool spreading with LAVA: Theoretical background and Status of Validation", Wiss. Ber. FZKA, 6475, 189 – 200.

[6.3_23] Piar, B., Michel, B.D., Babik, F., Latché, J.-C., Guillard, G., Ruggierri, J.-M., 1999, "CROCO: A Computer Code for Corium Spreading", Proc. Ninth International Topical Meeting on Nuclear Thermal Hydraulics (NURETH-9), San Francisco, Ca., USA.

[6.3_24] Wittmaack, R., 1997, "CORFLOW: A code for the numerical simulation of free-surface flow", Nucl. Technol., 116, 158-180.

[6.3_25] Journeau, C., Haquet, J.-F., Spindler, B., Spengler, C., Foit, J., 2006, "The Vulcano VE-U7 corium spreading benchmark", Progr. Nucl. Energ., 48, 215-234.

[6.3_26] Ramacciotti, M., Journeau, C., Sudreau, F., Cognet, G., 2001, "Viscosity models for corium melts", Nucl. Eng. Des. 204, 377-389.

[6.3_27] Seiler, J.M., Ganzhorn, J., 1997, "Viscosities of corium-concrete mixtures", Nucl. Eng. Des. 178, 259-268.

[6.3_28] Perez, M., Barbé, J.C., Neda, Z., Bréchetr, Y., Salvo, L., 2000, "Computer simulation of the microstructure and rheology of semi-solid alloys under shear", Acta Met., 48, 3773-3782.

[6.3_29] Journeau, C., 2006, "L’étalement du corium: Hydrodynamique, Rhéologie et solidification d’un bain d’oxydes à haute température", Thèse de Doctorat, Université d’Orléans.

[6.3_30] Journeau, C., Jeulain, G., Benyahia, L., Tassin, J.-F., Abélard, P., 2006, "Rheology of mixtures in the solidification range", Rhéologie, 9, 28-39.

[6.4_1] Szabo I. "comparison of advantages and drawbacks of typical core-catcher concepts", WG2 meeting sept 6th 1994, CEA Grenoble - Non-public reference.

[6.4_2] Szabo I., "Survey of design principles and basic mechanisms of ex-vessel molten core retention system ", Rapport provisoire CEA - Non-public reference.

[6.4_3] Seiler J.M., Froment K., "Material effects on multiphase phenomena in late phase of severe accidents o f nuclear reactor", Multiphase Science and Technology, 12, 2000.

[6.4_4] Beurtheret C.A. "Le processus de vaporisation et le Vapotron", Institut Français des Combustibles et de l’Energie, et revue technique CFHT n°24 Paris December 1956.

[6.4_5] Schmidt H., "Ex-vessel pool stabilisation: Experimental results on CHF-limits in the channels of the basemat cooling system", Work Report FANP TGT1/02/e52 - Non-public reference.

[6.4_6] Schulze G., Klemm L., "Experimental demonstration of the heat removing capacity of the EPR core catcher, considering the impact of boron and insulation material", AREVA Technical report NGTT1/2005/en/1001 - Non-public reference.

[6.4_7] Cenerino G., Chevalier P.Y., Fischer E. 1992 "thermodynamic calculations of oxide complex systems, prediction of fission product releases", 2nd OECD CSNI Specialist meeting on Molten Core Concrete Interactions FZK Karlsruhe, Germany 1-3 April 1992.

[6.4_8] Amblard M., Froment K., Seiler J.M., Tourniaire B., "ANAIS experiment: consequence of water injection on a molten metal layer in the lower head", 10th Int. Topical Meeting on Nuclear Reactor Thermalhudraulics (NURETH 10), 2003.

[6.4_9] Farmer M.T., Lomperski S.W., Basu S. 2005, "The results of the reactor material experiments investigating 2D core-concrete interaction and debris coolability", NURETH 11, Avignon October 2-6 2005.

[6.4_10] Alsmeyer H., Adelheim C., Benz H., Cron T., Dillmann G., Tromm W., Schmidt-Stiefel S., Schneider H., Schumacher G., Wenz T., "Corium cooling by bottom flooding: results of the COMET investigations", OECD Workshop on Ex-Vessel Debris Coolability Karlsruhe, Germany, 15-18 November 1999.

[6.4_11] Spindler B., Tourniaire B., Seiler J.M., Atkhen K., "MCCI analysis and applications with the TOLBIAC-ICB code based on phase segregation model", ICAPP'5, 2005.

[6.4_12] Seiler J.M., Labergri, Froment K., Valin F., 1999, "Model development fort eh calculation of corium-ceramic interaction during severe accidents in LWRs" NURETH 9 San Francisco October 3-8, 1999.

[6.4_13] Cranga M., Fabianelli R., Jacq F., Barrachin M., Duval F., "The MEDICIS code, a versatile tool for MCCI modelling", ICAPP'5, 2005.

[6.4_14] Alsmeyer et al., "Ex-vessel core-pool stabilization research (ECOSTAR) ", Proceedings of FISA 2003, Luxembourg, Nov. 2003.

[6.4_15] Fischer, M., Hellmann, S, Bonnet, J.M., Journeau, C., Tourniaire, B., Seiler, J.M., Alsmeyer, H, Bürger, M., Park, S., "European experimental activities on the coolability of molten corium and its interaction with concrete and ceramic material", The first European Review Meeting on Severe Accident Research (ERMSAR-2005), Aix-en-Provence, France, 14-16 November 2005.

[6.4_16] "Safety assessment of the EPR Project", IRSN/DSR report n°34, EPR report n°88, Oct. 2004 - Non-public reference.

[6.4_17] Eppinger B. et al., "Investigations on the Pool Gate Ablation by Ex Vessel Core Melts in the KAPOOL Experiments", Proceedings of ICONE 10, Arlington, VA April 20902.

[6.4_18] Lane J., "Results of the EPR core catcher test ring tests 2005", Lappeenranta University of Technology Nuclear Safety Research Unit, V 2/2005 - Non-public reference.

CHAPTER 7 : RELEASE AND TRANSPORT OF FISSION PRODUCTS

7.1 IN-VESSEL RELEASE OF FISSION PRODUCTS

7.1.1 DEFINITION AND OVERALL PHENOMENOLOGY

The main features of a severe accident occurring in a nuclear reactor that set it apart from all other industrial accidents are the potential for the massive release of radioactive products into the environment and the associated radiological consequences. The main sources of such radioactivity are fission products and the first stage in preventing their release into the environment is to evaluate their release into the reactor vessel, often called the source term.

The fission products are created by irradiation and the yield depends on both the type of product itself and, to a second order[4], on the type of fission (thermal in the case of ^{235}U or ^{239}Pu, rapid in the case of ^{238}U etc.). The presence of stable and long radioactive half-life fission products causes the inventory to increase in a quasi-linear manner as a function of the burnup at a rate of around 75 kg/(GWd/t), i.e. 2 tonnes of fission products created in a PWR-900 core at equilibrium[5]. Table 7.1-1 provides a detailed summary of the inventory of each fission product and each heavy nucleus, together with the change in total activity of the core for a period of one month following the shut-down of a reactor.

Under nominal irradiation conditions within a PWR, the fission products are found within the fuel matrix in a number of chemical forms [7.1_1], including:

- Dissolved oxides, in the case of around half the products, especially Sr, Y, Zr, La, Ce and Nd.
- Oxide precipitates, mainly in the case of Ba and Nb.
- Metal precipitates, in the case of Mo, Tc, Ru, Rh and Pd.
- The chemical state of the volatile fission products (Br, Rb, Te, I and Cs) is not fully understood at the present time. The majority are probably present in the form of dissolved atoms and, above a given temperature, these may migrate radially before condensing in cooler areas. The formation of certain compounds, including CsI, Cs_2MoO_4 and caesium uranates, has often been suggested, but this has never been confirmed experimentally.
- Dissolved atoms or bubbles of gas located in inter or intra-granular positions, in the case of the fission gas, xenon and krypton. It should be noted that the accumulated fraction of gas at the grain boundaries is more likely to be released in the event of an accident.

---

[4] It should be noted however that in the case of some fission products such as ruthenium, the fission yield may vary by a factor of between two (^{103}Ru) and ten (^{106}Ru) depending on whether the fissions take place in ^{235}U or ^{239}Pu.

[5] Taken from a PEPIN simulation for a reactor at the end of its life with a quarter core loading (10.5 - 21 – 31.5 - 42 GWd/t) and 72.5 tonnes of initial uranium.

Table 7.1-1: Changes in the activities of fission products and actinides in a 900 MWe(1) PWR following shut-down

		Activities as a fraction of the total activity			
Fission products	**Total mass at SD in kg (2)**	**At SD**	**1 hour later**	**1 day later**	**1 month later**
As	7,39E-03	0,20%	0,01%	0,00%	0,00%
Se	3,14E+00	0,58%	0,02%	0,00%	0,00%
Br	1,16E+00	1,17%	0,20%	0,00%	0,00%
Kr	2,21E+01	2,32%	1,46%	0,03%	0,06%
Rb	2,03E+01	3,22%	0,84%	0,01%	0,00%
Sr	5,51E+01	4,50%	3,85%	2,57%	6,10%
Y	2,89E+01	5,84%	5,11%	3,40%	8,16%
Zr	2,10E+02	4,73%	3,83%	4,63%	10,30%
Nb	3,24E+00	7,09%	5,68%	5,93%	13,18%
Mo	1,84E+02	4,28%	2,28%	2,90%	0,01%
Tc	4,52E+01	4,82%	2,50%	2,77%	0,01%
Ru	1,37E+02	1,85%	3,11%	3,67%	10,27%
Rh	2,36E+01	2,30%	3,42%	4,96%	10,26%
Pd	5,93E+01	0,19%	0,33%	0,18%	0,00%
Ag	3,97E+00	0,14%	0,11%	0,12%	0,05%
Cd	4,00E+00	0,03%	0,02%	0,01%	0,00%
In	8,20E-02	0,13%	0,03%	0,01%	0,00%
Sn	2,65E+00	0,66%	0,15%	0,02%	0,01%
Sb	8,98E-01	1,76%	0,68%	0,17%	0,06%
Te	2,62E+01	3,85%	4,16%	2,88%	0,69%
I	1,27E+01	5,70%	8,94%	6,39%	0,65%
Xe	3,07E+02	4,33%	3,60%	5,12%	0,41%
Cs	1,61E+02	3,82%	1,27%	0,46%	1,61%
Ba	8,21E+01	4,67%	3,75%	3,46%	3,45%
La	6,99E+01	4,71%	5,22%	3,57%	3,25%
Ce	1,63E+02	3,61%	5,04%	7,41%	16,01%
Pr	6,21E+01	3,10%	4,63%	5,49%	11,76%
Nd	2,07E+02	0,68%	1,07%	1,25%	0,82%
Pm	1,24E+01	0,65%	1,22%	1,65%	1,48%
Sm	3,57E+01	0,21%	0,46%	0,54%	0,00%
Eu	8,90E+00	0,08%	0,19%	0,29%	0,36%
Actinides					
U	6,99E+04	9,37%	3,91%	0,00%	0,00%
Np	3,15E+01	9,37%	22,76%	29,86%	0,02%
Pu	5,89E+02	0,05%	0,11%	0,19%	0,80%
Am	6,18E+00	0,00%	0,00%	0,00%	0,00%
Cm	2,09E+00	0,01%	0,03%	0,06%	0,21%
Total activity (Bq)		5,91E+20	2,42E+20	1,39E+20	3,30E+19

(1) : 900 MWe UO_2 PWR with 3.70% of ^{235}U – 72.5 tonnes of U intitially
End-of-life loading per quarter core: 10.5 - 21 – 31.5 - 42 GWJ/tU
All fission products and actinides present in the core (assumed to be no release)
(2) : Stable and radioactive elements

The chemical state of the fission products in the first three categories is not fixed and some of them may move from one category to another according to the operating temperature, the oxygen potential in the fuel (which increases with burnup as fission reactions tend to be oxidising), and the burnup (which increases the concentration of fission products in the matrix). This is especially the case with molybdenum which precipitates mainly in metallic form, but which may also be found as the oxide (especially on the surface of MOX fuel pellets), and for niobium and strontium whose oxides may be partly dissolved and partly precipitated.

The fission products play a number of roles in the event of a severe accident:

- The radioactive fission products, especially those with short half-lives, make up a small proportion of the total mass, but account for most of the radioactivity and residual power.
  - The more volatile products are released from the reactor vessel, transported and partially deposited in the reactor cooling system and then in the containment building. From the containment building, they can reach the outside through leakage and contaminate the environment. The most radiotoxic of these isotopes are ^{133}Xe, ^{132}Te, ^{132}I and ^{131}I in the short term, and ^{134}Cs and ^{137}Cs in the long term. Ruthenium is also a potential hazard in specific accident situations where air enters the core (^{103}Ru in the medium term and ^{106}Ru in the longer term).
  - The non-volatile fission products (^{239}Np and ^{140}La in the short term, ^{95}Zr, ^{95}Nb and ^{144}Ce in the medium to long term) remain in the corium, causing the core to heat up and melt if it is not sufficiently cooled.
- The stable fission products account for most of the mass and they exacerbate the degradation of the core as they form eutectic mixtures with the UO_2, which tend to reduce the melting point [7.1_2]. The effect can be significant as the concentration of fission products in the fuel at high burnup can exceed 10% atomic above 50 GWd/t). The pressure of gaseous fission products can also break down the fuel at grain boundaries resulting in the formation of fuel debris.

7.1.2 PHYSICAL PHENOMENA INVOLVED

7.1.2.1 Fission gas

During normal irradiation in a PWR, the fission gas form at the atomic level within the grain structure of the UO_2. These gas atoms diffuse towards the grain boundaries or accumulate in the form of nanometric intragranular bubbles, slowing their rate of migration towards the grain boundaries. The bubbles may also be redissolved under the influence of fission spikes, accelerating the flow of gas to the grain boundaries. Once the fission gasses arrive at the grain surface, mainly though atomic diffusion but also by migration of the bubbles, the fission gas accumulate until they can coalesce to form larger bubbles filling the intergranular space. Some of this gas can find a path out of the fuel and into the free space in the fuel rod [7.1-3].

At the instant an accident occurs, the gas population therefore consists of three types, as follows:

- Atoms of gas dissolved in the matrix.
- Low intra-granular bubbles of gas.
- The gasses accumulated in the intergranular spaces which can be released via a number of different mechanisms.

The first phase in the release process, often referred to as 'burst release', corresponds to the <u>release of gas that have accumulated in the intergranular spaces</u>, together with the gas fraction that has already been released into the fuel rod plenum during normal operational irradiation (between a few % and 10% depending on the burnup, the irradiation power and the type of fuel). This release occurs at the beginning of the temperature rise at around 1000°C, although the temperature may be lower in the case of fuels with a high burnup. The second phase involves the release of the intragranular gas as a result of a thermally activated diffusion process, beginning with the dissolved atoms. The gas trapped in the nanometric intragranular bubbles are the last to be released, often only at the point at which the fuel melts.

When modelling the release of gas, it is therefore important to have a good quantitative assessment of the respective proportions of these three populations. These depend on their radial position within the pellet (different temperatures and rim type micro-structure at the edges, etc.) and on the type of fuel (the intergranular fraction is higher in the case of heterogeneous MOX fuels). It should also be noted that these fractions also depend on the radioactive half-life of the gas, which has a beneficial effect on the source term as short half-life gas have had time to decay after they have migrated to the grain boundaries. This effect is also essentially beneficial in the case of LOCA and RIA accidents, as the total inventory of the gas is eventually all released in the event of a severe accident.

7.1.2.2 Fission products

It is generally accepted that the release of fission products is a two-stage process: (i) Fission products in solution in the matrix, or in the form of a precipitate once the limit of solubility has been reached, diffuse towards the grain boundaries, and (ii) a mass transfer or vaporisation process carries the fission products out of the grain boundaries. The second of these stages has a number of physical and chemical aspects. These include the potential formation of defined compounds (CsI, molybdates, zirconates and uranates of caesium, barium, strontium, etc.) and the oxidation or reduction of precipitates by steam and/or hydrogen. These chemical reactions significantly affect the volatility of some elements. It should be noted that the basic high-temperature thermodynamic data relating to the formation and decomposition of these compounds is not well understood at the present time. This makes the mechanistic modelling of these processes difficult.

In addition to the release of fission products directly from the fuel matrix, potential chemical interactions with the cladding and/or structural components of the core may also reduce the volatility of some elements by the formation of more refractory compounds. Finally, even once they have escaped from the core, a significant proportion of the fission products condense in cooler areas within the structure of the upper head before reaching either the reactor cooling system or the containment building. This is particularly true in the case of the less volatile fission products.

From a qualitative point of view, the main physical parameters affecting the release of fission products are as follows:

- The temperature is the main parameter, at least in terms of determining the loss of core geometry.
- The oxidising or reducing conditions experienced by the fuel also play a major role. In particular, the kinetics of the release of volatile fission products are accelerated under oxidising conditions. The total release of some fission products is also very sensitive to the oxygen potential. For example, the release of Mo is greater in the presence of steam, that of Ru may be very high in the presence of air and, conversely, the release of Ba, Ce and Eu is greater under reducing conditions.

- Interactions with the cladding and/or structural components may also play a significant role. For example, the presence of tin in the cladding slows the emission of the volatile elements tellurium and antimony. Barium, which makes a major contribution to the residual power via its descendant ^{140}La, is also partially trapped in the cladding (probably due to the formation of zirconates) and in the internal core structure (iron).
- The burnup increases the release, partly through the kinetics of the volatile fission products, and partly through the extent of the release of low volatile elements such as Nb, Ru, Ce and Np.
- The type of fuel also appears to play a significant role. The releases from MOX tend to be higher than those from UO_2. This phenomenon is probably associated with the heterogeneous microstructure, with the presence of clusters rich in plutonium where the local burnup is very high.
- Finally, the morphology of the fuel during its degradation in-vessel also plays an important role. The change from a ‘degraded fuel rod’ to a ‘debris bed’ geometry is accompanied by an increase in the release of fission products due to an increase in the surface area/volume ratio, while the opposite occurs during the transition from a ‘debris bed’ to a ‘molten pool’ as a solid crust forms on the surface of the pool.

7.1.2.3 Degrees of volatility

To conclude, the current knowledge resulting mainly from the VERCORS analytical experiments and the PHEBUS integral tests (see Section 7.1.3) has enabled the fission products and fission gasses to be classified in four classes of decreasing volatilities with the following characteristics (Figure 7.1-1):

- Volatile fission gas and fission products (Kr, Xe, I, Cs, Br and Rb, together with Te, Sb and probably Ag): Almost the total quantity of these products present is released even before the formation of the molten pool. The release kinetics of all these elements is accelerated under oxidising conditions, and slightly slowed in the case of Te and Sb by interaction with the tin in the cladding, although this is not really significant.
- Semi-volatile fission products (Mo, Ba, Y and Rh): These products are characterised by a level of release that can be very high, in some cases equivalent to the volatile fission products. The release can be almost total, but these products are very sensitive to the oxidising or reducing conditions and they tend to be retained to a large degree by the structures in the upper head.
- Low volatile fission products (Sr, Nb, Ru, La, Ce, Eu and Np): These fission products are characterised by a low but significant level of release, up to 10% during the fuel rod degradation phase prior to the loss of core geometry. However, this release may be considerably higher in the case of high burnup fuels (releases of between 15 and 30% have been measured for Nb, Ru and Ce from a UO_2 fuel at GWd/t) or under specific conditions (e.g. Ru in air). It is expected, however, that the retention in the structures of the upper head will be high.
- Non-volatile fission products (Zr and Nd): No significant release of these products has been measured experimentally at the present time. These are the two most refractory fission products.

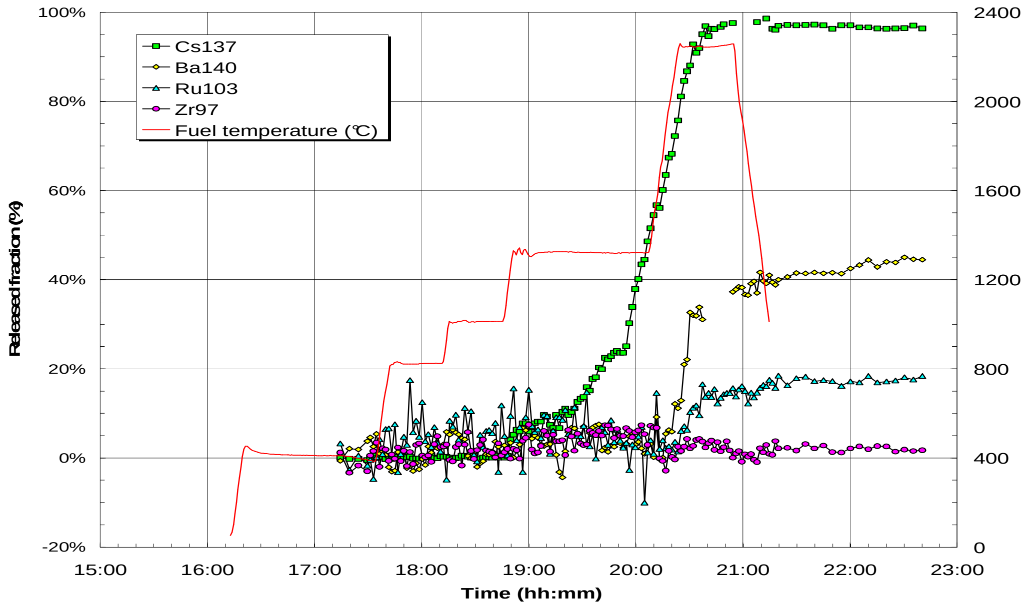


Figure 7.1-1: Illustration of the four fission product volatility classes following a VERCORS test

7.1.3 EXPERIMENTAL PROGRAMES, MODELLING AND COMPUTER CODES

7.1.3.1 Main experimental programmes

The experimental programmes devoted to the release of fission products have mainly been out of pile analytical experiments carried out on sections of irradiated fuel. Additional integral tests would be useful, especially in relation to the link between the degradation of the core and the release of fission products.

7.1.3.1.1 Analytical experiments

Five large-scale analytical programmes have been carried out since the end of the 1970s in Germany (SASCHA [7.1_4]), the USA (HI/VI [7.1_5]), Canada (CRL [7.1_6]), Japan (VEGA [7.1_7]), and France (HEVA/VERCORS [7.1_8]). The HEVA/VERCORS programme is described in more detail as the quality and accessibility of the results have made it one of the most useful.

- The SASCHA programme was the first of this type. It was carried out using non-irradiated UO_2 fuel under a range of atmospheres including argon, air and steam. The UO_2 pellets were manufactured specially with additives simulating the fission products. Although these conditions were not really representative of the actual implantation of fission products by irradiation, this programme did contribute the first estimates of the release of iodine and caesium at temperatures of up to 2000 °C.

- The HI/VI programme was carried out by ORNL between 1981 and 1993 (six HI tests and seven VI tests). The experimental configuration, at least in the case of the VI tests, was close to that of the VERCORS programme with a loop fitted with sequential thermal gradient tubes (TGT), filters, a condenser and traps for the collection of active gas. The fuel sample was a section of irradiated UO_2 approximately 15 cm long and closed at the ends. A hole was drilled in the cladding half way along. This programme provided important and representative results on the release of fission products, but only in relation to long half-life fission products (mainly ^{85}Kr, ^{106}Ru, ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce and ^{154}Eu) as the samples were not re-irradiated prior to the tests.

- The CRL programme is a highly analytical programme that is still in progress at the present time. It consists of a number of tests on fragments of irradiated fuel (100 mg to 1 g) or on short fuel sections (including the cladding). The use of a furnace heated by electrical elements limits the test temperatures to a maximum of 2000 °C. Some samples were re-irradiated immediately prior to the tests in order to measure the short half-life fission products. One of the most important results from this programme was the first quantification of the very high release of ruthenium in air.
- The VEGA programme is very similar to the VERCORS programme, especially the VERCORS HT series. This programme is now complete. Ten tests were carried out, eight on UO_2 fuel and two using MOX fuel. Some of the tests were carried out in a steam atmosphere up to the temperature at which the fuel melted. Some of the samples were re-irradiated prior to the tests. However, this was under less than optimal conditions (irradiation time too short and decay time too long). A unique feature of these tests was the inclusion of tests at a pressure of 1 MPa. These tests showed a reduction in the release of caesium.
- The HEVA/VERCORS programme, financed jointly by the IRSN and EDF, aimed to quantify the release of fission products and heavy nuclei (kinetics and total release) from irradiated nuclear fuel under conditions representative of a severe accident. These tests, carried out in a high activity cell, were performed on samples of different types of fuel irradiated in a PWR (around 20 g of fuel) under a range of experimental conditions. The majority of the samples were re-irradiated over a few days at low power in an experimental reactor in order to build up an inventory of the short half-life fission products producing the worst radiological effects. These samples were then heated in an induction furnace under an atmosphere consisting of a mixture of steam and hydrogen the conditions simulating a severe accident (Figure 7.1-2). The release of fission products was measured on line by means of gamma spectrometry during the accident sequence, with a direct view of the fuel. Twenty-five tests were carried out between 1983 and 2002 in three phases: Eight HEVA tests (release of volatile and some semi-volatile fission products at temperatures of up to 2100 °C), six VERCORS tests (volatile, semi-volatile and some low volatile fission products at temperatures of up to 2300 °C, the limit for collapse of the fuel), and eleven HT/RT tests involving all types of fission products up to the point at which the fuel melted. The results of these tests were used to establish one of the largest databases relating to the release of fission products. The parameters investigated during these tests included temperature reached (above or below the melting point of the fuel), the oxidising or reducing conditions of the atmosphere, the burnup, the type of fuel (usually UO_2 with two tests using MOX) and the initial fuel geometry (intact or fuel debris to simulate the final phase of a severe accident).

At the present time, there are still some remaining uncertainties relating to the release of fission products. This is particularly the case for high burnup UO_2 fuels (70 GWd/t and above), MOX fuels, and in certain scenarios including the entry of air or reflooding. The future VERDON programme (CEA-EDF-IRSN), which is part of the new international Source Term programme [7.1_9], should address these topics with the exception of reflooding. The Russian ISTC QUENCH programme should also be noted as this is the only analytical experiment addressing the reflooding issue on irradiated fuel. This programme will consider the reflooding case by heating a fuel section of irradiated UO_2 (60 GWd/t) up to 1600 °C under steam, followed by reflooding with cold water (see Section 3.1).

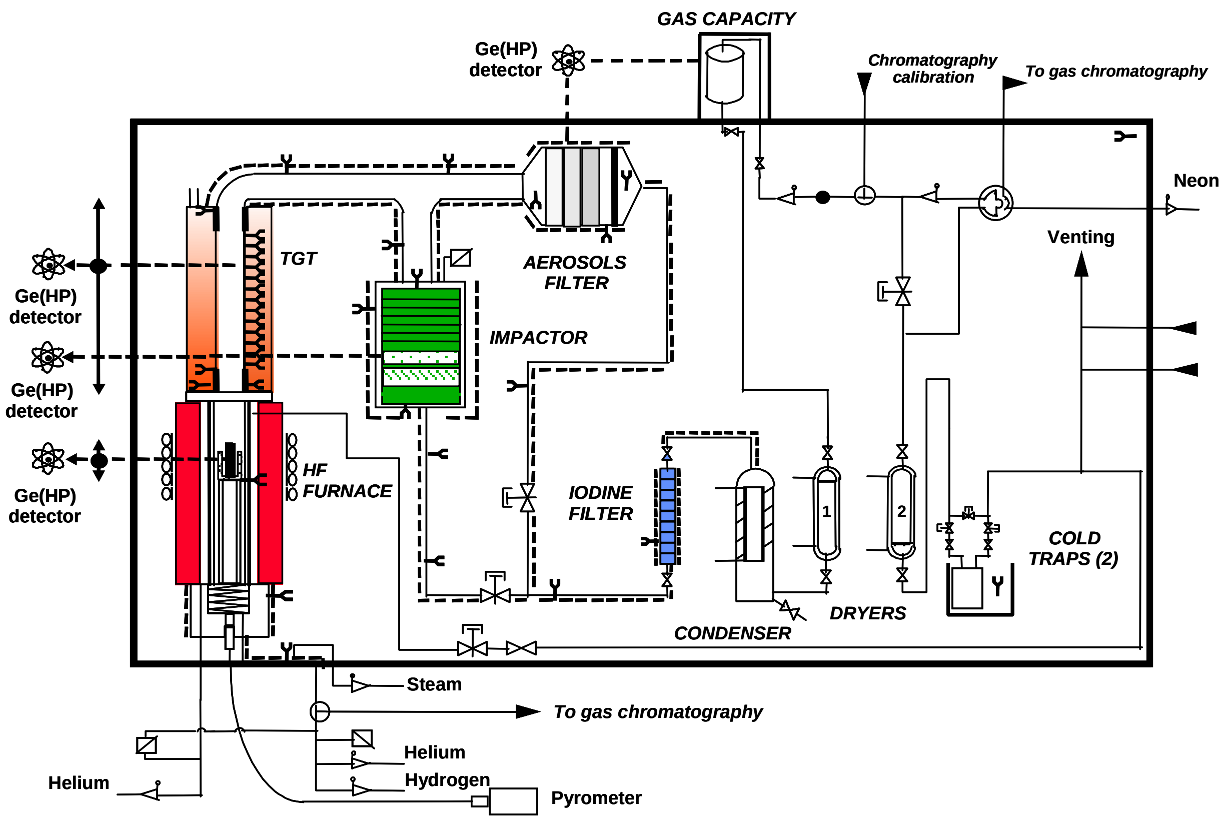


Figure 7.1-2: The VERCORS HT loop

7.1.3.1.2 Integral tests

The PHEBUS-FP programme is the most recent and has provided the most representative data relating to the overall situation during a severe accident [7.1_9]. The tests covered all phases of an accident, from the start of the degradation of the core through to the formation of a molten pool within the reactor vessel. The stacked section consisted of a bundle of 20 irradiated fuel rods surrounding a central rod simulating a control rod (AIC or B_4C), except for test FPT0 which used 20 un-irradiated fuel rods and test FPT4 which used a debris bed configuration. The released fission products were carried through temperature controlled lines in order to simulate the conditions in the reactor cooling system. This system fed into a 10 m^3 tank simulating the containment building of a PWR. The tank had a sump in the base containing water at a controlled pH.

Five tests were carried out between 1993 and 1999:

- Test FPT0 was carried out using fresh fuel with an AIC control rod under steam-rich conditions with an acid sump.

- Test FPT1 was identical to FPT0, but with the fuel irradiated at 24 GWd/t.
- Test FPT2 used the same fuel and control rod as in test FPT1, but in a less oxidising environment with an alkaline sump that was evaporating by the end of the test.
- Test FPT3 was carried out under the same conditions as FPT2, but with a B_4C control rod and an evaporating acid sump.
- Test FPT4 used an initial debris bed configuration to simulate the final phase of a severe accident.

The main advantage of the PHEBUS-FP programme over the analytical tests in terms of the in-vessel release of fission products was the identification of a link between the degradation and the release of barium, and the low level of release from the molten pool. The release of barium was much less than was the case in the analytical tests, the difference in behaviour being attributed to interactions between the Ba and the cladding, and possibly iron, that reduced its volatility. The results for the release of other fission products were consistent with those from the analytical tests.

7.1.3.2 Modelling and computer codes

Two approaches have been followed to model the release of fission products and to incorporate these models in simulation code. One is an empirical approach which is easy to integrate within scenario codes, the other is a mechanistic approach which better describes the totality of the physical phenomena. Examples of two IRSN software tools are given in order to illustrate these two approaches. These are: ELSA, the release model incorporated in the ASTEC code, and MFPR, a mechanistic model developed at the IBRAE in Russia.

7.1.3.2.1 The empirical approach used in the ELSA code

The ELSA code [7.1_10] models three categories for the release of fission products, based on the simple principle of the 'limiting phenomenon'.

- The release of <u>volatile</u> fission products (Xe, Kr, I, Br, Cs, Rb, Sb and Te) is governed by the mechanism of <u>diffusion within the grain of the fuel</u> using an improved Booth model. The diffusion coefficient is a function of the temperature and also the stoechiometry of the fuel. This coefficient is identical for all fission products except for Sb and Te. In the case of these two products, the release is delayed to take account of their retention within the grain until it has become fully oxidised.
- The release of <u>semi-volatile</u> fission products is governed by the <u>mass transfer</u> resulting from vaporisation at the grain boundaries. The tables of vapour pressures are generated from thermodynamic correlations performed using the GEMINI 2 (Sr, Ru, Ba and La) or FACT (Mo, Ce and Eu) solvers. It should be noted that the same mass transfer mechanism is then applied to all the fission products in order to model their release from the molten pool.
- The release of the remaining <u>non-volatile</u> fission products is governed by the <u>vaporisation of the over-stoechiometric UO_2</u> as it oxidises, eventually forming UO_3. This category also includes the actinides U, Np, Pu, Am and Cm.

7.1.3.2.2 *The mechanistic approach used in the MFPR code*

MFPR [7.1_11] is a 0-D mechanistic code used to simulate the releases from solid UO_2 fuel. The fission products, assumed to be stable, are incorporated in the fuel matrix in either atomic form or as oxides. Two types of model have been developed, one relating to fission gas and the other to fission products.

The fission gas model includes all the physical phenomena described in Section 7.1.2.1, including the intragranular diffusion of atoms and bubbles towards the grain boundaries, the bubble formation mechanisms (nucleation and growth) and bubble destruction mechanisms (return into solution). The gas are released from the grain boundaries following the coalescence and interconnection of the gas bubbles.

The current fission product model covers thirteen elements; Cs, I, Te, Mo, Ru, Sb, Ba, Sr, Zr, La, Ce, Nd and Eu. These are assumed to diffuse towards the surface of the grain, with some being oxidised on the way. They therefore form three distinct phases, a metal phase, a ternary phase known as the grey phase consisting of fission product oxides, and a separate phase describing the form of CsI. The release is subsequently governed by these three phases coming to thermodynamic equilibrium with the gas at the phase boundaries.

The validation process for these codes is well advanced. The release of volatile fission products gives a good reproduction of experimental observations despite the fact that the ELSA approach tends to under-estimate slightly the release at intermediate temperatures (between 1000 and 1500 °C). This is due to its failure to take account of the intergranular inventories. The highest uncertainties relate to the simulations of semi-volatile and low volatile fission products, mainly due to the difficulty of modelling accurately the chemical mechanisms involved and their potential interactions with structural components (control rods, internal core structures, etc.).

7.1.4 SUMMARY AND OUTLOOK

The database of experimental results obtained from the analytical tests carried out on sections of irradiated fuel is fairly extensive in the case of medium burnup UO_2 fuels. Additional data is available from the integral tests such as PHEBUS-FP. These experiments have refined the present knowledge base in relation to the parameters influencing release into the environment, including the temperature, the oxidising or reducing conditions, the interactions with structural materials especially the fuel rod claddings, the burnup, the type of fuel (UO_2 or MOX) and the state of the fuel (solid or molten fuel).

These results have been used to develop and validate two types of model. Mechanistic models are capable of describing the majority of the interactions taking place within the fuel and are often used in the interpretation of test results. Simplified models may be derived from them in order to describe the dominant phenomena. These simplified models are then used in the fission product release modules of integral codes.

Some of the hypotheses put forward to interpret the test results successfully reproduce the dependence of the release on the parameters of temperature, burnup, and the composition of the atmosphere surrounding the fuel. They are mainly based on the physical and chemical transformations taking place within the fuel. The MFPR code used in the interpretation of the test results is capable of describing the behaviour of the fission gas, the variations in the composition of the various phases containing the fission products within the fuel, and the chemical speciation of the elements concerned. However, these hypotheses suffer from a lack of validation at the present time. This is expected to be addressed by a programme of microanalyse performed on fuel samples, which where previously used in release tests as part of the international Source Term programme. The knowledge acquired in this way will enable the results to be extrapolated to new fuels, reducing the need for future release tests.

The database of experimental results available at the present time is, however, considered to be insufficient. It is therefore proposed to extend the experimental programme to include MOX fuel (2 tests) and high burnup UO_2 fuel (1 test) as part of the VERDON programme included within the international Source Term programme. The results of these tests will be used to validate, and even to improve, the models developed earlier.

The release of fission products during the reflooding of high burnup fuel has been studied experimentally as part of the ISTC-QUENCH programme. The case for additional experimental work will be examined in the light of the results of these tests.

Accidents involving the entry of air, for example following a breach in the reactor vessel or with the reactor shut down, will also be examined. Available data, mainly originating in Canada, shows that ruthenium behaves like a volatile fission product under these conditions and may be almost completely released [7.1_12]. Models are currently in development and should be available by the end of 2006. They will be validated by a specific VERDON test to investigate the release in air.

7.2 TRANSPORT OF FISSION PRODUCTS IN THE PRIMARY AND SECONDARY CIRCUITS

7.2.1 DEFINITION AND OVERALL PHENOMENOLOGY

The fission products and structural materials are released from the core mainly in the form of vapours. These vapours cool down in the upper part of the reactor pressure vessel and then in the reactor cooling system. A number of phenomena occur during this cooling. These include:

- The condensation of vapours on nucleation kernels to form fine particles. This process is commonly referred to as homogeneous nucleation.
- The condensation of the vapours on existing particles. This process is referred to as heterogeneous nucleation.
- The condensation of the vapours on the walls forming deposits.

The temperatures at which these phenomena occur depend on the chemical form of the fission products and structural materials, a point which will be discussed in Section 7.5. Following condensation of the vapours, the transport mechanisms for the fission products and structural materials are, with the notable exception of iodine and ruthenium, governed mainly by aerosol physics. The main phenomena are shown in Figure 7.2-1.

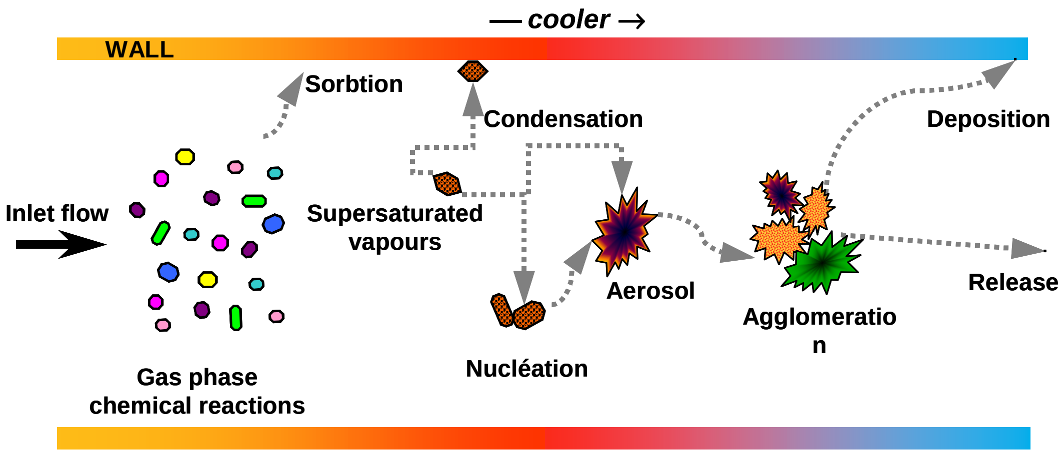


Figure 7.2-1: Aerosol transport phenomena

7.2.2 PHYSICAL PHENOMENA INVOLVED

7.2.2.1 Nucleation phenomena

The phenomenon of particle formation by homogeneous nucleation may be described using either a microscopic or a simplified approach. Simulations using a simplified approach provide results that are very close to those from a microscopic approach, and they are therefore generally chosen for use in calculation codes.

The phenomenon of heterogeneous nucleation involves the condensation of a vapour on to existing aerosol particles consisting of less volatile elements that have condensed at a higher temperature. The existing models, based on the diffusion of molecules in a carrier gas, are considered to be satisfactory.

7.2.2.2 Coagulation or agglomeration phenomena

The particles (aerosols) formed by nucleation are caused to move relative to the carrier gas by Brownian diffusion, turbulence and gravitational settling. This movement results in collisions between particles causing them to agglomerate. The aerosol particles therefore increase in size as they are transported. The typical size of an aerosol particle in the reactor cooling system is a few micrometres.

These phenomena are well understood and the available models give results that compare satisfactorily with experimental data. It should however be noted that some of the parameters of the models, such as the shape factor of the particles i.e. their deviation from perfect spheres, are less well understood. This lack of understanding does not, however, appear to lead to any major uncertainties.

7.2.2.3 Deposition by vapour condensation

The fission products and structural materials released from the core in vapour form are partially deposited on the cooler walls by condensation. This deposition phenomenon is usually described by means of an analogy between heat and mass transfer. This assumes that the properties of the molecules involved are known, especially their saturation pressure. The associated chemical aspects are discussed in Section 7.5.

Prior to the PHEBUS-FP integral tests, there was a high degree of confidence in the models of deposition by condensation. These tests have since shown that the condensation deposits were underestimated.

The underestimation could be by a factor of up to two. A number of hypotheses have been proposed to explain this difference. It has been concluded that it is due to effects associated with the fact that the the flow is neither thermally nor hydraulically fully developed in Phebus experiments [7.2_1]. In such situations, the heat exchange and mass transfer coefficients are increased, leading to a corresponding increase in the deposits. A CFD calculations applied to PHEBUS have validated this explanation. The improved models have yet to be incorporated in the simulation codes.

7.2.2.4 Deposition by gravitational settling

The aerosol particles are affected by gravity which imposes a downwards component on their trajectories. This effect results in particles being deposited on horizontal surfaces. This deposition phenomenon is only significant at low fluid velocities corresponding to long transit times.

This phenomenon is well understood and the available models give results that compare satisfactorily with experimental data.

7.2.2.5 Deposition by Brownian or turbulent diffusion

The Brownian motion of the particles may bring them into contact with the walls where they are then deposited. This phenomenon is only significant under laminar flow conditions, as turbulent diffusion becomes dominant at high speeds. These phenomena are well understood and the available models again give results that compare satisfactorily with experimental data.

7.2.2.6 Deposition by impaction

This phenomenon occurs particularly at points where the geometry changes, such as bends in the pipework, changes in cross-sectional area, obstacles, etc. The inertia of the particles causes them to deviate from the flow stream lines and they collide with the walls. One important parameter is the particle size. Deposition by impaction occurs mainly with larger particles.

In the case of the reactor cooling system, the most significant deposits occur at bends and changes in cross-sectional area (e.g. on entry to the steam generator tubes). These phenomena are modelled on the basis of correlations taken from the literature and the models are generally considered to give satisfactory results. Deposits in the secondary circuit in the event of a STGR type sequence are discussed in another section.

7.2.2.7 Deposition by thermophoresis

When particles are transported through a temperature field, the number of collisions between molecules of the carrier gas and the particles is greater in the hotter regions than in the cooler regions. This results in a movement towards the walls which are cooler than the gas. A large number of theoretical and experimental studies have been carried out in the past, and these have resulted in a generally accepted set of expressions for the deposition rate by thermophoresis [7.2_2]. Examples of past experimental studies include the TUBA thermophoresis tests carried out by the IPSN [7.2_3].

The validation of the models against this type of test is satisfactory. However, the PHEBUS-FP integral tests have revealed significant discrepancies between the model and the experimental results for deposition by thermophoresis in a simulated steam generator. The models overestimate the deposition by a factor of two. This is the case for all severe accident simulation codes. A number of ideas have been investigated in an attempt to explain this difference, in particular the influence of the differences between the PHEBUS tests and those used to validate the models such as the fluid-wall temperature differences and the aerosols concentration. CFD calculations with particles tracking have also been carried out [7.2_5], but there remains at the present time no convincing explanation of these differences. Considering the large break sequences simulated in PHEBUS, these differences between the calculations and experimental data have little impact on the source term as the retention in the reactor cooling system is low. However, in the case of containment bypass sequences such as the V_LOCA sequence, retention in the pipework has a significant impact on the potential releases. Work is therefore continuing with the aim of understanding the phenomena observed in PHEBUS.

7.2.2.8 Deposition by diffusiophoresis

Steam condenses on the walls when they are cooler than the saturation temperature of the steam. This condensation cause a flow of gas (at the Stefan velocity) which draws particles towards the wall [7.2_4]. A description of this deposition phenomenon requires an accurate calculation of the steam condensation which is generally considered to be low in the reactor cooling system. The validation of the models was mainly based on the results of the TUBA diffusiophoresis tests carried out by the IPSN [7.2_6].

7.2.2.9 Mechanical resuspension

A number of phenomena, including the production of steam following reflooding of the core, may result in significant deposits in the pipework. The deposited particles may then be flushed along mechanically. This phenomenon may be significant in the case of highly turbulent flows or dry deposits. The associated physics is relatively complex. It may be summarised by the statement that resuspension occurs when the aerodynamic forces exerted on the particles are greater than the adhesive forces holding them to the walls. Several models have been developed with a number of different assumptions such as whether the deposited particles are in a single layer or in multiple layers. The validation of these models is mainly based on the results of the STORM tests [7.2_7] carried out at the ISPRA Joint Research Centre in Italy. This validation is not considered to be adequate. It should be noted that this point may have a safety impact in containment bypass sequences such as V_LOCA and STGR, in which the retention in the pipework may be significant.

7.2.2.10 Revolatilisation

This phenomenon is the opposite of deposition by the condensation of vapour. When the conditions change (temperature, oxygen potential of the fluid, or vapour concentration), the deposited vapours may revolatilise. In view of the importance of the associated chemical aspects, this point will be discussed further in Section 7.5.

It should be noted that this phenomenon was clearly demonstrated during the PHEBUS integral tests and the VERCORS release tests.

7.2.2.11 Deposits in the secondary system (steam generator)

These occur during STGR sequences, with either an initial breach or an induced breach. Depending on the circumstances, the steam generator secondary system will be either dry or flooded. At the present time, conservative assumptions are used in the evaluation of retained aerosols on the secondary side of the steam generator as there have been very few experiments carried out under representative conditions and the existing simulation codes (SPARC, BUSCA and SUPRA) have not been fully validated. A more realistic approach to retention in the secondary side of a steam generator is therefore required. It will use the results of the current ARTIST experimental programme [7.2_8].

7.2.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

The phenomena associated with aerosol physics have been the subject of a large number of theoretical and experimental studies in the past, many outside the nuclear field, and the basic relationships and models are well established. Those models for which a general consensus exists have been incorporated in the codes simulating severe accidents, including the SOPHAEROS module in the IRSN ASTEC integral code.

The requirements for additional R&D programmes have been identified as part of the SARNET programme. The following two main directions have been identified for studying the transport or aerosols in the primary and secondary cooling systems:

- Mechanical resuspension:

  A number of experimental programmes are currently in progress (PECA, ARTIST and VTT-DEPOS) which should result in the development of improved models.

- Deposits in the secondary side of the steam generator:

  This is currently being studied as part of the ARTIST programme, carried out by PSI in Switzerland in partnership with the IRSN. The experimental steam generator is based on the FRAMATOME design, with a reduced height of 3.8 metres. The internal structures are all present and the upper structures (separators and dryers) are represented at the same scale as the actual steam generator in the Beznau nuclear power plant. The experiments will study the retention of aerosols under 'dry' conditions in a number of regions, including the broken tubes, close to the break, beyond the breaK, and in the separators and dryers. Retention in a flooded steam generator will also be investigated. The model derived from these experiments will be incorporated in the SOPHAEROS module of the ASTEC integral code.

- The retention of aerosols in cracks in the containment building:

  This topic will be studied as part of the SARNET network of excellence. Models for the retention of aerosols are currently under development or are being validated against the results of previous tests. One important point that needs to be more investigated is the increase in retention due to the condensation of steam. Suitable installations for carrying out additional tests have been identified (COLIMA at CEA Cadarache, and installations at the IRSN Saclay site). Proposals for additional tests are currently under discussion within SARNET.

7.2.4 SUMMARY AND OUTLOOK

The physical phenomena associated with the transport of aerosols in the reactor cooling system are generally well understood and models have been developed to describe them, often based on data obtained in fields other than nuclear engineering. The main deposition phenomena, such as thermophoresis and the diffusiophoresis, have been the subject of specific experimental programmes in the past with the aim of validating models.

The models describing the phenomena of the mechanical resuspension of deposits as a result of high flow rate are less well validated. Work is currently in progress as part of the European SARNET network of excellence with the aim of improving the mechanical resuspension models by the use of existing data or data obtained during current programmes on the behaviour of aerosols in the secondary side of steam generators.

The retention of aerosols in the secondary side of steam generators is poorly quantified, leading to the use of retention coefficients that may be too conservative in safety studies. This problem is currently being studied as part of the international ARTIST programme being carried out by PSI in Switzerland.

At the present time, the safety studies assume that the emission of aerosols through cracks in the containment building is proportional to the leakage rate without taking any account of retention. Current studies on this topic should eventually make it possible to use less penalising assumptions.

7.3 EX-VESSEL RELEASE OF FISSION PRODUCTS

7.3.1 DEFINITION AND OVERALL PHENOMENOLOGY

The release of fission products and aerosols outside the reactor vessel may be associated with a number of phenomena, including:

- The release of aerosols from a boiling sump.
- Releases during molten corium-concrete interaction.
- The resuspension of aerosols deposited on the walls.

Releases of iodine and ruthenium both involve complex chemical processes and a specific discussion of these elements is given in Section 7.5.

7.3.2 PHYSICAL PHENOMENA INVOLVED

7.3.2.1 Release of aerosols from a boiling sump

Most of the aerosols released within the containment eventually reach the water in the sump as a result of gravitational settling. If the water in the sump begins to boil, the trapped aerosols may re-enter suspension. The results of the REST tests [7.3_1] carried out by KFK in the past have been used to develop semi-empirical models for both soluble and insoluble aerosols. These models have been used in source term re-evaluation studies carried out by the IRSN.

7.3.2.2 Releases during molten corium-concrete interaction

This release phase involves mainly semi and low volatile fission products as the volatile fission products are for the most part released within the vessel. The rates of release depend on the composition of the corium, especially its metallic zirconium content, and on the composition of the concrete.

The releases may be estimated from the vapour pressures of the fission products calculated using thermodynamic codes such as GEMINI, with the variable parameters being the composition of the corium and the type of concrete. The highest values are obtained when the corium is rich in zirconium and the concrete is a silica type. The only elements with a significant release (greater than 1%) are barium and strontium.

As the rates of release are low, it has not been considered necessary to launch R&D programmes to refine the results.

7.3.2.3 Resuspension of aerosols deposited on the walls

Certain events, such as the combustion of hydrogen and the interaction between corium and water, may result in high flow rates close to the walls of the containment building. It is therefore likely that any aerosols previously deposited on the walls would be resuspended (see Section 7.2). These phenomena are not taken into account at the present time. The use of improved mechanical resuspension models should eventually make it possible to evaluate their effects. However, it is not expected that their influence will be significant in terms of radiological consequences for the environment.

7.3.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

The impact on the source term of releases from a boiling sump and during a molten corium-concrete interaction is relatively low. Specific models are available to the IRSN in order to evaluate these releases, but these are not incorporated in the ASTEC integral computer code. The work to incorporate these models has not been considered to take priority over other development requirements.

7.3.4 SUMMARY AND OUTLOOK

The ex-vessel release of fission products may occur if the sump begins to boil (release of aerosols) or they may be released from the corium during its interaction with concrete. The studies carried out up to now have indicated release rates that are low compared with in-vessel releases. Additional R&D programmes to address these two questions have not been considered necessary.

The possible resuspension of aerosols deposited on the walls in response to ‘violent’ events such as the combustion of hydrogen has not been investigated in detail at the present time. An evaluation using improved resuspension models might eventually be possible.

7.4 BEHAVIOUR OF THE AEROSOLS WITHIN THE CONTAINMENT BUILDING

7.4.1 DEFINITION AND OVERALL PHENOMENOLOGY

The aerosols emitted within the containment building are subjected to the phenomena of agglomeration and deposition. The aerosols may also be re-suspended from the deposits. The basic physical phenomena are the same as those governing transport in the reactor cooling system. They depend on the thermal hydraulic conditions (the relative humidity and whether or not the steam condenses). The main deposition phenomena are gravitational settling and diffusiophoresis. Some safety systems, such as spray, also have a major influence on aerosol concentrations.

7.4.2 PHYSICAL PHENOMENA INVOLVED

7.4.2.1 Agglomeration and coagulation

The agglomeration phenomena acting on the fission products are the same as those in the reactor cooling system. They result in an increase of the particle sizes, accelerating deposition by gravitational settling.

The phenomena of hygroscopicity may also play an important role. Some compounds, such as caesium hydroxide, have the ability to absorb water molecules up to the point at which droplets form and the solid material is dissolved. This results in an increase of the aerosol particle, accelerating gravitational settling.

Models are available to calculate the size of the droplets formed at equilibrium as a function of the temperature and relative humidity. These models use the Mason equation, which takes account of the Van't Hoff factor describing the number of ions in which a molecule dissociates to form an ideal solution. A correction is applied to take account of the surface tension of the liquid (the Kelvin effect).

One of the difficulties in using this type of model is the need to know the chemical species formed by the fission products. Until recently, it was generally accepted that caesium would be injected into the containment building mainly in the form of the hydroxide, which is strongly hygroscopic. The first two PHEBUS-FP tests showed that this was not the case. Thermodynamic calculations taking account of the actual release of molybdenum have shown that the most likely chemical form is caesium molybdate. This is consistent with the volatility of the caesium as measured during the tests. However, this result cannot be directly extrapolated to all situations and all sequences. The same tests also showed that the aerosol particles were formed from agglomerates containing all the emitted fission products and structural materials, most of which were insoluble.

Care is therefore needed when considering the hygroscopicity of the aerosols and sensitivity calculations may be needed in order to assess the relative weightings of the various parameters.

7.4.2.2 Washout of deposits by the condensates

The aerosols deposited on the walls may be washed out into the sumps by the condensed water. A simplistic solution, which is however justified by the physics, is to assume that insoluble aerosols are not washed out, while soluble ones are. This raises the same difficulties as those described in the previous Section. There is little risk of error, however, in assuming that the barium and caesium are always soluble.

7.4.2.3 Aerosol depletion by spray

The main aim of spray is to avoid an excessive rise in pressure within the containment building. This system is also capable of causing a rapid decrease in the concentration of aerosols in suspension. In the past, simple models based on a time constant have been used. More recent studies have enabled a better physical description of the phenomena involved to be developed.

The depletion of aerosols by spray depends on the characteristics of the droplets, especially their mass, velocity and temperature as they fall. The changes in these characteristics depend on evaporation and condensation, and on the coalescence of the droplets.

The aerosols are trapped by the droplets by means of the following mechanisms:

- Inertial capture and interception: Larger particles may not be pushed sufficiently far from the flow stream lines to avoid collision with the droplets.
- Brownian diffusion. This mechanism is particularly effective in the case of small particles close to the droplets.
- Phoretic capture, associated with the movement of the particles in a temperature field. This mechanism is particularly effective in the upper part of the containment building before the droplets reach thermal equilibrium with the atmosphere.

7.4.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

The basic mechanisms associated with aerosol physics are described by proven models, often based on data acquired outside the nuclear field. These models have been validated, where necessary, by means of specific experimental programmes such as PITEAS at the IPSN, Cadarache [7.4_1].

The most recent R&D work has been focused on the depletion of aerosols by spray, with the aim of quantifying the kinetics and the limiting efficiency of spray in the case of aerosols and gaseous iodine in current reactors, and of optimising the spray aerosol depletion function in future reactors.

- From the experimental point of view, the CARAIDAS tests [7.4_2] carried out at IRSN Saclay have measured the basic efficiencies of the various mechanisms involved in the collection of aerosols and gaseous iodine species by a droplet in steady-state conditions, over a range of conditions representative of a severe accident (temperature, pressure, humidity, droplet pH, and iodine concentration). The experiments were performed in a cylindrical tank, 5 metres high by 0.6 metres in diameter.

  A new programme, TOSQAN AEROSOLS, is about to start work on this topic. The aim of these tests, to be carried out using the TOSQAN facility (see Section 4.2.3.1.3), is to improve understanding of the collection of aerosols by a spray of water droplets in a situation representative of a severe accident in a PWR. The aim is therefore to extrapolate the models developed during the CARAIDAS programme to include operational conditions closer to those during actual spray operation.

- From a modelling point of view, a detailed physical description of the changes taking place in a water droplet as it falls, together with the various aerosol capture mechanisms, has been developed. The models are incorporated in the CPA module of the ASTEC integral code. The thermal hydraulic aspects have been validated against the CSE tests [7.4_3] carried out in the USA, and the aerosol aspects against the CARAIDAS tests [7.4_2].

7.4.4 SUMMARY AND OUTLOOK

The phenomena governing the behaviour of aerosols in the containment building are generally well understood and physical models have been developed to describe them, often based on data obtained in fields other than nuclear engineering. The main deposition phenomena, such as diffusiophoresis and gravitational settling, have been the subject of specific experimental programmes in the past with the aim of validating models.

7.5 THE CHEMISTRY OF FISSION PRODUCTS

7.5.1 DEFINITION AND OVERALL PHENOMENOLOGY

Fission products are emitted from the fuel in the form of vapours. They are emitted in a chemical form that depends on an equilibrium with the condensed phase in which they are present in the fuel. This equilibrium varies during the course of an accident, mainly due to variations in temperature and oxygen potential. The emitted fission products encounter a variety of environments during their transport, with variations in the temperature and composition of the carrier gas, and their chemical speciation in the reactor cooling system may thus change. Most of the chemical reactions take place in the gas phase, but the vapours also interact with the walls of the pipework. The structural materials released from the core also have play a role. These include silver-indium-cadmium or boron carbide materials from the control rods, and tin from the zircaloy cladding.

The majority of the fission products are released into the containment building in a condensed form (aerosols) or rapidly condense shortly after release (e.g. in the case of a hot leg break). It is generally accepted that the potential chemical reactions involving aerosols in suspension in the atmosphere of the containment building do not have a significant effect. A large fraction of the aerosols is washed down into the sump water where they may or may not dissolve depending on their chemical speciation.

The two specific cases of iodine and ruthenium merit consideration as they may be present in gaseous form within the containment building in quantities that are small but significant from a safety point of view. The chemistry of these two highly radiotoxic elements [7.5_1] is complex in both the gas and liquid phase, and they interact with both metal and painted surfaces. Their interaction with radiolysis products from water and air also needs to be taken into account.

7.5.2 PHYSICAL PHENOMENA INVOLVED

7.5.2.1 Chemistry in the gas phase in the reactor cooling system

It is possible in principle to calculate the chemical speciation at equilibrium if the concentrations of the various elements and the thermodynamic properties of the possible species are known. The second of these poses a number of difficulties as the available databases are limited in extent and uncertainties remain in the case of some of the data.

Thermodynamic equilibrium is also not always reached. While this is very likely at the high temperatures prevailing close to the core, it is certainly not the case in the cold leg where the temperatures are too low. This implies that the chemical kinetics in the temperature transition zones will affect the species formed.

The studies carried out in the past with the aim of determining the species formed have concentrated on volatile elements and a simple Cs-I-O-H system [7.5_2], taking account of the effects of boric acid. In the absence of boron, these studies conclude that iodine is transported in the form of caesium iodide, CsI, with the remaining caesium in the form of the hydroxide, CsOH. The presence of boron may result in the formation of caesium borate which is less volatile than the hydroxide. This would free some of the iodine to form hydroiodic acid, HI, which is more volatile than caesium iodide [7.5_4]. These studies are based on experiments with simulated fission products similar to those in the Winfrith FALCON programme [7.5_3].

The results of these studies led the developers of early developed codes such as MELCOR to fix the chemical form of the elements being transported without calculating the chemistry. In more recent codes, such as SOPHAEROS, the chemical speciation is calculated. Simplified calculations based on an extended database are used rather than full thermodynamic calculations.

The chemical forms assumed in the past have been brought back into question following the results of the PHEBUS-FP programme [7.5_7], [7.1_9]. The advantage of this programme is that a bundle of irradiated degrading fuel rods is used as the source of the fission products and structural materials. It is probably the most realistic source that can be possibly achieved, both in terms of the composition and the release kinetics.

In the PHEBUS-FP tests, the chemical forms are not measured directly, given the detection limits of the various techniques available for use on small molar quantities of radioactive materials. However, an indirect indication is provided by the volatility of the elements (condensation temperature) and their solubility in water or acid.

In the case of caesium, the results from the first two tests have shown that it exists mainly in condensed form at 700 °C, which is incompatible with the hydroxide form. The results also show that the release of molybdenum in an oxidising atmosphere is greater than predicted. The molybdenum is also in excess relative to the caesium favouring the formation of caesium molybdate which is less volatile than the hydroxide. This behaviour is predicted by the SOPHAEROS code. The developers of the USNRC MELCOR code have now incorporated this species in their code.

In the case of iodine, excluding the gaseous forms which will be discussed later, the SOPHAEROS code predicts the formation of caesium or rubidium iodide (with equivalent properties). The PHEBUS tests, particularly FPT-2, show that this is not always the case. Depending on the conditions, caesium iodide may be absent or present, together with at least one other more volatile species. This chemical speciation is not currently reproduced correctly by the models. The IRSN experimental CHIP programme, described below, is expected to indicate the mechanisms involved and the species formed.

7.5.2.2 Interactions with the walls in the reactor cooling system

The fission product vapours may interact with the walls. Some elements, such as tellurium, may also be chemisorbed increasing the deposition. However, this does not occur if sufficient tin from the zircaloy cladding is transported with the tellurium. In this case, tin telluride is formed.

Fission product vapours condense on and interact with the metal walls. This is especially the case with caesium. This interaction has been demonstrated in a number of experimental programmes, including the PHEBUS-FP programme and the DEVAP programme [7.5_5] carried out at CEA Grenoble on behalf of the IRSN. Re-vaporisation tests carried out on deposits on samples from the PHEBUS-FP FPT-1 test have shown that the interactions between caesium and steel result in the formation of a number of species with a range of volatilities [7.5_6]. The deposited fission products may be revolatilised some time after the main phase of emission from the core. This may result in delayed emission.

7.5.2.3 Chemistry of iodine

7.5.2.3.1 In the reactor cooling system

Iodine is likely to combine with a large number of other elements in the reactor cooling system, including both fission products and structural materials. The main elements involved are caesium, rubidium, silver, indium and cadmium. Iodine may also be present in atomic form I, in molecular form I_2 or as hydroiodic acid HI. All these forms are gaseous under the conditions present in the reactor cooling system during a severe accident.

As a result of studies carried out following the TMI-2 accident, it was generally accepted that iodine was transported in the form of caesium iodide CsI. However, in safety studies such as those carried out during the evaluation of the NRC reference source term (NUREG 1465) [7.5_8], it has been assumed that some of the iodine (5%) may be emitted in a volatile form (I and HI). The determination of this fraction was not based on experimental results, but rather on accident sequences and thermodynamic calculations on a simple Cs-I-O-H system taking account of the possibility of kinetic limiting.

The results of the PHEBUS-FP tests have shown that the situation is more complex. They have shown that the iodine is not always transported mainly in the form of caesium iodide, and that many more elements must be considered in studies aimed at determining its chemical form. A considerable body of work has been carried out to compile and verify the thermodynamic data relating to the species that may potentially be involved, and this has resulted in an extended database for use in simulation codes such as SOPHAEROS. However, despite these efforts, this type of code still cannot reproduce the behaviour of iodine in all the situations explored during the PHEBUS-FP programme.

An important phenomenon, the very short term presence of gaseous iodine in the containment building during the PHEBUS-FP tests, cannot be explained by chemical reactions taking place within the containment building. This has been attributed to an injection of gaseous iodine from the reactor cooling system. The instantaneous fractions reached 30 % during the FPT-0 test and 4 % during the FPT-1 test [7.5_9], [7.5_10]. It should be noted that:

- In these two tests, the maximum fractions were measured when the concentration of hydrogen in the reactor cooling system was at a peak (~ 50 %).
- These estimates were based on the assumption of a the cold leg break. The results from FPT-1 tend to show that the gaseous fraction would be higher in the hot leg.
- The difference in concentration between the two tests (a factor of 30 times less iodine in FPT-0, carried out using very low irradiated fuel, than in FPT-1) suggests an effect due to the kinetics of the chemical reactions.
- The dependence of the gaseous iodine fraction on the concentration of hydrogen was less clear in the FPT-2 test.

The three PHEBUS tests, FPT-0, FPT-1 and FPT-2, were carried out using a silver-indium-cadmium alloy control rod, as used in a 900 MWe PWR. The most recent FPT-3 test was carried out using a boron carbide B_4C control rod. This material is used in the 1300 and 1450 MWe PWRs (and others) and is planned to be used in the EPR. Although the processing of the measurements is still in progress, it is sufficiently advanced to show that the fraction of iodine in gaseous form is much greater at over 80 % [7.5_11]. At this stage, it is only possible to suggest hypotheses, that will later need to be validated or rejected by specific experimental programmes, to explain this figure (see Section 7.5.3). These hypotheses include the absence of silver, indium and cadmium, reducing the number of elements available to combine chemically with the iodine, and the presence of large quantities of boric acid arising from the oxidation of the control rod which could combine with caesium preventing the formation of caesium iodide. The most volatile chemical forms of iodine may be favoured.

Care should be taken in attempting to extrapolate these results to accident sequences in a reactor. While the results from the FPT-0 and FPT-1 tests may be taken into account in studies to re-evaluate the source term, especially those at the IRSN in which an overall fraction of 5 % of gaseous iodine is assumed following a break in the reactor cooling system, additional studies will be needed in order to take account of the results from the PHEBUS-FPT-3 test. These studies must also cover the ultimate fate of the iodine within the containment building as discussed below.

7.5.2.3.2 Liquid phase within the containment building

The iodine fraction injected into the containment building in the form of aerosols will behave identically to the other aerosol particles (see Section 7.4). Most of these particles will be washed down into the sump water. The majority of the metal iodides (CsI, RbI, CdI_2, and InI) are soluble, with the notable exception of silver iodide AgI. The soluble iodides are dissolved in water forming I^- ions.

The large quantity of fission products released into the aqueous phase within the containment building results in a very high dose rate and the formation of radiolysis products from the water. These include reactive radicals such as OH^-, O_2^-, etc. A large number of chemical reactions occur, with the net result being the thermally activated and radiolytic oxidation of the iodide ions I^- into volatile iodine I_2. The formation of I_2 depends on many parameters, the most important being the pH of the water. If the water remains alkaline, the rate of production of I_2 is very low.

The water in the sump also contains organic materials originating mainly from the submerged paints. The iodine reacts with organic radicals to produce volatile organic iodides such as methyl iodide CH_3I, or low volatile compounds with higher molecular weights.

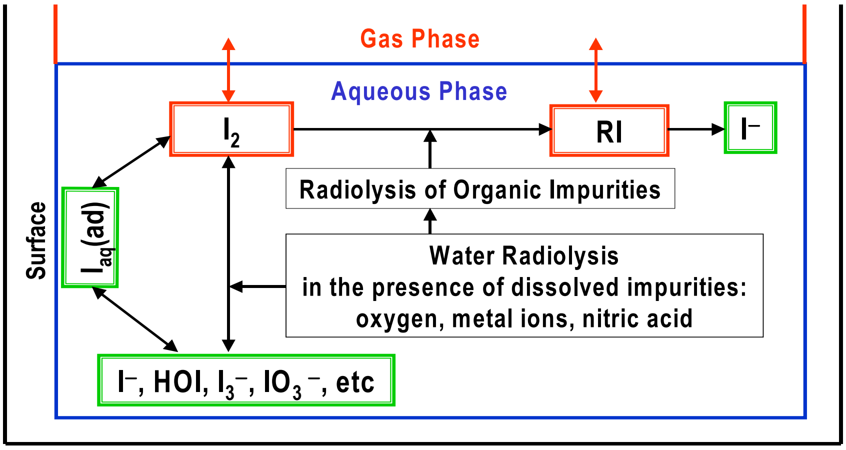


Figure 7.5-1: The main phenomena in the liquid phase

These reactions in the liquid phase (see Figure 7.5-1) have been studied in depth both experimentally and theoretically, and the associated phenomena are reasonably well understood [7.5_12], [7.5_13]. Some uncertainties remain in relation to certain impurities such as the NO_3^-/NO_2^- ions produced by the radiolysis of air, the Fe^{2+}/Fe^{3+} ions dissolved from the steel surfaces, and the Cl^- ions resulting from pyrolysis of the cables. The effects of these impurities are currently being studied in a number of experimental programmes, particularly in Canada and Switzerland.

Following the first PHEBUS-FP tests it was noticed that the silver released from the silver-indium-cadmium control rods could react with iodine in the liquid phase to form insoluble silver iodide. If there is a sufficient excess of silver relative to the iodine, the concentration of iodide I^- ions is greatly reduced, resulting in a very low production of gaseous iodine I_2. These phenomena have been quantified in dedicated experiments and models have been developed. Of particular note are the PHEBUS-RTF tests [7.5_14] carried out by AECL in Canada and the tests on the stability of silver iodide AgI carried out by PSI in Switzerland as part of the PHEBUS-FP programme [7.5_15].

It should be noted that the chemical reactions in the liquid phase depend on a number of boundary conditions including the temperature and the pH. The pH is not always easy to estimate if it is not under control, for example in the event of a failure of the spray system in direct mode.

7.5.2.3.3 Gaseous phase within the containment building

The volatile iodine present in the containment building in the gaseous phase originates from:

- Volatile iodine injected from a break in the reactor cooling system.
- Volatile iodine produced by radiolytic reactions in the sump water.

The transfer of volatile iodine from the sump is determined by classical laws of physics. Mass transfer models are available for both non-evaporating and, more recently, evaporating conditions.

Iodine present in the gaseous phase will interact with the various surfaces of the containment building, mainly metallic and painted ones. These include physical adsorption and desorption in addition to chemical reactions. These reactions are affected by the temperature and by radiation.

The available data relating to the adsorption and desorption of iodine results from both laboratory scale experiments and more integral tests such as the RTF tests [7.5_16] in Canada and the CAIMAN tests [7.5_17] in France. The parameters investigated were principally the type of paint, its aging, the temperature and dose rate. The derived correlations correspond to a first order kinetics.

The most important interaction from a safety point of view is that between iodine and paints as this results in the formation of organic iodides that are difficult to filter in the event of a filtered containment venting (U5 procedure). The rates of production of organic iodides have been established from the results of a large number of small-scale tests using samples of iodine-loaded painted surfaces exposed to radiation in an atmosphere representative of that in the containment building. These tests have shown that the effect of radiation is greater than that of temperature. The results of these tests have been used to develop semi-empirical models. Given the wide variations in the results and the difficulty in separating the influence of the various parameters, these models are only capable of reproducing experimental results to within an order of magnitude.

Inorganic iodine I_2 is oxidised when irradiated by the products of air radiolysis, ozone and the nitrogen oxides. The resulting iodine oxides and nitroxides are claimed to be non-volatile. A number of phenomena are involved simultaneously, including:

- The formation and destruction of oxidising ozone and NOx species under irradiation.
- The interaction of these oxidising species with metal and painted surfaces.
- The interaction of the oxidising species with iodine.
- The interaction of iodine with metal and painted surfaces.

Tests carried out in the past have used very high iodine concentrations, and the resulting models are difficult to extrapolate to real conditions. The results of more recent tests (the PARIS programme [7.1_9]) are still being analysed, but it is expected that they will further refine understanding of these phenomena.

Organic iodine CH_3I is also broken down under these conditions. The tests carried out showed that the rate of breakdown is proportional to the dose with the accelerating effect of the temperature being small.

One important mechanism for reducing the concentration of gaseous iodine within the containment building is its capture by droplets from the spray system [7.4_2]. The following phenomena are involved:

- The transfer of iodine in the gaseous phase towards the droplets.
- Transfers at the gas-droplet interface and within the droplets.
- The chemical reactions taking place in the liquid phase.

These chemical reactions are influenced by the pH of the droplets, with the capture being more efficient if the pH is alkaline, i.e. when the spray system is used in direct mode. The capture of organic iodine CH_3I is relatively inefficient compared with that of inorganic iodine I_2.

The RECI tests [7.5_18] carried out by the IRSN have indicated a potential interaction between iodine and the recombiners. The following phenomena are involved:

- Heating of the metallic iodide aerosols as they pass between the plates in the recombiners.
- Vaporisation and dissociation of the iodides.
- Quenching at the outlet of the recombiners with the formation of gaseous iodine and fine aerosols by nucleation.

These phenomena are well understood, but require further quantification, especially in relation to the effects of the chemical kinetics, in order to verify the true impact on the reactors operated by EDF.

7.5.2.4 Chemistry of ruthenium

As indicated in Section 7.1, the release of ruthenium can be significant in the presence of air. This type of situation can arise during a severe accident in a reactor if there is a break in the reactor pressure vessel, or during storage pool handling or emptying accidents [7.5_19]. The radiotoxicity of ruthenium is similar to that of iodine in the short term, and similar to that of caesium in the mid-term [7.5_20], hence the interest in this element. The chemistry of ruthenium is also complex.

Ruthenium, present in the fuel in metallic form, is released mainly in the form of dioxide vapour in the presence of air. As it cools down, the dioxide is converted into tetroxide vapour. This is then thermodynamically deposited by non-congruent condensation as the solid dioxide. This final reaction is kinetically limiting and some of the ruthenium remains in the tetroxide form, which is gaseous under the conditions prevailing within the containment building. These phenomena have been demonstrated during the RUSET tests [7.5_21], [7.5_22], [7.5_23], [7.5_24] carried out by AEKI, and confirmed by the tests carried out by VTT [7.5_25], [7.5_26] in Finland. During the RUSET tests, measurements were made of the partial pressures of gaseous ruthenium corresponding to an equilibrium between the gaseous tetroxide and deposited dioxide at temperatures of around 600 to 700 °C. As an approximation, this implies that below this temperature the chemistry is frozen and the tetroxide subsists in a metastable form.

Gaseous ruthenium tetroxide can interact with the walls of the pipework, becoming trapped. These effects have been studied experimentally by VTT [7.5_25], [7.5_26] in Finland. Deposits of RuO_4 were measured on alumina and steel tubes. The deposits on the alumina tubes were small, but they were considerable on the steel tubes, except in the presence of steam. The reasons for the effect of the steam are still not well understood. The conclusion to be drawn from these tests is that a significant proportion of the ruthenium may be injected into the containment building in gaseous form.

The behaviour of gaseous ruthenium within the containment building is currently the subject of an experimental study at IRSN Cadarache [7.5_27], [7.5_28] as part of the ‘ruthenium containment’ programme within the source term programme. This study will cover the phenomena of adsorption and desorption on steel walls and painted surfaces, and the effect of radiation via the radiolysis products from the air, such as ozone, which could result in the re-volatilisation of the deposited ruthenium or that dissolved in the water in the sump.

While the initial results are still being analysed, they do indicate that, despite the surface deposits, a significant fraction of the gaseous ruthenium could subsist in the atmosphere within the containment building, and that the deposited or dissolved ruthenium may be re-volatilised under the effect of the radiation. The kinetics of these mechanisms are still to be quantified before models can be developed to predict the quantity of ruthenium present in a gaseous form within the reactor containment building.

7.5.3 EXPERIMENTAL PROGRAMMES, MODELLING AND COMPUTER CODES

7.5.3.1 Chemistry in the reactor cooling system

In addition to the problems of gaseous iodine, most of the current work is concentrated on the interpretation of the results of the PHEBUS-FP and VERCORS HT tests. At the IRSN, this interpretation is being carried out using the 'extended database' version of the SOPHAEROS module of the ASTEC integral code. The simulated and experimental results are being compared mainly in relation to the volatility of the fission products. Following corrections to the database, the results so far have been encouraging, with the exception of those for iodine.

The full thermodynamic simulations from codes such as GEMINI and the results from SOPHAEROS do not reproduce the results of the PHEBUS tests correctly in relation to the behaviour of iodine. In particular, they do not predict the fractions of gaseous iodine in the reactor cooling system correctly. The experimental CHIP programme has been started with the aim of obtaining additional data on the iodine chemistry in the reactor cooling system from the point of view of both the thermodynamics and the chemical kinetics. There are two main aspects to this programme:

- The phenomenological aspect will study extended $\{X_1, X_2, ..., X_n, O, H\}$ systems containing around ten elements with the aim of obtaining data on the quantity of volatile iodine as a function of the elements present and the boundary conditions (nature of carrier gas, temperature, transit time, etc.). These experiments will be carried out at the IRSN in Cadarache.
- The analytical aspect will study restricted $\{X, I, O, H\}$ systems with the aim of obtaining kinetic and thermodynamic data for use in the development of models. The experiments will be carried out by the CNRS in Grenoble in collaboration with the IRSN.

In parallel with the CHIP experiments, detailed kinetic models similar to those described in [7.5_29] are being developed to interpret the data. These will eventually be used as the basis of simplified models to be incorporated in the SOPHAEROS module of the ASTEC integral code.

The specific case of ruthenium is being considered in the RUSET experimental programmes being carried out by AEKI and those being carried out by VTT. The results confirm that some of the ruthenium is transported into the containment building in the form of metastable gaseous RuO_4. A portion of this gaseous ruthenium is deposited on the steel pipework. The results of these tests are currently being interpreted and preliminary models are expected to be available by the end of 2006. These models are intended to be incorporated in the SOPHAEROS module of the ASTEC integral code.

7.5.3.2 Chemistry in the containment building

Most of the current experimental and modelling work is concentrated on the chemistry of iodine and ruthenium.

The EPICUR programme being carried out by the IRSN is studying the physical and chemical transformations iodine under irradiation with particular emphasis on the following:

- The formation of organic iodides in the gaseous phase from painted surfaces.
- The radiolysis of iodine in the aqueous phase.
- The radiolysis of iodine in the gaseous phase.

This programme began in 2005 and is scheduled to last three years. At the IRSN, the results are being interpreted using the IODE module in the ASTEC integral code. One expected result is an improvement to the models of the conversion of inorganic to organic iodine by the paint.

The PARIS experimental programme being carried out by Framatome-ANP on behalf of the IRSN is now completed and the results are currently being interpreted. The interpretation is using partly the mechanistic IODAIR module developed at IRSN. The improved simplified models will be incorporated in the IODE module of the ASTEC integral code.

The participation of PSI in the international Source Term programme will result in an extension of the experimental database available for use in the validation of the models. Their contribution will include the provision of hitherto unpublished results on the effects of impurities in the water in the sump on the volatility of the iodine. AECL are also expected to take part in the Source Term programme, although their precise contribution is not yet defined.

At the same time, the interpretation of the results of the PHEBUS-FPT-2 and FPT-3 tests is continuing, with the participation of a number of foreign partners. The FPT-3 test is the subject of particular attention. Initial analysis of the experimental results appears to show that the concentration of gaseous iodine declines rapidly from a high initial value, with considerable trapping by painted surfaces.

The current IRSN experimental programme on ruthenium will be completed in 2006. The main topics studied include:

- The adsorption and desorption of gaseous ruthenium on metal and painted surfaces.
- The re-volatilisation of the ruthenium deposited on the surfaces under irradiation and the action of ozone.
- The re-volatilisation of the ruthenium trapped in the liquid phase under irradiation and the action of ozone.

The interpretation of the results of these tests has begun. This process may reveal the need for additional tests.

7.5.4 SUMMARY AND OUTLOOK

Chemistry of fission products in the reactor cooling system

The PHEBUS-FP integral tests have called into question a number of paradigms relating to the chemistry of the fission products in the reactor cooling system, in particular caesium and iodine. Recent studies have also showed that the case of ruthenium needs to be re-examined.

The caesium is not necessarily transported in the form of the hydroxide CsOH as was previously assumed. It may exist in the form of the less volatile molybdate. Recent models take account of this fact. The re-volatilisation of deposits in the reactor cooling system has also been observed which may result in a delayed release into the containment building.

The case of iodine is more complex. It is not always transported in the form of caesium iodide. The PHEBUS-FP tests have shown that at least two other non-gaseous species may be present depending on the conditions (temperature and oxygen potential). A fraction of the iodine is also transported in gaseous form. The results of the first PHEBUS-FP tests and the results of specific studies using all the available experimental data indicated a value for this gaseous iodine fraction of 5 %. This value was used in the source term re-evaluation studies carried out by EDF and the IRSN. In the latest PHEBUS-FP test (FPT-3), this fraction exceeded 80 %. It should however be noted that the concentration of gaseous iodine in the vessel simulating the containment building fell rapidly during this test due to significant trapping by the painted surfaces. The interpretation of the behaviour of iodine from the results of this test has only just begun, and it will be necessary to await the results of this process before the consequences in terms of potential radioactive releases into the environment can be evaluated.

Considerable work has been undertaken with the aim of understanding the behaviour of the iodine. Critical reviews of the thermodynamic data available for the various species have been carried out resulting in successful simulations of the chemical speciation. However, these simulations do not currently reproduce all the observed effects correctly. There is a strong suspicion that the errors observed are due to phenomena associated with the chemical kinetics (incomplete reactions) that were not taken into account in current simulations performed at thermodynamic equilibrium. The experimental CHIP programme being carried out as part of the international Source Tem programme should identify the chemical reactions concerned and should enable the kinetic data needed to model them to be determined. Validated physical models describing the behaviour of iodine in the reactor cooling system are expected to be available by 2010.

Experiments performed in Hungary and Finland have shown that, in the presence of air, ruthenium may be transported into the containment building in the form of metastable gaseous RuO_4 with a certain fraction being retained by interactions taking place with the pipework. The results of these tests are currently being analysed. Validated models are expected to be available by 2008.

In general, there is little experimental data available regarding the re-volatilisation of deposited fission products. Addition experiments carried out within the CHIP facility may result in further data becoming available.

Chemistry of the fission products in the containment building

The two fission products whose chemistry in the containment building may have a significant impact on radioactive releases are iodine and ruthenium. This is due to the fact that they can both exist in gaseous form under the conditions prevailing within the containment building.

Iodine

The importance of iodine has been known for a long time and many experimental and theoretical studies have been carried out. These studies have investigated the chemistry of iodine in both the liquid and gaseous phases, its interactions with surfaces, and the effects of radiation on the various processes. Experiments have been performed at the laboratory scale, at intermediate levels, and in the PHEBUS-FP integral tests. The models available to describe this chemistry are either mechanistic, covering several hundred reactions, or simplified models covering a more restricted range of reactions. The simplified models are more generally used in the integral simulation codes.

Despite this work, a number of uncertainties remain that have a significant impact on the levels of radioactive releases. This has been demonstrated in the recent OECD International Standard Problems (ISP) ISP-41 and ISP-46. This is especially the case in relation to the formation of organic iodides in the gaseous phase. The results of the experimental EPICUR programme being carried out as part of the international Source Term programme are expected to provide new data in this field, and in relation to some aspects of radiolysis in the liquid and gaseous phases. This data, together with that from the recently completed PARIS programme, should lead to the availability of improved and validated models by 2010.

The impact of spray on gaseous iodine, partially capturing the iodine in the droplets, has been quantified and validated models are now available.

The recent demonstration of interactions between iodine aerosols and the recombiners capable of producing gaseous iodine has led to a requirement for additional experimental work to quantify the impact of this phenomenon on the concentration of gaseous iodine in the containment building. This topic is currently being studied within SARNET.

Ruthenium

The demonstration of the possibility that some of the ruthenium could be present in gaseous form within the containment building has raised questions relating to the ultimate fate of this element within the building. Recent experimental work has investigated the adsorption and desorption of gaseous ruthenium on the surfaces, together with the re-volatilisation of the ruthenium deposited or trapped in the liquid phase under irradiation. The results obtained show that ruthenium is likely to persist in gaseous form within the containment building. The next stage will be to quantify the kinetics of the various phenomena in order to be in a position to evaluate the concentration of gaseous ruthenium within the containment building during an accident.

TABLES IN CHAPTER 7

FIGURES IN CHAPTER 7

REFERENCES IN CHAPTER 7

[7.1_1] H. Kleykamp, "The chemical state of the fission products in oxide fuels", Journal of Nuclear Materials 131 (1985) 221-246.

[7.1_2] Y. Pontillon et al. "Lessons learnt from VERCORS tests. Study of the active role played by UO2-ZrO2-PF interactions on irradiated fuel collapse temperature", Journal of Nuclear Materials 344 (2005) 265-273.

[7.1_3] S. Valin, "Etude des mécanismes microstructuraux liés au relâchement des gaz de fission du dioxyde d'uranium irradié", Thèse à l'Institut National Polytechnique de Grenoble (1999).

[7.1_4] H. Albrecht et al. "Release of Fission and Activation Products during Light Water Reactor Core Meltdown", Nuclear Technology 46 (1979), 559-565.

[7.1_5] R.A. Lorenz, M.F. Osborne, "A summary of ORNL Fission Product release tests with recommended release rates and diffusion coefficients", ORNL/TM-12801 – NUREG/CR-6261 (1995).

[7.1_6] D.S Cox et al. "Fission Product releases from UO2 in air and inert conditions at 1700-2350 K: analysis of the MCE-1 experiment", ANS-ITM on the safety of thermal reactors, Portland, USA (1991).

[7.1_7] T. Kudo et al. "VEGA; an experimental study of radionuclides release from fuel under severe accident conditions", ANS Water reactor fuel performance meeting, Kyoto, October 2005.

[7.1_8] G. Ducros et al. "Fission Product release under severe accidental conditions; general presentation of the program and synthesis of VERCORS 1 to 6 results", Nuclear Engineering and Design 208 (2001) 191-203.

[7.1_9] B. Clément, "The PHEBUS Fission Product and Source Term International Programme", Int. Conf. Nuclear Energy for New Europe, Bled, Slovenia (2005).

[7.1_10] W. Plumecoq et al. "Fission product release modelling in the ASTEC integral code: the status of the ELSA module", 8th Int. Conf. On CANDU fuel, Honey Harbour, Ontario (2003).

[7.1_11] G. Nicaise et al. "Analysis of accidental sequence tests and interpretation of fission product release: interdependence of Cs, Mo and Ba release", 8th Int. Conf. On CANDU fuel, Honey Harbour, Ontario (2003).

[7.1_12] R.D. Barrand, R.S. Dickson, Z. Liu, D.D. Semeniuk, "Release of fission products from CANDU fuel in air, steam and argon atmosphere at 1500-1900°C: the HCE3 experiment", Canadian Nuclear Society, 6th Int. Conf. CANDU fuel, Niagara Falls, 26-30 Sept. 1999.

[7.2_1] M. P. Kissane, I. Drosik, "Interpretation of fission-product transport behaviour in the PHEBUS FPT-0 and FPT-1 tests", Nuclear Engineering and Design, vol. 236, pp. 1210-1223 (2006).

[7.2_2] L. Talbot et al, "Thermophoresis of particles in a heated boundary layer, Journal of Fluid Mechanics", 101, pp 737_758 (1980).

[7.2_3] A. Zoulalian, T. Albiol, "Analysis of fluid-wall interactions in an open material system where the residence time distribution of a fluid is known. Application of aerosol deposition by diffusiophoresis", Canadian Journal of Chemical Engineering, vol. 73, Issue 6, pp 800_897, December 1995.

[7.2_4] M. Missirlian, "Modélisation des dépôts d'aérosols par diffusiophorèse dans un écoulement. Application aux réacteurs à eau sous pression en situation accidentelle", Thèse Université de Provence / Aix-Marseille 1 (1999).

[7.2_5] C. Housiadas, K. Müller, J. Carlsson, Y. Drossinos, "Two-dimensional effects in thermophoretic particle deposition: the PHEBUS-FP steam generator", Journal of Aerosol Science, vol.32, supplement 1, pp. S1029-S1030.

[7.2_6] A. Zoulalian, T. Albiol, "Evaluation of aerosol deposition by thermo and diffusiophoresis during flow in a circular duct – application to the experimental programme 'Tuba diffusiophoresis", Canadian Journal of Chemical Engineering, vol. 76, Issue 4, August 1998, pp. 799-805.

[7.2_7] A. De loss Reyes et al., "International Standard Problem 40 – Aerosol Deposition and Resuspension", Final Comparison Report, NEA/CSNI/R(99)4 (1999).

[7.2_8] S. Güntay, D. Suckow, A. Dehbi, R. Kapulla, "ARTIST: introduction and first results, Nuclear Engineering and Design", 231 (2004), pp 109-120.

[7.3_1] W. Shoek, M. Wagner-Ambs, "Resuspension of fission product aerosols from the boiling sump", Proceedings of the International Centre for Heat and Mass Transfer 1990, pp.539-546.

[7.4_1] V. Saldo, E. Verloo, A. Zoulalian, "Study on aerosol deposition in the PITEAS vessel by settling, thermophoresis and diffusiophoresis phenomena", J. Aerosol Science, vol 29, suppl.1, pp. S1173-S1174 (1998).

[7.4_2] V. Fournier-Bidoz, V. Layly, D. Roblot, J. Vendel, "Efficiency of the PWR spray system – information derived from the CARAIDAStests", Rapport Scientifique et Technique IRSN 2000.

[7.4_3] R.K. Hilliard, A.K. Postma, "Large-scale fission product containment tests", Nuclear technology, vol. 53, Issue 2, May 1981, pp.163_175.

[7.5_1] "Insights into the control of the release of iodine, cesium, strontium and other fission products in the containment by severe accident management", Rapport NEA/CSNI/R(2000)9.

[7.5_2] E.C. Beahm, C.F. Weber, T.S. Kress, "Iodine chemical forms in LWR Severe Accidents", NUREG/CR-5942 (ORNL/TM-11861), April 1992.

[7.5_3] A.M. Beard, L. Codron, A. Mason, "Boric acid experiments: vaporisation, deposition on Inconel and interaction with caesium hydroxide", EUR 15766/1 EN, Commission of the European Communities.

[7.5_4] E. Hontanon, M. Lazaridis, Y. Drossinos, "The effect of chemical interactions on the transport of caesium in the presence of boron", J. Aerosol Sci., vol.27, n°1, pp. 19-38 (1996).

[7.5_5] G. Le Marois, M. Megnin, "Assessment of fission product deposits in the reactor coolant system : the DEVAP program", Nuclear Safety, vol.35, Issue 2, pp. 213-222, July 1994.

[7.5_6] A.B. Anderson, A. Auvinen, P.D.W. Bottomley, C.J. Bryan, N.E. Freemantle, J.P. Hiernaut, J.K. Jokiniemi, A.F. Kingsbury, A.T. Tuson, "Revaporisation tests on samples from PHEBUS fission products", final report, European Commission 4th framework program, Report ST:RVP(00)-P029 (2000).

[7.5_7] B. Clément, N. Hanniet-Girault, G. Repetto,D. Jacquemain, A.V. Jones, P. von der Hardt, "LWR Severe Accident Simulation: Synthesis of the results and interpretation of the PHEBUS FP experiment FPT0", Nuclear Engineering and Design, 226, pp. 5-82, (2003).

[7.5_8] L. Suffer, S.B. Burson, C.M. Ferrell, "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465 (1995).

[7.5_9] N. Hanniet, D. Jacquemain, "An overview of the iodine behaviour in the first two PHEBUS tests FPT0 and FPT1, Proceedings of OECD Workshop on Iodine Aspects in Severe Accident Management", Vantaa (Finland), NEA/CSNI/R(99)7.

[7.5_10] N. Hanniet, S. Dickinson, F. Funke, A. Auvinen, L. Herranz, E. Kraussmann, "Iodine behaviour under LWR accident conditions: learnt from analyses of the first two PHEBUS-FP tests", Nuclear Engineering and Design, 236, pp. 1293-1308, (2006).

[7.5_11] Ph. March et al., "First Results of the PHEBUS-FPT3 test, Proceedings of ICÔNE 14, 14th International Conference on Nuclear Engineering", July 17-20, 2006, Miami, Florida, USA.

[7.5_12] J.C. Wren, J.M. Ball, G.A. Glowa, "The Chemistry of Iodine in Containment", Nuclear Technology, (129) 297, (2000).

[7.5_13] E. Krausmann, "A state-of-the art report on iodine chemistry and related mitigation mechanisms in the containment", Report EUR 19752 EN (2001).

[7.5_14] J.M. Ball, G.A. Glowa, J.C. Wren, "Summary of the PHEBUS Radioiodine Test Facility", Proceeding of the 4th Technical Seminar of the PHEBUS-FP Program, Marseille, France, March 2000 – Non-public reference.

[7.5_15] S. Güntay, R.C. Cripps, B. Jäckel, H. Bruchertseifer, "On the radiolytic decomposition of colloidal silver iodide in aqueous solution", Nuclear Technology, 150, pp 303-314, 2005.

[7.5_16] G.J. Evans, W.C.H. Kupferschmidt, R. Portman, A.S. Palson, G.G. Sanipelli, "Radiochemical analysis of iodine behaviour in the Radioiodine Test Facility, Journal of Radioanalytical and Nuclear Chemistry", vol 180, issue 2, pp 225-235, 1994.

[7.5_17] L. Cantrel et al., "Radiochemistry of iodine : Outcomes of the CAIMAN program", submitted to Nuclear Technology.

[7.5_18] F. Deschamps, J.C. Sabroux, "Etude de la production d'iode gazeux par un aérosol d'iodure de césium soumis à des températures élevées dans l'air humide", ASFERA, Actes du 18e congrès français sur les aérosols, Paris 11-12 December 2002.

[7.5_19] D.A. Powers, L.N. Kmetyk, R.C. Schmidt, "A review of the technical issues of air ingression during severe accidents", US NRC NUREG/CR-6218 (1994).

[7.5_20] D.J. Alpert, D.I. Chanin, L.T. Ritchie, "Relative importance of individual elements to reactor accident consequences assuming equal release fractions", NUREG/CR-4467, March 1988.

[7.5_21] L. Matus, O. Prokopiev, B. Alföldy, A. Pintér, Z. Hózer, "Oxidation and release of ruthenium in high temperature air", IRSN-AEKI PHEBUS-FP Programme Agreement, Report on in-kind contribution of AEKI part 1, November 2002 – Non-public reference.

[7.5_22] L. Matus et al., "Oxidation and release of ruthenium from short fuel rods in high temperature air, IRSN-AEKI PHEBUS FP Programme Agreement", Report on in-kind contribution of AEKI part 2 and part 3, December 2004 – Non-public reference.

[7.5_23] Péter Windberg, "CODEX and RUSET Air Oxidation Experiments", International Conference Nuclear Energy for New Europe 2004, Portoroz, Slovenia, September 6-9.

[7.5_24] I. Nagy et al., "Oxidation and release of ruthenium from short fuel rods above 1500°C", EUR 21752 EN, April 2005.

[7.5_25] U. Backman, M. Lipponen, A. Auvinen, J. Jokiniemi, R. Zilliacus, "Ruthenium behaviour in severe nuclear accident conditions", final report, VTT report PR03/P27/04, June 2004.

[7.5_26] U. Beckmann, M. Lipponen, A. Auvinen, U. Tapper, R. Zilliacus, J. Jokiniemi, "On the transport and speciation of ruthenium in high temperature oxidising conditions", Radiochimica Acta, 93, 297-304 (2005).

[7.5_27] C. Mun, L. Cantrel, C. Madic, "Study of the ruthenium fission product behaviour in the containment of a nuclear reactor in case of a severe accident occurring in oxidizing conditions", proc. Nuclear Energy for New Europe, Bled, Slovenia, September 5-8 2005.

[7.5_28] C. Mun, L. Cantrel, C. Madic, "A literature review on ruthenium behaviour in nuclear power plants severe accidents", to be published in Nuclear Technology.

[7.5_29] L. Cantrel, E. Krausmann, "Reaction kinetics of a fission product mixture in a steam-hydrogen carrier gas in the PHEBUS primary circuit", Nuclear Technology, vol. 144 1, pp.1-15, October 2003.

CHAPTER 8 : SEVERE ACCIDENT CODES – CURRENT DEVELOPMENT AND VALIDATION STATUS

The numerical simulation of severe accidents general adopts a two-tier approach:

- integral codes or code systems that simulate the entire accident rapidly, from the initiating event to the possible release of radionuclides outside the containment and taking into account the main safety systems,
- detailed or mechanistic codes that provide a finer simulation of the phenomena involved a finer simulation of single phenomenon or group of phenomena (core degradation or hydrogen risk in the containment).

The three integral codes discussed in this chapter share a number of common features: they handle phenomena exhaustively (steam explosion and containment mechanical strength are the only points not dealt with), they process coupled phenomena (simplified to varying degrees), they have a modular design (but to varying degrees) and they process accident scenarios at a high computing speed (from 1 to 10 hours per accident day). They are used for reactor safety studies, in particular for source term assessment, Level 2 PSAs and studies carried out to support accident management (prevention and mitigation).

The process of validating these codes by comparison with experimental results (Table 8.1-1 illustrates the main international programmes used by the three codes) is identical:

- the codes are qualified through "separate-effect" or "coupled-effects" analytical tests, concerning one or more physical phenomena or component response,
- they are then verified on the basis of integral experiments, frequently performed on large-scale facilities to check that phenomena are correctly coupled and that none of them has been overlooked.

8.1 ASTEC

The ASTEC (for **A**ccident **S**ource **T**erm **E**valuation **C**ode) integral code is developed jointly by the IRSN and its German counterpart, the GRS [8.1_1]. It plays a leading role in SARNET, the European FP6[6] network of excellence, as it gradually integrates in the form of models all the knowledge generated by the network and because the partners carry out a considerable amount of work on validation and reactor applications [8.1_2]. One of the goals of SARNET is to see ASTEC become the European reference code.

6 Technological Research and Development Framework Programme

8.1.1 CURRENT CAPABILITIES OF THE CODE

The ASTEC code simulates all stages of the accident for situations where the reactor is in operation and takes into consideration the main safety systems (controlled blowdown of reactor cooling primary and secondary systems, accumulators, safety injection, recombiners, containment spraying and venting/filtering, etc.). It is also used for preparation and interpretation of experimental programmes. Figure 8.1-1 illustrates the different modules making up the code and shows how inter-module coupling is managed.

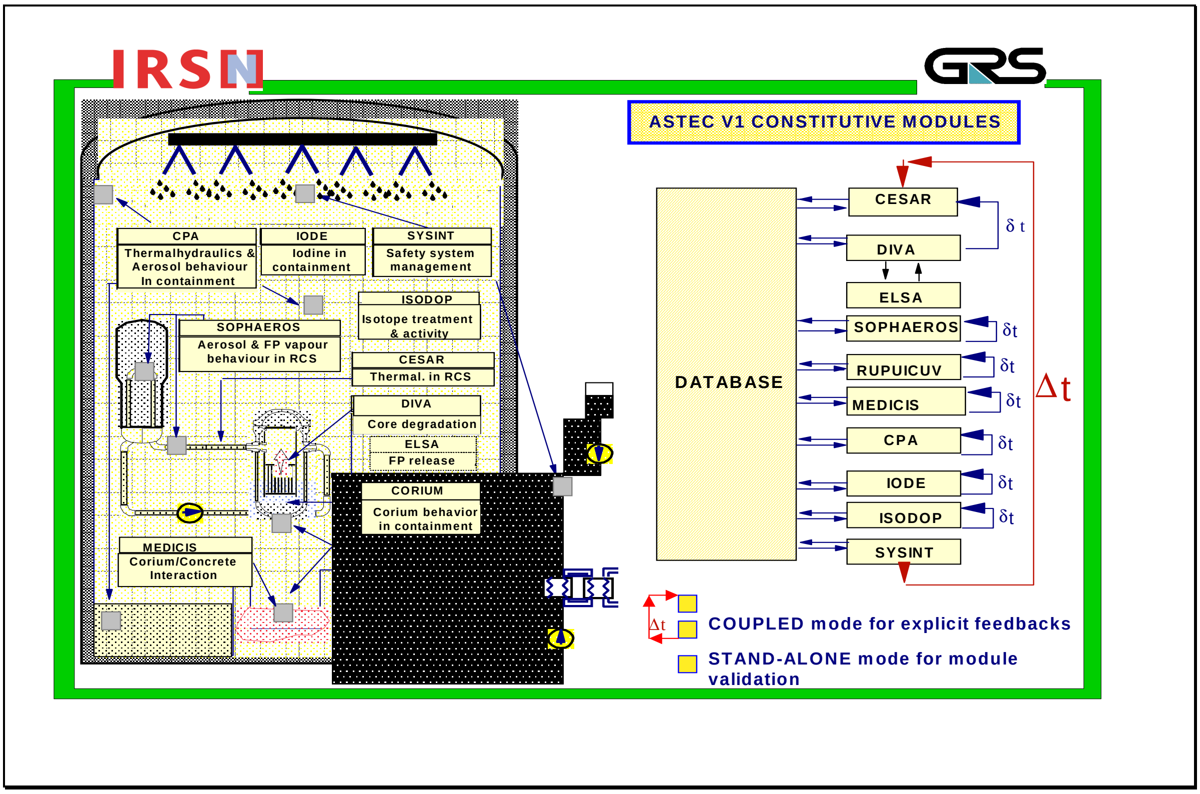


Figure 8.1-1: ASTEC V1 Structural Diagram

The latest version, V1.2 rev1, was released for use by 28 European and 6 Russian and Asian partners at the end of 2005. It applies to PWR 900 and 1300 MWe and VVER[7]-440 and 1000 units. It adopts a five-equation numerical approach for the reactor cooling system (RCS) thermal-hydraulics and a "zone-based" 0D approach for molten corium-concrete interaction and containment thermal-hydraulics. Using a PC or workstation (SUN, DEC. etc.), it takes a few hours to simulate one day of an accident.

Almost all the models used are at the state-of-the-art, especially as far as fission product behaviour is concerned, where all the lessons learnt from the PHEBUS-FP experimental programme and years of analytical testing around the world have been taken into account.

The main limit of this latest version is to do with core degradation, which is dealt with using 1D corium flow models along the fuel rods. This means that no consideration is given to the radial movement of

[7] Russian-designed PWR – literally translated as water-water energetic reactor

corium within the core or on the lower core plates and, therefore, its relocation at the lower head after passing through the core shroud (behaviour observed during TMI-2 accident). Preliminary comparison with the more advanced 2D flow models used in the IRSN's ICARE2 mechanistic code has revealed the impact of this 1D modelling on corium flow kinetics into the vessel lower plenum and on in-vessel hydrogen production. The masses of corium found in the lower plenum at the time of failure, however, do not seem to vary significantly with the type of model used. This shortcoming will be rectified in ASTEC versions to be released as of 2008.

The main developments or improvements in the latest models based on data from the experiments mentioned in brackets are as follows:

- core degradation: improved processing of corium heat transfer in the lower head (BALI at the CEA), new model added to simulate the fragmentation of corium as it falls into the water remaining in the lower head (FARO at the JRC), new model covering lower head mechanical strength (LHF and OLHF at the SNL),
- FP release: improvements in modelling the release of semi-volatile species (such as Ba and Mo), structural materials (Fe, Zr) and control rod Ag-In-Cd (VERCORS at the CEA, PHEBUS-FP at the IRSN),
- FP transport in the system: deposits in various coolant flows (parallel channels in the core, RCS intact loops, reverse flows, etc.), model added to cover FP nucleation at their arrival into the containment, thermo-chemistry databank for FP chemistry in gas phase checked against international references,
- MCCI: evolution of corium pool configurations (homogeneous, stratified, inversion of layers), corium cooling by water injection (in particular, quench front progress through the crust) (OECD-CCI, BETA, ACE),
- hydrogen combustion in containment: model added to cover turbulent deflagration conditions,
- behaviour of iodine in the containment: metal iodide dissociation in recombiners (RECI at the IRSN) and formation/destruction of radiolysis products in the air (PARIS at Framatome-Erlangen).

At present, ASTEC is used intensively at the IRSN as part of Level 2 PSA studies on a French 900 MWe PWR and sensitivity studies are carried out as part of a source-term review. This work involves the analysis of a considerable number of scenarios proposing different initiating events and the implementation of safety systems. The main areas of uncertainty concern high-pressure sequences, in particular the risks associated with induced breaks in steam generator tubes (leading to a possible containment bypass) and DCH, and sequences involving water injection on a damaged core. The problem in the last mentioned sequences is to determine whether core failure can be avoided depending on core state at the time of injection and on injection flow rate (the IRSN uses the ICARE/CATHARE mechanistic code to back up modelling work on these sequences).

Within the SARNET context, ASTEC is used by twenty or so European partners to calculate various sequences on PWR 900, Konvoi 1300, Westinghouse 1000, VVER-440 and VVER-1000 reactors.

8.1.2 VALIDATION STATUS

For more than ten years, intensive validation work has been carried out on the codes from which ASTEC was developed (ESCADRE, RALOC and FIPLOC), leading to a solid reference base. The basic validation matrix, composed of some thirty tests from Table 8.1-1, is applied whenever a significant version is delivered, to cover the main phenomena involved in severe accidents and to identify more clearly uncertainties in the models. For each module, validation is complemented by less frequent, but extensive, campaigns focusing on all available tests (in 2005, for example, about forty tests were performed on the subject of FP release). Validation work is also carried out by partners, within the SARNET context, on reference experiments such as inter-code comparison exercises selected by the OECD (ISP) and on tests dedicated to VVER plants. Examples of this work include BETHSY 9.1b (ISP27) and PACTEL (ISP33) for RCS thermal-hydraulics, KAEVER (ISP44) for aerosol behaviour in the containment and PHEBUS-FPT1 (ISP46) for the comprehensive overall study of an accident.

Taken together, V1 versions have undergone a total of more than 120 tests. On the whole, the results of this validation work have been satisfactory and show that the code reflects the state of the art in terms of understanding and modelling. Figure 8.1-2 and Figure 8.1-3 respectively show the example of the qualification of the CESAR module on the BETHSY 9.1b thermal-hydraulic test (CEA), which reproduces the scenario of a 2" break in the RCS cold leg without high-pressure safety injection, and the qualification of the IODE module on the CAIMAN 97/02 test (CEA), which simulates the production of molecular and organic iodine in the gaseous phase in the containment.

Improvements are expected in fields where new experimental programmes have been launched to increase knowledge and reduce uncertainties. These include: iodine chemistry in the RCS (CHIP at the IRSN); heat flux spatial distribution in the corium pool during MCCI (ARTEMIS and VULCANO at the CEA, OECD-CCI); iodine chemistry in the containment (EPICUR at the IRSN); FP retention on the secondary side of steam generators (ARTIST at the PSI); DCH (DISCO at the FZK); effect of high fuel burnup and of MOX fuel on core degradation and fission product release (VERCORS then VERDON at the CEA). Whatever the code used, modelling the reflooding of intact and damaged cores is always a complex task, as demonstrated by applications to the QUENCH and CORA tests at the FZK.

Table 8.1-1 - Illustration of the main experimental programmes used for code qualification

Physical process	Programme name	Organisation (country)
Integral tests	TMI-2 accident	-
	LOFT-LP- FP2	INEL (USA)
	PHEBUS-FP	IRSN (France)
Core degradation	CORA (7,12,13….)	FZK (Germany)
	QUENCH (01,04,06/ISP45, 07….)	FZK (Germany)
Fission product release	ORNL (VI-2 to VI-7)	ORNL (Canada)
	VERCORS (1 to 6)	CEA (France)
Aerosol transport in the reactor cooling system	FALCON	AEAT (GB)
	VERCORS (HT1 to HT3)	CEA (France)
	LACE	INEL (USA)
Vessel mechanical failure	LHF-OLHF	SNL (USA)
Heat transfer in corium molten pools	COPO	VTT (Finland)
	UCLA	UCLA (USA)
	BALI	CEA (France)
Fragmentation of corium in water	FARO (L06,L08,L11,L14…)	JRC Ispra
Corium entrainment during DCH	ANL (U1B…)	ANL (USA)
	Surtsey (IET-1, IET-8B…)	SNL (USA)
	DISCO (C, H)	FZK (Germany)
Molten Corium-Concrete Interaction	BETA (v5.1,v5.2,v6.1…)	FZK (Germany)
	SWISS	SNL (USA)
	OECD-CCI (1, 2…)	ANL (USA)
	ACE (L2, L5…)	ANL (USA)
Iodine chemistry in containment	ACE/RTF	AECL (Canada)
	CAIMAN	CEA (France)
Containment thermal-hydraulics	NUPEC (M4.3, M7.1….)	NUPEC (Japan)
	VANAM-M3 (ISP 37...)	Battelle (Germany)
	TOSQAN (ISP-47…)	IRSN (France)
	MISTRA (ISP-47…)	CEA (France)
Hydrogen combustion in the containment	HDR (E12.3.2…)	Battelle (Germany)
	RUT	RRC-KI (Russia)

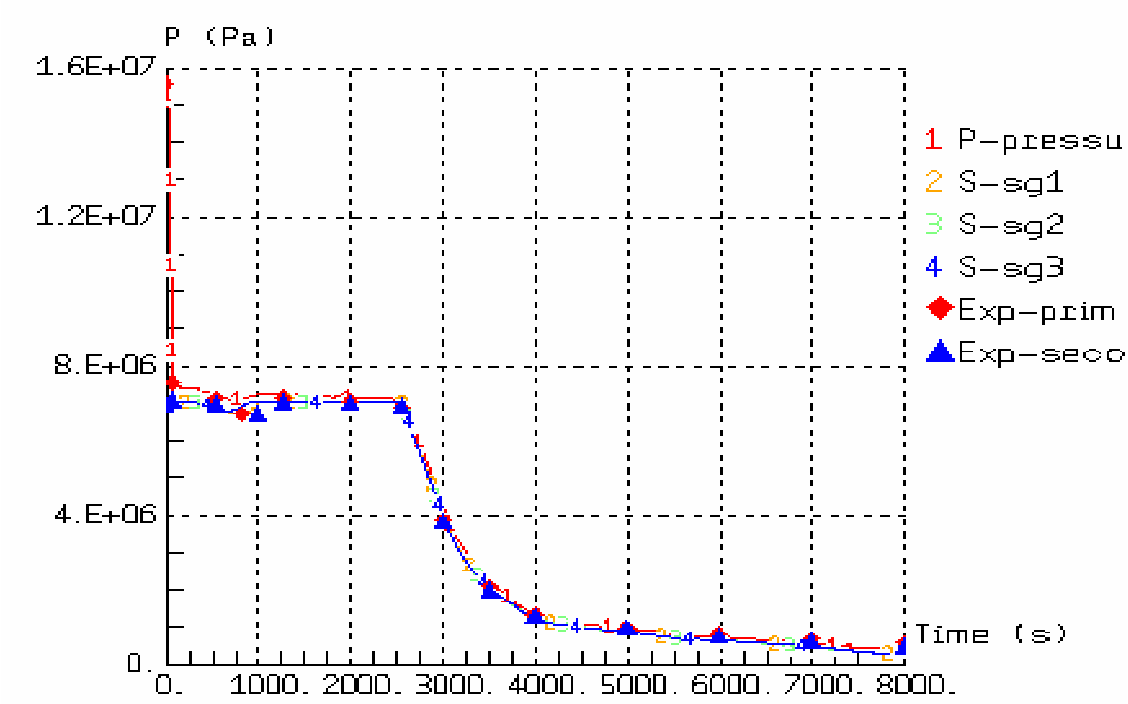


Figure 8.1-2 - ASTEC qualification on the BETHSY 9.1b test: pressure trends in the reactor cooling primary and secondary systems

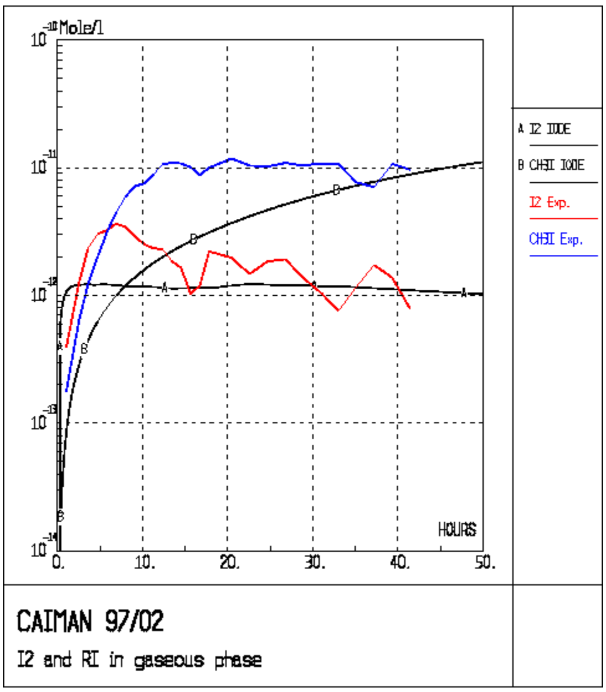


Figure 8.1-3 - ASTEC qualification on the CAIMAN 97/02 test: iodine trend in the gaseous phase in the containment

In addition to comparison to experimental results, the ability of the code to compute all accidents liable to occur in the reactors must be demonstrated. To this end, the code is applied to the main scenario families (breaks in the reactor cooling primary and secondary systems, loss of power supply, etc.), introducing variants to take into account the activation of safety systems, for example. This method can demonstrate that the trends based on results are physically credible. Benchmarks are then made using other codes: within the SARNET context, the above-mentioned sequences are being compared using the MELCOR and MAAP4 integral codes and detailed codes such as ICARE/CATHARE and CONTAIN, for example. These benchmarks serve to identify the models responsible for any discrepancies observed from one code to another, thereby helping to measure the uncertainty remaining in the assessment of certain physical phenomena.

8.1.3 CODE UPGRADE PROSPECTS

In the short term, ASTEC will be used intensively as part of the IRSN's Level 2 PSA work on the P4-series plants, which began in 2006. The development version is currently being tested on large-break and SGTR scenarios which had not been considered for the earlier versions.

The main model developments to be integrated in the next versions (V1.3 delivered in 2006 and V1.4 in 2007) result from the interpretation of the experimental programmes mentioned above. At the same time, it is planned to adapt existing models to simulate ex-vessel core catchers in a reactor like the EPR. The CEA and the IRSN also plan to work together closely to model in-vessel corium retention by external cooling of the vessel lower head.

The specifications of the future "V2" versions of ASTEC will integrate requirements expressed at the end of 2005 by the SARNET partners with regard to models, documentation and user support. The core degradation models of the IRSN's mechanistic code, ICARE2, are now being integrated into ASTEC. This will produce two types of model: fast simplified models for integral application and detailed models (2D corium movement in the core and through the vessel internals, corium behaviour in the lower head) for zooms on in-vessel phenomena. The first version, V2.0, to be released in 2008 should also be applicable to the EPR and to the reflooding of degraded cores, and include a complete modelling of ruthenium radiochemistry.

Other developments are scheduled in the following versions to model air and gas ingress resulting from MCCI in the vessel following failure, shutdown states and comprehensive coverage of iodine chemistry in the RCS (feedback from the CHIP programme, possibly taking chemical kinetics into account). At the same time, work will be in progress to adapt models to BWRs[8] (with partners KTH in Sweden and IKE in Germany as well as support from the PSI in Switzerland) and CANDU reactors (with partners INR in Romania, AECL in Canada and BARC in India) and to increase the code's computing speed for possible use in severe accident simulators.

[8] Boiling Water Reactors

8.2 MAAP

Development work on the MAAP computer code began in the USA in the early 80s. The code was intended a) to meet the requirements of physical studies in support of PSAs and b) to contribute to the IDCOR (Industry Degraded Core Rulemaking) research programme, which involved some sixty American industrialists.

When the IDCOR programme came to an end, EPRI became owner of the MAAP code, although development work was still carried out by Fauske&Associates, Inc (FAI).

Many nuclear operators have acquired a MAAP licence and use the code in their safety studies. They have formed a users' group (known as the "MAAP Users Group") representing more than 40 organisations.

EDF uses MAAP for severe accident studies, including Level 2 PSA, hydrogen recombiner system design, reassessing the source term for severe accidents, SAMG support studies, as well as studies of the DCH risk and slow pressurisation in the containment building.

Since it acquired the code in 1991, EDF has developed its own skills in MAAP development and validation, complementing FAI activities. EDF began to take this development approach a step further in 1996 by making its own MAAP versions integrating specific contributions. The version of MAAP currently in use at EDF is 4.04d, EDF's 4^{th} version of the MAAP 4.04 code.

8.2.1 CURRENT CAPABILITIES OF THE CODE

The MAAP computer code is used to process PWR or BWR operation under accident conditions (specific versions are also available for CANDU and VVER reactors). It focuses particularly on core meltdown sequences, whatever the initial status of the installation (reactor at power or shut down).

MAAP is a modular scenario code, representing the reactor cooling primary system, the secondary system, the engineered safeguard systems, and the containment and auxiliary buildings.

Functional modelling is particularly well adapted for examining the impact of operator action on the sequences.

One of the code's advantages is its fast computing time. It only takes about 2 hours CPU (with a PC in a Linux environment) to simulate 24 h in real time of a sequence involving core degradation.

As the reactor cooling circuit and containment building are coupled, the thermal-hydraulic conditions in the reactor building are dependent on thermal-hydraulic changes in the RCS and boundary conditions at break. This coupling can be used, for example, to take into account automatic actions taken on the RCS when a pressure threshold is reached in the containment building. It also takes into account containment building feedback effects on the RCS (e.g. simulation of shutdown states with the RCS open).

Fission product transport (emission during core degradation, migration in the RCS and containment building, chemistry) can be modelled to determine release to the environment or surface and volume contamination in the buildings.

MAAP is a nodal code. It solves mass and internal energy conservation equations for each control volume. Momentum conservation equations are not differential equations and can be considered as Bernoulli equations.

The RCS is represented by 14 volumes, excluding the pressuriser. The containment building can be represented by 30 volumes at the most (note that the nodalisation system adopted for EDF reactors comprises 12 to 14 volumes). The core is modelled axisymmetrically with 175 mesh cells.

Figure 8.2-1 gives an illustrated example of core degradation as represented by MAAP.

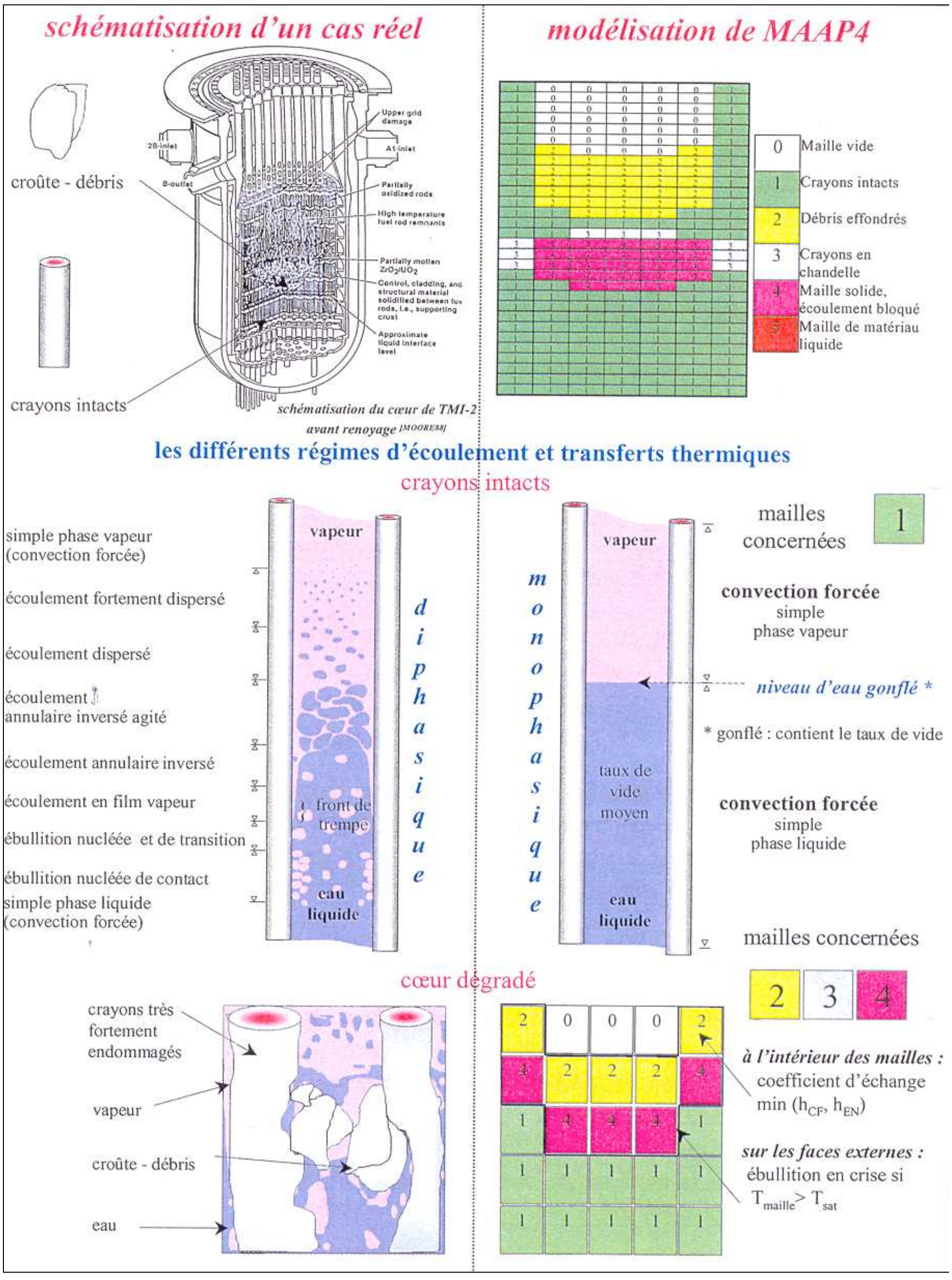


Figure 8.2-1: Comparison between the MAAP4 modelling of various flows during core degradation and a schematic illustration of an actual flow

A new version of MAAP, MAAP4.05a, will be commissioned during the second half of 2006. Developed by EDF R&D, it is based on the EPRI standard 4.05 version. The latest improvements of the code include:

- consideration of boron carbide thermochemistry (degradation, oxidation and B_4C melt): integration of three separate models, based on the oxidation laws taken from the MELCOR and ICARE codes, together with separate-effect experiments carried out at FZK (BOX RIG),
- addition of a U-Zr-O oxidation model (qualified during QUENCH test No. 07),
- refinement of the axial mesh of the reactor pressure vessel lower head,
- improved release model for U, Pu and Te (VERCORS tests),
- improved silver-indium-cadmium release model qualified on PHEBUS-FPT1,
- corium entrainment out of the reactor pit, following several possible pathways in the containment qualified on Lacomera1-DISCO-H (in the event of a high-pressure ejection accident in the reactor vessel),
- model of the EPR heavy reflector.

8.2.2 VALIDATION STATUS

In practice, the physical validation of the MAAP code is a continuous process fuelled by new test simulations. Thus, EDF regularly enhances the MAAP qualification base, in particular through its participation in the "International Standard Problems" (ISP) and European projects (COLOSS, ENTHALPY, SARNET).

In addition to checking code performance, validation tests are used to:

- provide the user with recommended variation ranges or default values for various model parameters or to fix a particular set of correlations,
- identify ways of improving the various code models.

In addition to the many tests performed by FAI, further tests have been carried out by EDF since it acquired the code. These have focused particularly on core degradation, hydrogen production, containment thermal-hydraulics and iodine chemistry. In all, MAAP4.04 code qualification chart covers Table 8.1-1.

Validation studies have been carried out on reactor coolant system thermal-hydraulics. Results have been compared with data from experimental tests or with the results of computer codes more specifically dedicated to RCS thermal-hydraulics, such as RELAP or RETRAN. This comparison was based on multiple sequences (RCS breaks, loss of power, SGTR, etc.).

A series of experimental tests (Westinghouse loop) on natural convection modes was used to validate models adopted to represent these phenomena.

The models used to simulate containment thermal-hydraulics were validated on the basis of "bench mode" calculations on the isolated containment building. These calculations simulated HDR and VANAM-M3 experimental tests (concerning the hydrogen risk and aerosol behaviour respectively) as well as the NUPEC M-4-3 and M-7-1 tests for the mixture and distribution of hydrogen in the containment.

The CORA and QUENCH tests provide information on fuel assembly degradation, corium relocation and hydrogen production, with a particular focus on the reflooding situation in the case of the QUENCH tests. The MAAP qualification calculations of these models are performed in "bench" mode, with MAAP representation confined to the reactor core.

It is important to remember that qualification work does not only consist in applying the code to analytical separate-effect or semi-integral tests. One of its key aims is to validate the code on available integral tests (PHEBUS, LOFT) and the TMI-2 accident. This ensures that code performance is properly verified and that all the predominant phenomena are taken into account.

8.2.3 CODE UPGRADE PROSPECTS

MAAP5 will introduce the main future improvements expected. Managed by the FAI, the historical developer of the code, MAAP5, which will be rolled out at EDF around 2008, represents a major step forward. The following list includes only some of these improvements, since the final technical content of the first version of MAAP5, is not yet known.

- RCS thermal-hydraulics: processing in this area will be far more detailed than at present (nodal approach in single- and two-phase modes, decorrelated management of the reactor coolant loops including, in particular, realistic positioning of the safety injection on each loop, vortex in RHR system, improved modelling of natural circulation, particularly between downcomer, core and upper plenum, improved modelling of the formation of loop seals and their impact on natural circulation flows),
- core degradation model: improved numerical methods, 1D neutron and point-kinetics models, specification of coolant activity,
- model of critical dryout flux on the outer surface of the reactor vessel, when the vessel is externally flooded, combined with a calculation of the heat flux through the wall. The model may integrate insulation and a vessel-insulation gap,
- modelling of corium catchers, such as the one used in the EPR,
- integration of a new containment model, including several options for calculating heat transfers and improved modelling of energy sinks.

These improvements are now financed by some ten organisations that already hold a MAAP4 licence. Such commitments offer a clear view of the long-term future of this code for use in severe accident studies by EDF.

8.3 MELCOR

8.3.1 CURRENT CAPABILITIES OF THE CODE

MELCOR is a fully integrated code developed by Sandia National Laboratories (USA) since 1982. It is used to analyse all aspects of accidents liable to occur in a light water reactor (PWR, BWR). It handles coherently a broad range of phenomena involved in severe accidents, such as core degradation, fission product release and transport, ex-vessel corium behaviour and FP and aerosol behaviour in the containment building. Initially specialised in PSA work, applications now cover a wider scope: managing accidents and mitigating their effects, operator training and preparing and analysing experimental programs to improve knowledge.

The code has a modular structure. The thermal-hydraulics module is one of its linchpins, as it is closely linked to all the other modules: the "control volume" approach is used to model the entire reactor, with two regions in each volume, a sump (single liquid phase or boiling sump) and an atmosphere that may contain a water fog. The zones are connected by junctions where the fluid momentum equation is processed in a simplified 1D mode.

Originally, most models were relatively simple and parametric, derived from empirical correlations, with coarse spatial discretisation for quick calculation. Recently, however, the trend has been to favour mechanistic models and fine discretisation. For this reason, most of today's MELCOR models are mechanistic. The use of parametric models is limited to areas where phenomenological uncertainty remains high. Models that are more physical must be used for accident management. This means reducing uncertainties and a more detailed representation of physical phenomena. These objectives are now accessible, thanks to greater computing power and a better understanding of phenomena.

The latest version, released in October 2005, is MELCIOR 1.8.6, which features the following main improvements:

- the formation of molten corium pools in the core is modelled now, which was not the case in the previous version. The models used describe the progress of these pools in the RPV lower head: crust formation, convection, stratification into metal and oxide layers due to the density effect (no representation of inversion effects revealed in MASCA) and partitioning of fission products between these two layers (using simplified models or functions input by the user),
- the descriptions of in-vessel structures – shells, plates and lower head – are extensive and integrated within the same module to allow consistent processing of all in-vessel phenomena. The core shroud structure and in-vessel bypass can now be modelled. This will make possible to obtain realistic assessments of RPV lower head failure, addressing problems relating to the focussing effect and in-vessel corium retention by ex-vessel cooling,
- the intact fuel rod reflooding model has been improved (the quench front and water level are now monitored). The existing oxidation model, based on the Urbanic-Heidrick correlation, has proven satisfactory for most accident situations but, like other codes, cannot calculate the "runaway"

effect observed in certain QUENCH tests when the relocating corium is exposed to a high water flow rate,

- silver (Ag-In-Cd) release and B_4C oxidation observed during control rod degradation are now modelled. Modelling B_4C oxidation in the gas phase should, theoretically, determine the conditions governing organic iodine formation in the system. To achieve this, however, the use of gas chemistry models would have to be extended to cover containment phenomena,
- flashing in superheated flows of water entering the containment at low pressure has now been modelled.

The conclusions and development plans regarding the iodine chemistry model, based on the INSPECT model, remain unclear.

A number of improvements have recently been made to the interactive user interface including, in particular, the MELSIM simulator for managing data sets, calculations and results display. A tool has also been released for automatically initiating a series of sensitivity studies for uncertainty analysis. Many models feature optional user-adjustable parameters to simplify these studies.

MELCOR has been used to simulate accident sequences in many types of reactor including PWRs, BWRs, VVERs and RBMKs.

8.3.2 VALIDATION STATUS

MELCOR qualification work is based mostly on internationally available tests (including ISP) mentioned in Table 8.1-1 . In addition to the TMI-2 accident, the following can be mentioned (see Figure 8.3-1, Figure 8.3-2 and Figure 8.3-3):

- for the early in-vessel phase: VERCORS and VERCORS-RT (CEA) for FP release; PHEBUS-FP (IRSN) for silver release, B_4C behaviour and FP release in the system; QUENCH (FZK) for reflooding,
- for the late in-vessel phase: RASPLAV and MASCA (RRC-KI, Russia) for molten pools, OECD LHF-OLHF (SNL, USA) for reactor vessel mechanical strength,
- for systems and the containment: ACE and MACE (ANL, USA) for water injection under MCCI conditions, ARTIST (PSI, Switzerland) for FP retention in steam generators.

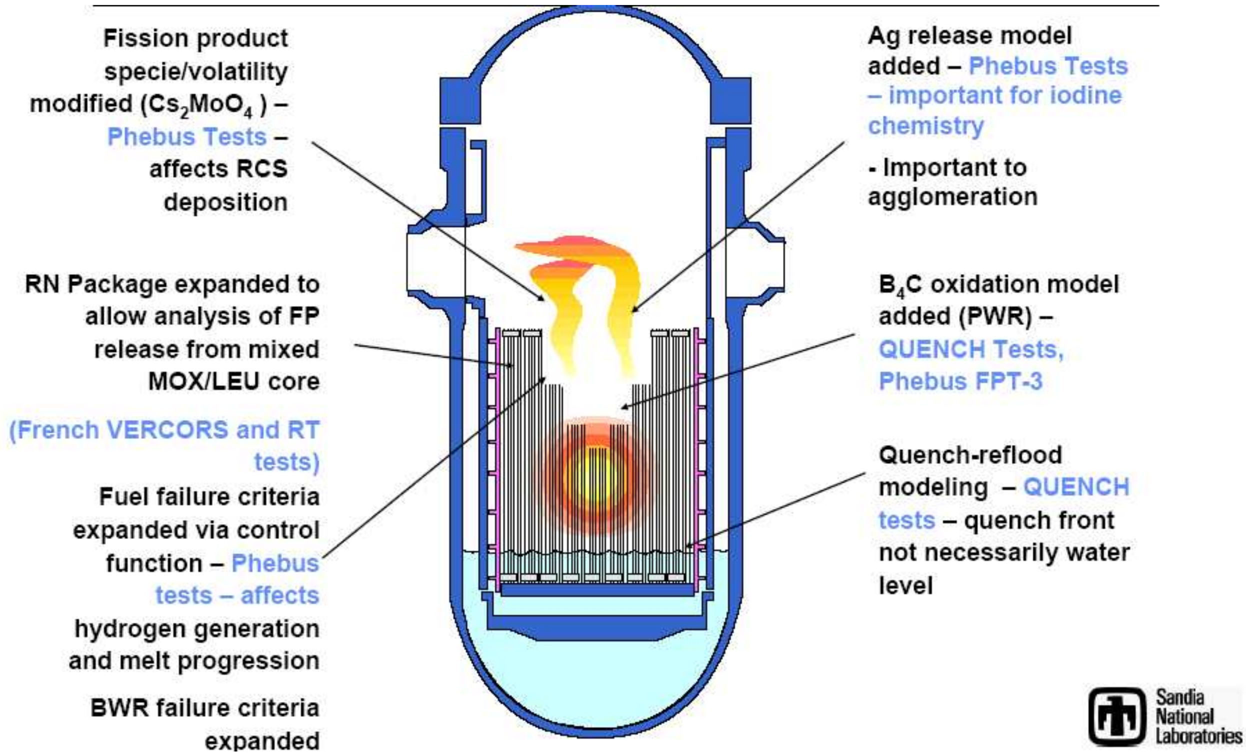


Figure 8.3-1: MELCOR development and qualification work in the early in-vessel phase

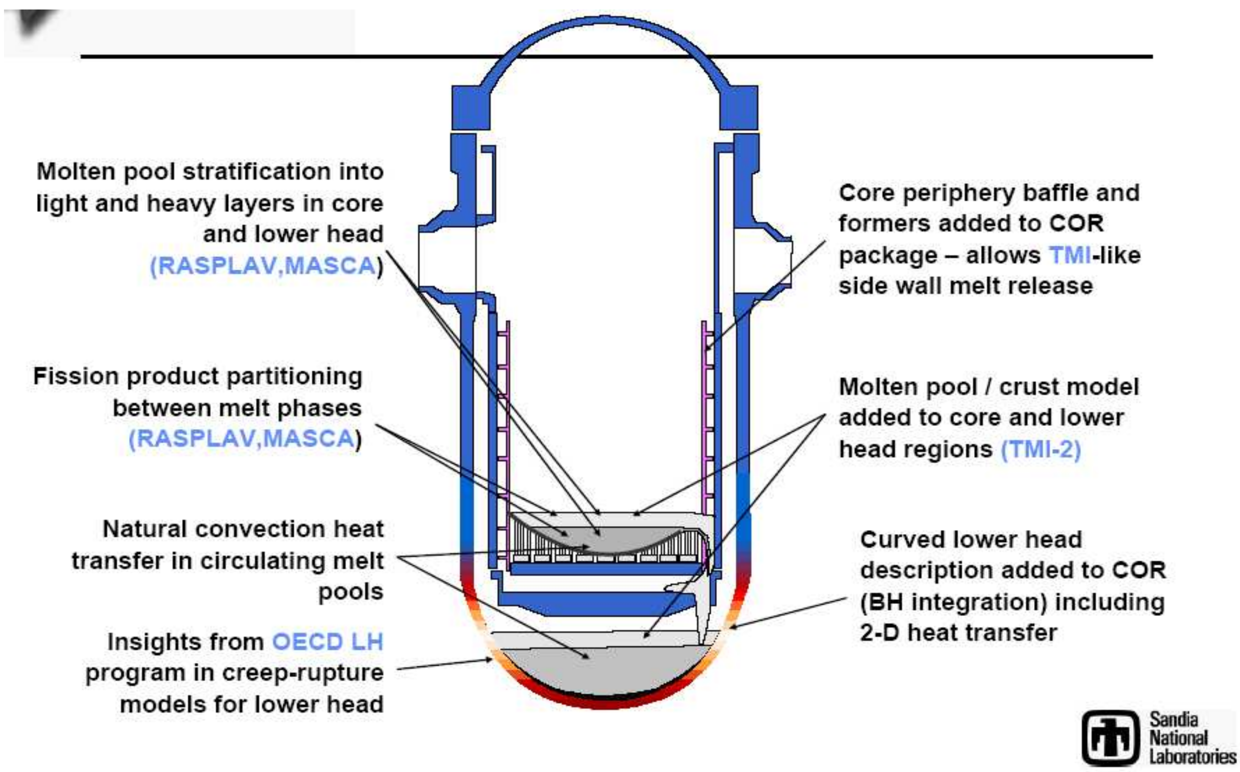


Figure 8.3-2: MELCOR development and qualification work in the advanced in-vessel phase (1)

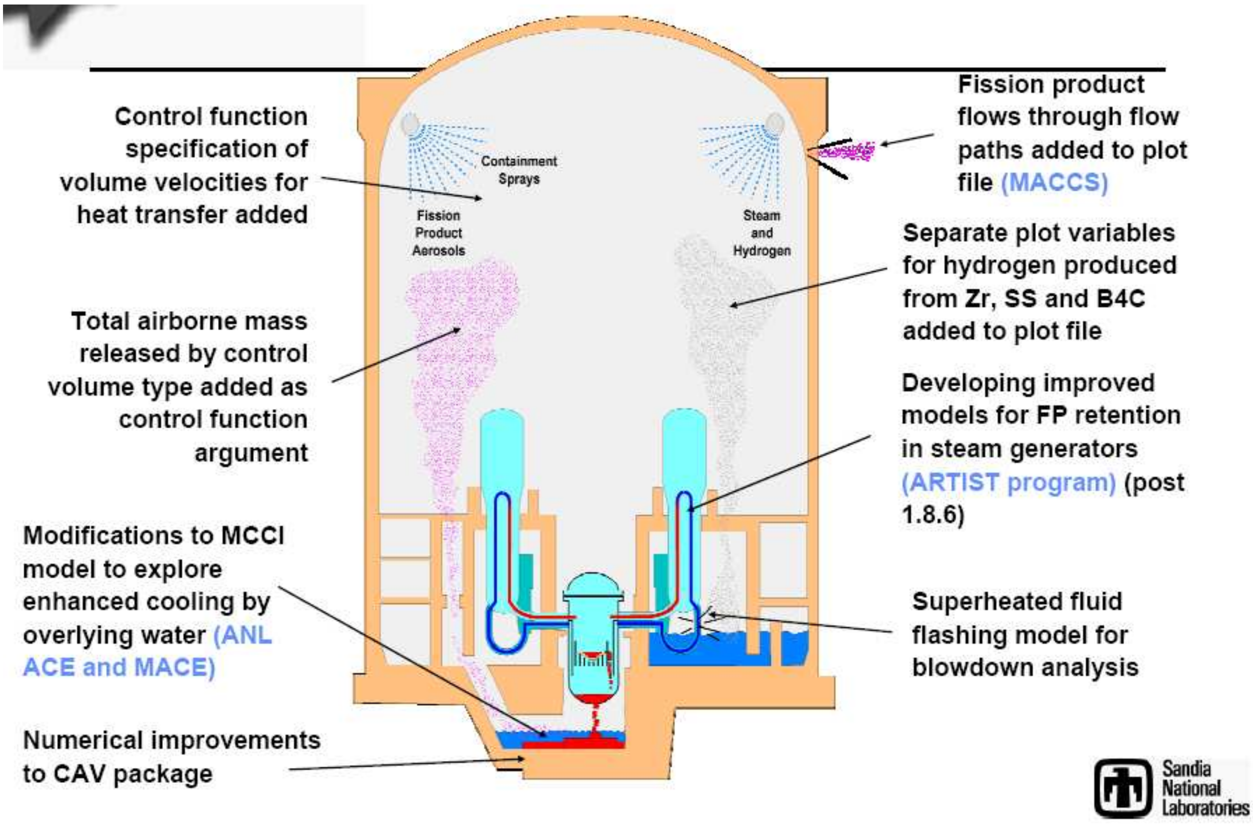


Figure 8.3-3: MELCOR development and qualification work in the advanced in-vessel phase (2)

According to the presentations at the CSARP in September 2005 [8.3_1], [8.3_2] and [8.3_3], and excluding TMI-2 applications, the latest qualification work concerned the following tests: LHF (using the 1D model, reasonable assessment of vessel failure time and maximum displacement with a peaked heat flux distribution, but underestimation with a uniform distribution), RASPLAV, QUENCH-07 (B_4C rod) and QUENCH-06 (ISP46), together with PANDA on BWR containment behaviour.

As for the previous codes, this validation work is completed by a consistency check on the results of reactor accident simulation and by participation in benchmark work.

8.3.3 CODE UPGRADE PROSPECTS

Current work concerns:

- advanced BWR or CANDU reactors (including adaptation of core degradation models with specific geometry),
- high-burnup fuel,
- HTRs, including the development of graphite oxidation and FP release models,
- generation IV reactors, including a generalisation of processed coolants (sodium, molten salt, etc.).

Furthermore, the project to modernise the code (led by IBRAE, Russia) by switching to the Fortran 95 programming language and reworking the data sets, will be completed in 2006.

TABLES IN CHAPTER 8

FIGURES IN CHAPTER 8

REFERENCES IN CHAPTER 8

[8.1_1] "Development and assessment of ASTEC code for severe accident simulation", NURETH-11 International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Avignon, 2-6 October 2005, J.P. Van Dorsselaere, S.Pignet, C.Seropian, T.Montanelli, P.Giordano, F.Jacq (IRSN), B. Schwinges (GRS).

[8.1_2] "ASTEC and SARNET – Integrating severe accident research in Europe", ICAPP'05, Séoul (Corée), 15-19 May 2005, J.P. Van Dorsselaere, J.C. Micaelli (IRSN), H.J. Allelein (GRS).

[8.3_1] "Overview of MELCOR development, assessment and applications activities", CSARP-05, Cooperative Severe Accident Research Meeting, Albuquerque (USA), 22-23 September 2005, R.Gauntt (SNL).

[8.3_2] "MELCOR 1.8.6 verification", CSARP-05, Albuquerque (USA), 22-23 September 2005, V.Nosatov (IBRAE).

[8.3_3] "Post-test analysis of ISP-42 (PANDA tests) using the MELCOR code", CSARP-05, Albuquerque (USA), 22-23 September 2005, J.Tills (SNL).

CHAPTER 9 : CONCLUSION

This document presents an overview of the current status of research related to studies of severe accidents. These studies were conducted in France and other countries by various players in the nuclear world, including industrial companies, research centres and safety organisations.

Looking at possible accident **scenarios** involving reactor core meltdown in PWRs, it first describes the main accident families in Level 1 PSA liable to induce core meltdown. The various physical phenomena which might occur in the reactor pressure vessel and in the containment building in the event of a severe accident are then presented in general terms.

The rest of the document presents the current understanding of these physical phenomena, how they follow on from each other and how their impact can be mitigated. A brief description of the main experiments – recently performed, underway or planned for the future – concerning these topics is then given, together with the main models and specific codes used to simulate the phenomena. Lastly, short-term prospects are analysed, with particular regard to experimental programmes and the development of models and simulation tools proposed by the severe accident community in the nuclear field on each of the themes discussed.

The overall results of this assessment work are presented below. It should be noted that the document raises the issue of a number of R&D needs, both in the body of the report and in this general conclusion. In this respect, it is important to point out that while the document helps to identify and prioritise these requirements, the suggested priorities should be confirmed in light of the safety analysis needs associated with the various risks and physical phenomena, drawing in particular on Level 2 PSA work. It should also be mentioned that, apart from the few exceptions given, a relative consensus has been reached on most of the needs mentioned here, at least in France (IRSN, CEA, EDF), and that if the CEA and IRSN have chosen to discuss them in this joint document, it definitely does not mean that these two organisations are committed to meet them.

In-vessel accident scenario

Although the initial stages of reactor core degradation (fuel rod oxidation and cladding failure) are now relatively well understood and modelled, many uncertainties remain regarding the phenomena characterising the later stages of degradation and the behaviour of corium in the lower head of the reactor pressure vessel (RPV). Since no new experimental programmes on core degradation are scheduled, remaining areas of uncertainty can only be reduced through further analysis of past tests and more modelling work. The current trend is to consider that some uncertainty should be tolerated here and that further progress should be based on sensitivity studies. On the other hand, some experimental programmes relating to the behaviour of corium in the RPV lower head are planned, focusing in particular on the interactions between a molten pool and the steel from which the RPV is made (INVECOR). If the CORTRAN programme is carried out, it should provide insight on the transition from a debris bed to a molten pool configuration. Depending on the knowledge gained from this work, sufficiently accurate models should become available, although this does not rule out the possibility of other requirements being identified for further experimental studies.

The moment at which lower head vessel failure occurs and the zone in which the break is initiated can be adequately determined for a given thermal load using the mechanical models developed as part of the LHF and OHLF programmes. However, there is no model available yet for predicting the crack propagation at high temperature and thus the final size of the breach. An experimental programme (IRSN/CEA/INSA in Lyon) to determine the impact of steel properties on the crack propagation in the lower head is currently nearing completion and should lead to the development of such a model.

Lastly, the IRSN has completed its R&D programme on the risk of a high-pressure core meltdown induced break in the reactor coolant system or in the steam generator tubes. The results of this work are providing input for Level 2 PSA studies. The assessment of uncertainties carried out as part of this work shows that it is still difficult to reliably predict the location of an induced break. Although the many improvements made to the ICARE/CATHARE system code with the introduction of a 2D axisymmetric model of the pressure vessel should bring significant progress in this area, considerable uncertainty will remain concerning the gas mixtures in the steam generator inlet plenum. No improvement is possible in these areas without highly complex CFD studies. Note that some of the experiments carried out as part of the OECD ROSA project, in which EDF, AREVA, the CEA and the IRSN are taking part, will be used to validate computer codes.

Phenomena liable to induce early containment failure

Experiments investigating the risk of direct containment heating (DCH) have shown that the occurrence of DCH at high reactor coolant pressure would have significant consequences in reactors where the reactor pit and the containment are directly connected.

At present, DCH is simulated in computer codes using simple models which do not sufficiently reflect the complexity and diversity of the phenomena involved. The use of multiphase simulation codes seems more promising. However a satisfactory prediction of the consequences of DCH at reactor scale would require filling a certain number of gaps in current knowledge. It is particularly difficult to extrapolate results from oxidation tests to the reactor case, especially for zirconium. Similarly, it would seem ill-advised at the moment to draw conclusions as to the contribution of hydrogen combustion to DCH-induced loads.

For these reasons, the IRSN considers that the R&D effort should be continued in the short or medium term. In particular, tests dedicated to the study of hydrogen behaviour during the DCH phase are planned. At the same time, mechanistic codes (MC3D and AFDM) are used not only in an attempt to make up for the lack of measurements and to extrapolate results to the case of the reactor, but also to contribute to the development of simplified models that could be implemented in scenario codes.

Dedicated research work has confirmed the decision to install devices to mitigate the hydrogen risk in all nuclear power reactor units in France. According to some studies, however, a risk of flame acceleration can persist.

The R&D work already carried out regarding the hydrogen risk has significantly improved our understanding of the phenomena governing the distribution of gas mixtures and the related risks of combustion. In particular, criteria based on experimental data are used to identify situations liable to induce pressure loads that can damage the containment. Although computing tools are now nearing maturity, their predictive performance needs to be enhanced by improving models and, in the case of multidimensional codes, by overcoming the computational limits that currently prevent us from using sufficiently refined meshes and obtaining mesh-independent results. Further R&D work should help to refine the analyses carried out for existing Nuclear Power Plants and assess the measures proposed for future reactors. Recent comparative exercises on thermal-hydraulics in the containment have shown that gas stratification phenomena and depressurisation when the spraying system is activated are not accurately represented by existing codes. Transient flow conditions in stratified situations would be analysed in the frame of the on-going programmes: TOSQAN, PANDA and MISTRA. Models used in combustion studies still need to be improved and validated to simulate flame propagation in heterogeneous media. The impact of water drops on flame acceleration should also be studied. These topics will be treated in the ENACCEF programme. The impact of recombiner location on the recombination rate and the risk of recombiner-initiated ignition are investigated within SARNET. The first topic is investigated by a numerical work while the second is based on the REKO tests.

Lastly, as regards steam explosion, significant effort has been devoted in recent years to developing sophisticated simulation tools. These software programmes are now operational but their current level of accuracy is not considered sufficient. These deficiencies are mainly related to a poor understanding of local phenomena. Phase 1 of the SERENA international programme, which has just been completed, highlighted the divergences of the various specialists as to the adopted modelling assumptions, especially for the premixing phase. These differences have led to considerable disparity in reactor calculation results.

As for the safety impact of in-vessel steam explosion, the international community seems to have reached a consensus that this phenomenon should not lead to reactor pressure vessel failure. There is, however, growing international interest in a more precise evaluation of the ex-vessel situation, as defined in the SERENA-2 proposal.

In conclusion, given the volume of knowledge already acquired, the research and development programmes currently being prepared should lead to a significant step forward in the understanding and prediction of phenomena related to the risk of early containment failure.

Phenomena liable to induce late containment failure

The results of recent experiments, together with reactor calculations performed using the MEDICIS and TOLBIAC-ICB codes, have shown that great uncertainty remains concerning basemat erosion kinetics during corium-concrete interaction. In view of the impact of such uncertainties on radioactive release, major R&D programmes on this subject have already been launched. The ARTEMIS 2D and VULCANO ICB experimental programmes, conducted under a tripartite agreement between the CEA, the IRSN and EDF, and the continuation of the OECD-MCCI programme under the auspices of the OECD/CSNI, will make it possible to understand more fully the physical phenomena governing the structure of the interfaces and to supplement knowledge on the 2D heat flux distribution in a homogeneous pool. The VULCANO-ICB and ARTEMIS oxide-metal tests in a stratified pool configuration should also provide results for heat transfers between oxide and metal layers and for the evolution of the pool

configurations. Furthermore, an additional study programme is currently being defined to reduce uncertainty concerning the thermochemical properties of corium/concrete mixtures.
More recently, an IRSN, CEA and EDF working group agreed on the interest of conducting additional experimental programmes if the results of the above tests proved insufficient. So far, two test programmes with simulant materials have been considered: The ABI programme will focus on heat transfers between the oxide and metal layers and the CLARA programme will study 2D heat transfer between a heated pool and a porous wall in presence of gas sparging.

In conclusion, given the considerable impact of basemat erosion kinetics on radioactive release and in view of remaining uncertainties in this area, this theme is considered a priority in terms of research and development.

In- and ex-vessel corium retention and cooling

With regard to in-vessel corium retention by in-vessel reflooding, existing software provides a relatively good evaluation of situations where the core suffers limited damage but no loss of geometry. Existing experimental data, the QUENCH tests in particular, should further improve the qualification of this software. Considerable uncertainty remains, however, as to the effectiveness of cooling a degraded core by in-vessel reflooding for debris bed or liquid corium pool situations. As for the "debris bed" configuration, the ISTC QUENCH programme should provide information on debris size. In addition, the IRSN has begun preparations for defining an analytical test programme aimed at confirming the multidimensional effects of reflooding a debris bed (increasing its "coolability").
On the subject of in-vessel retention by external reflooding of the reactor pit, considerable R&D work has been conducted to assess the effectiveness of ex-vessel cooling. Studies carried out on this subject – by the CEA, EDF or the IRSN – showed that it is difficult to demonstrate the systematic preservation of vessel integrity using this means. On the other hand, in-depth studies could be made on the effectiveness of external cooling in particular situations (e.g. low residual power, limited core damage, etc.). Notwithstanding this, further work in this area will consist of engineering studies or analyses based on existing knowledge and no new R&D programmes are planned.

Tests carried out on corium cooling by water injection above the corium during its interaction with concrete have been unable to quantify the effectiveness of this method, although it is recognized that it leads a reduction in the basemat ablation rate. New tests will be performed as part of the MCCI-2 programme. Given the significance of this problem in terms of severe accident management, however, and depending on the results obtained both from these tests and other MCCI tests, further research work could well prove worthwhile.

The R&D programme devoted to corium spreading, in particular within the EPR project context, has settled the question of spreading under dry conditions. Models have been developed and validated and it has been demonstrated that the corium-concrete mixture spreads well if the flow rate is high enough. The inclusion of a thin layer of water (to simulate condensation water) or a concrete substrate (releasing steam and CO_2) has little impact on spreading. Further studies are required, however, to obtain validated models of flowing mechanisms with a significant thickness of overlying water.

In recent years, a considerable amount of R&D work has been devoted to the EPR's ex-vessel core catcher. This work, carried out by the designer as well as research and safety organisations within the framework of national and international programmes, provides a good basis for the validation and assessment of technical solutions adopted for design.

In conclusion, our current understanding of in- and ex-vessel corium retention and cooling does not allow us to determine the effectiveness of cooling by reflooding a damaged core, or ex-vessel cooling, or cooling the corium by water injection above the corium during MCCI. Additional R&D work is required to settle this question.

Fission product release and transport

Most studies of fission product release from the fuel in the vessel have considered medium-burnup UO_2 fuels. The relatively broad spectrum of experimental data from analytical tests, combined with PHEBUS-FP integral tests, has been used to develop and validate mechanistic and simplified models that correctly simulate the dependence of release mechanisms on various parameters, including temperature, redox conditions, interactions with structural materials, burnup, the type of fuel (UO_2 or MOX) and its state (solid or liquefied).
Some areas of uncertainty remain here, but R&D programmes have already been launched to clarify them. Micro-analyses of fuels from the VERCORS tests are planned as part of the international Source Term programme to check the assumptions proposed during the interpretation of these tests and used as a basis for building models. The programme also includes tests to be performed in the CEA's VERDON facility to extend the experimental database to incorporate MOX and high-burnup UO_2 fuel and air-ingress scenarios.

Fission product release in an ex-vessel situation might occur in the event of sump boiling or during the corium-concrete interaction. Studies concerning these phenomena pointed out that the release rates in ex-vessel situations are lower than those found for in-vessel situations. For this reason, there would appear to be no need for further R&D programmes concerning these two aspects.

The physical phenomena governing the transport of aerosols in the reactor cooling system and their behaviour in the containment are generally well known and models are available to describe them. The most significant deposition phenomena, such as thermophoresis, diffusiophoresis and gravitational settling, have been the focus of dedicated experiments in the past for model validation purposes.
On the subject of aerosol transport in the reactor cooling primary and secondary systems, significant uncertainties remain concerning the phenomena involved in aerosol retention on the secondary side of the steam generator. The international programme, ARTIST, is currently investigating this problem. Work is also underway as part of SARNET to improve the models used to describe the phenomena governing the mechanical resuspension of deposits in response to high flow rates. This work makes use of existing data and will also benefit from new data acquired from the ARTIST programme.
The most recent R&D work on aerosol behaviour in the containment has concentrated on how the spray system can push aerosols down. The recently started TOSQAN AEROSOLS programme should teach us more about aerosol collection mechanisms using a water spray system in accident situations.

Lastly, a number of points remain to be clarified regarding fission product chemistry.

The PHEBUS-FP integral tests have challenged a number of paradigms concerning fission product chemistry in the reactor cooling system, caesium and iodine in particular. New models have been developed to take into account results obtained for caesium. Iodine, however, is a more complex matter. Chemical speciation calculations have been carried out but have so far proven incapable of reproducing the observed effects correctly. It would appear that the differences observed are due to currently ignored chemical kinetic phenomena. The CHIP experimental programme, part of the international Source Term programme, should identify the chemical reactions responsible and assess the kinetic data required to model them. Validated physical models describing the behaviour of iodine in the reactor cooling circuit are expected by 2010. Recent experiments carried out in Hungary and Finland have shown that ruthenium should also be reconsidered. The results of these tests should be analysed to obtain validated models by 2008.

In the containment, the two fission products with chemical properties having a potentially significant impact on radioactive release are iodine and ruthenium, as they can both exist in the gaseous state under the conditions encountered in the containment.

The importance of iodine has long been a recognised fact and many studies, both experimental and theoretical, have been carried out to develop and qualify models on the subject. In particular, the impact of spray on gaseous iodine has been quantified and validated models are now available. Nonetheless, some areas of uncertainty with a significant impact on the levels of radioactive release remain, particularly concerning the formation of organic iodides in the gaseous phase. EPICUR, an experimental programme that is part of the international Source Term programme, is expected to provide new data in this area as well as on certain aspects relating to liquid- and gas-phase radiolysis. This data, together with that of the recently completed PARIS programme, should lead to improved, validated models by 2010. In addition, the RECI tests have recently brought to light interactions between iodine aerosols and recombiners that can result in the production of gaseous iodine. The IRSN plans to continue this programme with a view to measuring the impact of these interactions on the concentration of gaseous iodine in the containment.

The "ruthenium in containment" programme currently in progress and carried out as part of the Source Term programme, has shown that ruthenium was liable to remain in the containment in the gaseous state. The next step is to quantify the kinetics of the various phenomena involved in order to assess the concentration of gaseous ruthenium in the containment during an accident.

In conclusion, the experimental programmes scheduled to last until 2010 cover those aspects of fission product release and transport where further knowledge is required. It is likely, however, that some of these programmes will stretch beyond that deadline.

Severe accident codes

All the research carried out on the physical phenomena related to severe accidents has led to the development of integral codes designed to simulate this type of accident in a small computing time, from the initiating event to the possible release of radionuclides outside the containment and taking into account the main safety systems.

This document has described three of these codes: ASTEC, MAAP and MELCOR. They are used for reactor safety studies, particularly for assessing the source term and for Level 2 PSAs. These software packages are constantly upgraded to keep up with the state of the art. They are validated both against analytical tests and global experiments. Comparison exercises or benchmarks are also arranged, in particular as part of SARNET, to compare results not only between these three codes, but also with the results of mechanistic codes. This type of exercise serves to identify the models responsible for any discrepancies observed from one code to another, thereby helping to measure the degrees of uncertainty remaining in the assessment of certain physical phenomena.

Note that the ASTEC code plays a leading role in SARNET, as it gradually integrates in the form of models all the knowledge generated by the network, and because the partners carry out a considerable amount of work on validation and reactor applications. One of the goals of SARNET is to see ASTEC become the European reference code.

GLOSSARY

AEAT: Atomic Energy Authority Technology (GB)

AECL: Atomic Energy of Canada Limited (Canada)

Ag-In-Cd (silver, indium and cadmium alloy)

AICC pressure: Adiabatic and Isochoric Complete Combustion pressure

ANL: Argonne National Laboratories (USA)

ASTEC: Accident Source Term Evaluation Code

ATWS: Anticipated transients without scram

AVN: *Association Vincotte Nucléaire* (Belgium)

BARC: Bhabha Atomic Research Centre (India)

BNL: Brookhaven National Laboratory

CEA: French Atomic Energy Commission

CFD: Computational Fluid Dynamic

CHF: Critical Heat Flux

CHRS: Containment Heat Removal System

CSNI: Committee on the Safety of Nuclear Installations

CVCS: Chemical and Volume Control System

DCH: Direct Containment Heating

DDT: Deflagration-Detonation Transition

EDF: Electricité de France (France)

EFWS: Emergency Feedwater System

EPR: European Pressurised-water Reactor

EPRI: Electric Power Research Institute

FP: Fission Products

FWLB: Feedwater Line Break

FZK: *Forschungszentrum Karlsruhe GmbH* (Germany)

FZR: *Forschungszentrum Rossendorf e.V* (Germany)

GAEC: Guide to Emergency Response Team Action

GIAG: Severe Accident Action Guide

GRS: *Gesellschaft für Anlagen und Reaktorsicherheit mbH* (Germany)

IBRAE: Nuclear Safety Institute of the Russian Academy of Science

IET: Integral Effect Test

IKE: University of Stuttgart (Germany)

INEL: Idaho National Engineering Laboratories (USA)

INR: Institute of Nuclear Energy (Romania)

INSA: National Institute of Applied Science (France)

IRSN: Institute for Radiological Protection and Nuclear Safety (France)

ISP: International Standard Problem

ISTC: International Science and Technology Centre

IVR: In-Vessel Retention

JAERI: Japan Atomic Energy Research Institute (Japan)

JRC: Joint Research Centre (EEC)

KAERI: Korea Atomic Energy Research Institute (South Korea)

KTH: Royal Institute of Technology in Stockholm

LDV: Laser Doppler Velocimetry

LOCA: Loss of Coolant Accidents

LUCH (FSUE SRI SIA): Federal State Unitary Enterprise Scientific Research Institute Scientific Industrial Association (Russia)

MAAP: Modular Accident Analysis Program

MCCI: Molten Corium-Concrete Interaction

MELCOR: Methods for Estimation of Leakages and Consequences of Releases

MOX: Mixed uranium- and plutonium-oxide fuel

MSB: Main Steam Bypass

MSRT: Main Steam Relief Train (atmospheric steam dump)

NIIAR: Research Institute of Atomic Reactors (Russia)

NITI: Research Institute for Nuclear Technology (Russia)

NRC: Nuclear Regulatory Commission (United States)

NUPEC: Nuclear Power Engineering Corporation (Japan)

OECD: Organisation for Economic Cooperation and Development

ORNL: Oak Ridge Nat. Lab. (Canada)

PIV: Particle Image Velocimetry

Pr: Prandtl number

PSA: Probabilistic Safety Assessment

PSI: Paul Scherrer Institute (Switzerland)

PWR: Pressurised Water Reactor

Ra: Rayleigh number

Re: Reynolds number

RCS: Reactor Cooling System

RHR: Residual Heat Removal system

RRC-KI: Russian Research Centre – Kurchatov Institute (Russia)

SG: Steam Generator

SGTR: Steam Generator Tube Rupture

SLB: Steam Line Break

SARNET: Severe Accident Research NETwork of excellence

SNL: Sandia National Laboratories (USA)

TGTA: Secondary system transients

TMI 2: Three Mile Island 2

TRCP: Reactor coolant system transients

UCLA: University of California-Los Angeles

UJV: *Ustav Jaderneho Vyzkumu Rez a. s.* (Czech Republic)

VTT: Technical Research Centre of Finland