Nuclear Power Reactor Core Melt Accidents

Current State of Knowledge

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Cover illustration: Radiographic image of Phebus FP test devices and an artist's impression of the TMI-2 reactor core after fuel melt.

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Preface

This new publication on what are referred to as "severe" core melt accidents, which may occur in pressurised light-water reactors, is the result of one of the most comprehensive surveys ever conducted on this subject. The knowledge it contains is presented with a strong educational focus. I would like to take this opportunity to thank all those mentioned in the foreword who contributed to this vast project, with a special mention for its coordinator D. Jacquemain.

Although the project was not yet completed, considerable headway had already been made when the Fukushima Daiichi disaster struck. This was the world's third severe accident and resulted in the destruction of three nuclear power reactors and the release of large quantities of radioactive material to the sea and atmosphere. It raised the question as to whether the project should be postponed to take into account feedback from these major events. It was however decided to complete the book as soon as possible as it would be several years before any detailed scientific information from the Fukushima Daiichi accident became available. Furthermore, the knowledge and models already available within IRSN on the phenomenology of this type of accident had enabled the Institute to carry out valuable real-time assessments of changes in the state of the reactors.

For more than thirty years, IRSN has been carrying out experimental studies on the phenomena that lead to reactor core melt and those induced by this type of event. Back in the 1960s when the first nuclear power reactors were designed, a core melt was considered impossible because of the design measures taken to prevent it, such as design margins and redundant safety systems to halt the chain reaction and remove the heat generated in the reactor core. Consequently, no measures were included in reactor design to mitigate the impact of this type of event. This approach had to be rethought following the accident at the Three Mile Island nuclear power plant in the United States in 1979. It was then necessary to determine how fuel could be damaged in a reactor core and, more especially to understand the melting process induced by a loss of cooling that could ultimately lead to failure of the reactor coolant system – and the reactor vessel in particular. The next step was to grasp how chemical or radiolytic reactions could induce a significant release of hydrogen and many fission products exhibiting varying degrees of volatility and toxicity.

An experimental programme unlike any other in the world was then launched using Phebus, a reactor built by the CEA at Cadarache in the south of France. As part of the programme, fuel melt tests were performed on a reduced scale, representative of the actual operating conditions in a pressurised water reactor. New knowledge was to emerge from this impressive programme, including some surprises that called into question certain theoretical predictions. Models aimed at simulating these extreme phenomena in a full-scale reactor were then developed and incorporated in computer tools and validated during these tests.

As knowledge of severe accidents grew over the years, some countries took concrete steps to improve the safety of power reactors – whether existing or planned.

SARNET, an international network of experts and researchers led by IRSN from 2004 to 2013, coordinated continuous improvement of knowledge and the standards of models used to simulate severe accident phenomena in various types of reactor. This collaboration is being continued as part of the European NUGENIA association. Further experiments are needed, however, to reduce uncertainty on various phenomena with a significant impact on the consequences (especially for health) of a severe accident, although, based on data from the Phebus programme, such experiments are now designed as analytical tests, known as separate-effect tests. These are designed to target individual phenomena for which greater knowledge is required: what happens if an attempt is made to “reflood” a severely damaged, partially melted reactor core? What happens to the corium – the chemically and thermally aggressive mixture of fuel and molten metal – once it is released from the reactor core? Another question, of prime importance for radiation protection, concerns the behaviour of the different chemical species of radioactive iodine and ruthenium which are produced in large quantities inside the reactor containment, with varying degrees of volatility.

IRSN and its national and international research partners will continue to devote considerable resources in these areas over the coming years. For the past fifteen years, the Institute has never lost sight of the fact that severe accident research is vital. Unfortunately, the accident at Fukushima proved it right. The knowledge already acquired, as well as that yet to come, should be used not only to go on improving existing reactors wherever possible, but also to ensure that in the future, the nuclear industry at last develops reactors that no longer expose countries opting for nuclear energy to the risk of accidents, and the ensuing radioactive contamination of potentially large areas, that most human societies consider unacceptable. I hope that this publication helps to disseminate existing knowledge on this crucial topic as the new generation of nuclear engineers takes over from the old. I also hope it serves to illustrate how important it is to continue research and industrial innovation, without which no essential progress can be made in the field of nuclear safety.

Jacques Repussard
IRSN Director-General

List of abbreviations

Institutions

AEAT: Atomic Energy Authority Technology, UK (AEC Technology plc)

AECL: Atomic Energy of Canada Limited, a nuclear science and technology research institute

AEKI: Atomic Energy Research Institute, Budapest, Hungary

ANCCLI: *Association nationale des comités et commissions locales d'information* (French National Association of Local Information Commissions and Committees)

ANL: Argonne National Laboratory, USA

ANR: *Agence nationale de la recherche* (National Research Agency, France)

ASN: *Autorité de sûreté nucléaire* (Nuclear Safety Authority, France)

AVN: *Association Vinçotte nucléaire* (Vinçotte Nuclear Association, Belgium)

BARC: Bhabha Atomic Research Centre, India

BNL: Brookhaven National Laboratory, USA

CEA: *Commissariat à l'énergie atomique et aux énergies alternatives* (Alternative Energies and Atomic Energy Commission, France)

CLI: *Commission locale d'information* (French Local Information Commission)

CNL (formerly AECL): Canadian Nuclear Laboratoires

CNRS: *Centre national de la recherche scientifique* (French National Centre for Scientific Research)

CSNI: Committee on the Safety of Nuclear Installations, OECD

EDF: *Électricité de France* (French power utility)

EPRI: Electric Power Research Institute, USA

FAI: Fauske & Associates, Inc., USA

FzD: *Forschungszentrum Dresden-Rossendorf* (research laboratory in Dresden, Germany)

FzK: *Forschungszentrum Karlsruhe* (Karlsruhe Institute of Technology, Germany)

GRS: *Gesellschaft für Anlagen – und Reaktorsicherheit*, (reactor safety organisation in Germany)

IAEA: International Atomic Energy Agency, Vienna, Austria

IAE-NNC-RK: Institute of Atomic Energy – National Nuclear Centre – Republic of Kazakhstan

IBRAE: Nuclear Safety Institute of Russian Academy of Sciences

ICRP: International Commission on Radiological Protection

IKE: *Institut für Kernenergetik und Energiesysteme*, *Universität Stuttgart* (Institute for Nuclear Technology and Energy Systems, University of Stuttgart, Germany)

INEL: Idaho National Engineering Laboratories, Idaho, USA

INL: Idaho National Laboratory, USA

INSA: *Institut national des sciences appliquées* (National Institute of Applied Science, France)

IPSN: *Institut de protection et de sûreté nucléaire* (Institute for Nuclear Safety and Protection, France)

IREX: *Institut pour la recherche appliquée et l'expérimentation en génie civil* (Institute for Applied Research and Experimentation in Civil Engineering, France)

IRSN: *Institut de radioprotection et de sûreté nucléaire* (Institute for Radiological Protection and Nuclear Safety, France)

ISS: Innovative Systems Software, USA

ISTC: International Science and Technology Centre, EC

JAEA: Japan Atomic Energy Agency

JAERI: Japan Atomic Energy Research Institute

JNES: Japan Nuclear Energy Safety

JRC: Joint Research Centre, EC

JSI: Jozef Stefan Institute, Slovenia

KAERI: Korea Atomic Energy Research Institute, South Korea

KAIST: Korea Advanced Institute of Science and Technology, South Korea

KINS: Korea Institute of Nuclear Safety, South Korea

KIT (ex-FzK): *Karlsruher Institut für Technologie* (Karlsruhe Institute of Technology, Germany)

KTH, see RIT

LUCH: Scientific Manufacturer Centre, Russia

MIT: Massachusetts Institute of Technology, USA

NEA: Nuclear Energy Agency, OECD

NIIAR: Scientific Research Institute of Atomic Reactors, Russia

NITI: Aleksandrov Scientific Research Technological Institute, Saint Petersburg, Russia

NRC-KI (formerly RRC-KI): National Research Centre Kurchatov Institute, Moscow, Russia

NUPEC: Nuclear Power Engineering Corporation, Japan
OECD: Organisation for Economic Co-operation and Development
ORNL: Oak Ridge National Laboratory, USA
PSI: Paul Scherrer Institute, Switzerland
RIT (formerly KTH): Royal Institute of Technology, Stockholm, Sweden
SKI: Swedish Nuclear Power Inspectorate
SNL: Sandia National Laboratory, USA
UCLA: University of California, Los Angeles, USA
UCSB: University of California, Santa Barbara, USA
UJV: Nuclear Research Institute Rez, Czech Republic
US NRC: United States Nuclear Regulatory Commission, USA
VTT: Technical Research Centre, Finland

Technical abbreviations

Ag-In-Cd: Silver-Indium-Cadmium
AICC: Adiabatic Isochoric Complete Combustion
ARTIST: Aerosol Trapping in a Steam Generator (experimental programme carried out by the Paul Scherrer Institute [PSI])
ATWS: Anticipated Transient Without Scram (automatic reactor shutdown without insertion of control rods or transients with failure of the automatic reactor shutdown system – also known as ATWR for anticipated transient without (reactor) trip)
AVS: Annulus Ventilation System (1300 MWe, 1450 MWe reactors and EPR)
BIP: Behaviour of Iodine Project (international programme on iodine behaviour under the auspices of the OECD)
BL: Electrical Building
BWR: Boiling Water Reactor
CANDU: CANada Deuterium Uranium reactor (a heavy-water reactor)
CCWS: Component Cooling Water System
CFD: Computational Fluid Dynamics
CHF: Critical Heat Flux
CHRS: Containment Heat Removal System (a reactor spraying system in the EPR designed for use in severe accidents)
CODIR-PA: French Post-accident Management Steering Committee
CRP: Coordinated Research Programme on Severe Accident Analysis, IAEA
CSA: Complementary Safety Assessment
CSARP: Cooperative Severe Accident Research Programme (coordinated by the US NRC)
CSD: Severely Degraded Fuel
CSS: Containment Spraying System
CVCS: Chemical and Volume Control System
DAC: Facility construction licence

DCH: Direct Containment Heating (of gases)

DDT: Deflagration-Detonation Transition

E3B: Extension of the third containment barrier

EEE: Containment annulus (1300 MWe, 1450 MWe reactors and EPR)

EFWS: Emergency Feedwater System

ENACEEF: Flame acceleration facility, an experimental installation of the CNRS/ICARE in Orleans, France

EPR: European Pressurised Water Reactor

EPS: Emergency Power Supply

ESWS: Essential Service Water System

ETY: Hydrogen Reduction and Measurement System

FB: Fuel Building

FNR: Fast Neutron Reactor

FP: Fission Products

FP + number: European Commission Framework Programme for research and technological development (e.g., FP6, FP7 for the sixth and seventh framework programmes)

FPCPS: Fuel Pool Cooling and Purification System

FWLB: Feedwater Line Break

GAEC: Assistance Guide for Emergency Response Teams

GCR: Gas-Cooled, Graphite-Moderated Reactor

GIAG: Severe Accident Operating Guidelines

HHSI: High Head Safety Injection

HRA: Human Reliability Analysis

HTR: High Temperature Reactor

IRWST: In-containment Refuelling Water Storage Tank (borated water tank located inside the EPR containment building)

ISP: International Standard Problem

ISTP: International Source Term Programme

LHF: Lower Head Failure (failure in the lower part of the reactor vessel)

LHSI: Low Head Safety Injection (or Low Head Safety Injection System according to context)

LOCA: Loss-of-Coolant Accident

LUHS (H1): Loss of Ultimate Heat Sink (H1 in France)

MCCI: Molten Core-Concrete Interaction

MFWS: Main Feedwater System

MHPE: Maximum Historically Probable Earthquake

MHSI: Medium Head Safety Injection

MOX: Mixed Oxide Fuel (fuel composed of a mixture of $UO2 + PuO2$)

MPL: Maximum Permissible Level (of radioactivity)

NAB: Nuclear Auxiliary Buildings

OLHF: OECD Lower Head Failure (OECD research programme on failure in the lower part of the reactor vessel)

ORSEC: French emergency response plan

PBMR: Pebble Bed Modular Reactor (a type of high-temperature reactor or HTR)

PDS: Plant Damage State

PHWR: Pressurised Heavy Water Reactor

PPI: Off-site Emergency Plan

PRT: Pressuriser Relief Tank

PSA: Probabilistic Safety Assessment

PUI: On-site Emergency Plan

PWR: Pressurised Water Reactor

RB: Reactor Building

RBMK: *Reactor Bolshoy Moshchnosty Kanalny* (high-power Russian reactor with pressure tubes)

RCS (transients): Transients on the Reactor Coolant System

RCS: Reactor Coolant System

RFS: Basic Safety Rule

RHRS: Residual Heat Removal System

SAB: Safeguard Auxiliary Buildings

SARNET: Severe Accident Research NETwork of excellence, a European research project to study core melt accidents in water reactors

SBO (H3): Station Blackout

SERENA: Steam Explosion REsolution for Nuclear Applications (an OECD research programme)

SG: Steam Generator

SGTR: Steam Generator Tube Rupture

SI: Safety Injection

SIS: Safety Injection System

SLB: Steam Line Break

SME: Seismic Margin Earthquake

SOAR: State-of-the-Art Report

TAM: Equipment hatch

TGT: Thermal Gradient Tube

TGTA-H2: Accident with total loss of steam generator feedwater supply and failure of “feed and bleed” operating mode (or transients on the secondary system)

TMI: Three Mile Island, USA

TMI-2: Reactor 2 at the Three Mile Island NPP, USA

VCI: Pre-service Inspection

VD: Ten-Yearly Outage Programme

V-LOCA: Loss of Coolant Accident (containment bypass accidents or loss-of-coolant accidents outside the containment building)

VVER: *Vodo-Vodyanoi Energetichesky Reaktor* (Russian water-cooled, water-moderated nuclear power reactor)

ZPP: Population Protection Zone

ZST: Reinforced Environmental Monitoring Zone

Foreword

This summary of knowledge on core melt accidents is a collective work written for the most part by authors from the *Institut de Radioprotection et de Sûreté Nucléaire* (French Institute for Radiological Protection and Nuclear Safety or IRSN). Some sections include contributions from authors from the *Commissariat à l'énergie atomique et aux énergies alternatives* (French Alternative Energies and Atomic Energy Commission or CEA). Experts from both these organisations and from EDF, a French power utility, also took part in carefully proofreading various chapters. We would like to express our thanks to all those who contributed in one way or another to this publication.

Didier Jacquemain from IRSN, who was the project coordinator.

Contributions were made by the following authors:

- Chapters 1, 2, 3 and 9: Didier Jacquemain;
- Chapter 4: Gérard Cénérino, François Corenwinder, Didier Jacquemain and Emmanuel Raimond from IRSN;
- Chapter 5: Ahmed Bentaïb from IRSN (Section 5.2.2), Hervé Bonneville from IRSN (Section 5.1.4), Bernard Clément from IRSN (Section 5.5), Michel Cranga from IRSN (Sections 5.3, 5.4.2 and 5.4.3), Gérard Ducros from CEA (Section 5.5), Florian Fichot from IRSN (Sections 5.1.1, 5.1.2 and 5.4.1), Christophe Journeau from CEA (Section 5.4.3), Vincent Koundy from IRSN (Section 5.1.3), Daniel Magallon from CEA (Section 5.2.3), Renaud Meignen from IRSN (Sections 5.2.1 and 5.2.3), Jean-Marie Seiler from CEA (Section 5.4.1) and Bruno Tourniaire from CEA (Sections 5.3 and 5.4.2);
- Chapter 6: François Corenwinder, Denis Leteinturier, Frédérique Monroig, Georges Nahas and Frédérique Pichereau from IRSN;

– Chapter 7: Bernard Clément (Section 7.3) and Didier Jacquemain (Sections 7.1 and 7.2);

– Chapter 8: Jean-Pierre Van-Dorsselaere from IRSN.

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– for IRSN: Jean Couturier, Cécile Debaudringhien, Anna Duprat, Patricia Dupuy, Jean-Michel Evrard and Grégory Nicaise;

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Randall O. Gauntt from the US Nuclear Regulatory Commission (USA) and Jonathan Birchley from the Paul Scherrer Institute (Switzerland) approved the section on the MELCOR computer code in Chapter 8.

Denis Boulaud, Bernard Chaumont, Bernard Clément, Richard Gonzalez, Didier Jacquemain, Daniel Quéniart, Jean Peltier, Frédérique Pichereau from IRSN and Michel Durin from CEA proofread the whole work several times to ensure consistency.

Georges Goué, Odile Lefèvre and Sandrine Marano from IRSN prepared the work for publication.

Lastly, this work would not have been possible without the active participation of IRSN's Scientific Directors Jean-Dominique Gobin and Michel Schwarz, and its Director-General Jacques Repussard.

Tim Haste from IRSN largely contributed to improve the quality of the present English version by proof-reading minutely many chapters.

Summary

Chapter 3
Safety Principles for Pressurised Water Reactors in the French Nuclear Power Plant Fleet

Chapter 4
Safety Principles for France's Pressurised Water Reactors

Chapter 5
Development of the core melt accident

Chapter 1
Introduction

1.1. General objectives of the book

The operation of nuclear power reactors utilising nuclear fission involves risks of possible radioactive substance dispersion and human and environmental exposure to radiation. In order to mitigate these risks, the nuclear industry attaches the greatest importance to the safety of its facilities. The nuclear facilities are therefore designed, constructed and used in such a way as to prevent potential abnormal and emergency situations and limit their consequences. Furthermore, measures are taken to continuously improve the facilities' level of safety by acting upon feedback on their design and operation, periodically reassessing their safety and integrating advances in scientific knowledge and the applicable techniques.

Despite all the measures taken, however, the possibility of an accident resulting in partial or complete melting of the nuclear fuel contained in the reactor core and, over the relatively long term, large quantities of radioactive substances being released into the environment cannot be excluded, as the Fukushima Daiichi accident in Japan in March 2011 has shown. Studying this type of accident, which is commonly classified as a "severe accident", is an important element of the safety approach adopted for nuclear fission power reactors. It is done with the aim of setting up suitable measures to reduce the probability of such an accident and, should one nevertheless occur, to mitigate its impact upon populations and the environment. All stakeholders in the nuclear industry have conducted considerable research in France and worldwide with the aim of achieving this objective and so improving the equipment and procedures of the reactors currently in operation.

The objective of this book is to present the scientific aspects of core melt accidents, and notably the knowledge acquired through the research carried out over the course of the last thirty years in order to understand and model the physical phenomena that can occur in such an accident. It is intended for any reader wishing to obtain an overview of the knowledge acquired, any remaining gaps and uncertainties, and past and present research in the field of core melt accidents.

It therefore reviews the current state of knowledge and prospects regarding research in the field, little more than thirty years after the Three Mile Island (TMI) accident in the United States which resulted in the partial melting of the core but fortunately caused very minor radioactive releases, nearly four years after the Fukushima Daiichi accident which resulted in a core melt in three reactors and major radioactive releases, and during the construction of the first third-generation pressurised water reactors (PWRs) in France; in the case of these reactors, core melt accidents are being addressed at the design stage.

The preliminary lessons learned from the Fukushima Daiichi accident do not seem to fundamentally challenge the existing state of knowledge regarding the phenomenology of core melt accidents or highlight new, hitherto unknown phenomena. Four years after the accident, however, the full sequence of events is still not exactly known. Feedback from the TMI accident, in which the damage to the reactor core could only be seen when the damaged reactor pressure vessel was opened around seven years after the accident, leads us to suppose that it will take several years to reconstruct the detailed scenario of the accident that caused the radioactive releases. As long as the cores of the three damaged reactors remain inaccessible, the available data will be too limited to allow the progression of the damage to be reconstructed. It therefore seems too early to present any lessons learned from the Fukushima accident regarding the phenomenology of nuclear core melt accidents at this stage[1].

It should be noted that although the physical phenomena described in this book can occur in different models of French or foreign pressurised water reactors currently in operation or under study as well as widely in the boiling water reactors such as those at the Fukushima Daiichi site, this book focuses more specifically on the reactors currently in operation and under construction or planned in France: the second-generation 900, 1300 and 1450 MWe pressurised water reactors and third-generation 1600 MWe European Pressurised Water Reactors (EPRs).

1. Following the Fukushima Daiichi accident, the consequences of external hazards such as flooding and earthquakes have been assessed in greater detail with a view to preventing and mitigating the effects of a core melt accident. In France, the Prime Minister asked the President of the French Nuclear Safety Authority (ASN) to conduct a safety audit of the French nuclear facilities in 2011, giving priority to the power reactors, regarding the following five points: the flooding risks, the seismic risks, the loss of electrical power, the loss of the heat sink, and the operational management of accident situations. ASN therefore asked the nuclear facility operators to conduct additional safety assessments on their facilities with the aim of learning the first lessons from the events that occurred at the Fukushima Daiichi nuclear power plant, firstly in order to assess the robustness of the French nuclear facilities in confronting severe external events, and secondly in order to reinforce the existing safety measures to increase their robustness.

1.2. Structure of the book

Following this introduction, which describes the structure of this book and highlights the objectives of R&D on core melt accidents, this book briefly presents the design and operating principles (Chapter 2) and safety principles (Chapter 3) of the reactors currently in operation in France, as well as the main accident scenarios envisaged and studied (Chapter 4). The objective of these chapters is not to provide exhaustive information on these subjects (the reader should refer to the general reference documents listed in the corresponding chapters), but instead to provide the information needed in order to understand, firstly, the general approach adopted in France for preventing and mitigating the consequences of core melt accidents and, secondly, the physical phenomena, studies and analyses described in Chapters 5 to 8.

Chapter 5 is devoted to describing the physical phenomena liable to occur during a core melt accident, in the reactor vessel and the reactor containment. It also presents the sequence of events and the methods for mitigating their impact. For each of the subjects covered, a summary of the physical phenomena involved is followed by a description of the past, present and planned experiments designed to study these phenomena, along with their modelling, the validation of which is based on the test results. The chapter then describes the computer codes that couple all of the models and provide the best current state of knowledge of the phenomena. Lastly, this knowledge is reviewed while taking into account the gaps and uncertainties, and the outlook for the future is presented, notably regarding experimental programmes and the development of modelling and numerical simulation tools.

Section 5.1 provides a detailed description of the sequence of events of a core melt accident in the reactor vessel; it discusses the core damage in the reactor vessel (Section 5.1.1), the behaviour of the corium[2] at the bottom of the reactor vessel (Section 5.1.2), the reactor vessel failure (Section 5.1.3) and high-pressure core melt (Section 5.1.4). Section 5.2 concerns the phenomena that can result in an early[3] failure in the containment, consisting of direct heating of the gases within the containment building (Section 5.2.1), the "hydrogen risk" (Section 5.2.2) and the "steam explosion" risk (Section 5.2.3). Corium erosion of the concrete basemat of the containment building, which is one of the phenomena that can result in the containment failing later[4], is discussed in Section 5.3. Section 5.4 focuses on the phenomenology of corium retention and cooling, both within the reactor vessel by reflooding the reactor coolant system and outside it by reflooding the reactor pit (Section 5.4.1), as well as of the under-water cooling of the corium during the corium-concrete interaction (Section 5.4.2) and of corium spread (Section 5.4.3). Section 5.5 discusses the release and transport of the fission products (FPs). It covers the release of FPs both within the vessel (Section 5.5.2) and outside the vessel (Section 5.5.4), the transport of FPs within the primary and secondary coolant

2. The mixture of melt materials resulting from the degradation of the structures comprising the reactor core (the fuel rods, control rods, spacer grids and plates within the core).
3. The word "early" means within such a very short time that it is not possible to set up measures to limit the spread of the radioactivity in the environment and its potential consequences upon the populations.
4. "Later" is used as the opposite of "early".

systems (Section 5.5.3), the behaviour of the aerosols (Section 5.5.5) and the chemistry of the FPs (Section 5.5.6) within the containment building.

Chapter 6 focuses on the behaviour of the containment enclosures during a core melt accident. After summarising the potential leakage paths of radioactive substances through the different containments in the case of the accidents chosen in the design phase, it presents the studies of the mechanical behaviour of the different containments under the loadings that can result from the hazards linked with the phenomena described in Chapter 5. Chapter 6 also discusses the risks of containment building bypass[5] in a core melt accident situation.

Chapter 7 presents the lessons learned regarding the phenomenology of core melt accidents and the improvement of nuclear reactor safety from:

- the Three Mile Island accident that occurred on 28 March 1979 in the United States;
- the Chernobyl accident that occurred on 26 April 1986 in the Soviet Union's Ukrainian territory;
- the integral simulation testing of core melt accidents in the Phebus FP international research programme, which took place between 1993 and 2004.

For the reasons stated above (Section 1.1), it is too early to draw detailed lessons from the core melt accidents during the Fukushima Daiichi accident; as a result, this book does not contain a specific section on this accident. Further information on this accident is contained in the public report listed as reference document [1], which describes the initial analyses of the accident and its consequences one year after the accident.

Lastly, Chapter 8 presents a review of development and validation efforts regarding the main computer codes dealing with "severe accidents", which draw on and build upon the knowledge mainly acquired through the research programmes: ASTEC, which is jointly developed by IRSN and its German counterpart, GRS (*Gesellschaft fűr Anlagen- und Reaktorsicherheit*), MAAP-4, which is developed by FAI (Fauske & Associates, Inc.) in the United States and used by EDF and by utilities in many other countries, and MELCOR, which is developed by SNL (Sandia National Laboratories) in the United States for the US Nuclear Regulatory Commission (US NRC).

1.3. Objectives and approach of R&D on core melt accidents

1.3.1. Objectives

Analysis of the feedback, which includes an analysis of the incidents and, therefore, of the accidents, must be supplemented by research on safety notably relating to core melt accidents, as this is essential in maintaining and improving the safety of the nuclear reactors currently in operation.

5. An accident in which the containment building is bypassed can result in the direct release of radioactive products into the environment.

Research and studies of core melt accidents will undoubtedly not only provide a better understanding of the conditions under which the accidents occur as well as their sequence of events, but also improve our knowledge of their phenomenology with the aim of developing measures to stop them progressing and limit their effects. The results of this research can therefore be used to develop, on the basis of existing experience and knowledge, simulation tools and models that can predict the accidents' sequence of events and consequences, as these tools are used in the nuclear facilities' safety studies.

The knowledge acquired as a result of this research can also help to develop new concepts for improving safety and thereby reduce the risks and consequences of core melt accidents. This research includes that relating to the "core catcher" developed for the EPR with the aim of limiting the consequences of a core melt accident, which are described in Section 5.4.3.

1.3.2. International R&D

Even before the Three Mile Island accident, which occurred in 1979 in the United States (Section 7.1), probabilistic safety assessments were performed on core melt accidents that occurred in the United States, with the aim of assessing the risks of radioactive releases into the environment and the consequences of these releases upon the populations [2]. At the time, these studies were widely considered to be theoretical.

More advanced research programmes on core melt accidents began at the beginning of the 1980s, following the awareness caused by the Three Mile Island accident, which clearly demonstrated that a nuclear reactor core melt accident was possible. Most of the countries using nuclear reactors (United States, Finland, France, Japan, Germany, Belgium, Canada, South Korea, United Kingdom, Netherlands, Switzerland, Sweden, Russia as well as some central Europe and eastern European countries [Hungary, Czech Republic, Slovakia, Slovenia, Lithuania and Ukraine]) have conducted research programmes in the field of core melt accidents. The Chernobyl accident, which occurred in 1986 in the Ukraine (Section 7.2), has merely underlined the need to continue and extend the research in this field. In general, each of these countries has focused on one or more particular aspects of the issue, as the field is too vast to allow the investigation of all phenomena in any one national programme.

The United States was the first country to conduct major research in the field. The research programmes were directed by the US NRC and based on national laboratories including the Electric Power Research Institute (EPRI), SNL and the Oak Ridge National Laboratory (ORNL) [3].

In France, the first major research programmes on core melt accidents began at the beginning of the 1980s and include the Phebus CSD (severely degraded fuel) programme. Bearing in mind the number of its nuclear power plants, France, like the United States, has developed national or international programmes on almost all subjects relating to core melt accidents. This research is primarily conducted by IRSN, CEA, EDF and AREVA. All these entities either develop or help to develop simulation software and have facilities in which they conduct testing.

Extensive research has been carried out in the field of core melt accidents, involving very considerable human and financial resources as a result of their great complexity, as well as collaboration between nuclear stakeholders, industry groups, research centres and safety authorities, at both the national and the international levels. In France, IRSN, CEA, EDF and AREVA have conducted joint programmes on many subjects and participate in international programmes, including those supported by the European Commission through its Framework Programmes for Research and Development and those conducted under the auspices of the OECD. In particular, IRSN has jointly conducted the Phebus FP integral test programme with CEA from the end of the 1980s onwards, thereby structuring international research efforts regarding core melt accidents (Section 7.3).

As part of the Sixth Framework Programme, a Network of Excellence called SARNET (Severe Accident Research NETwork of excellence) was set up to optimise the use of the available resources and increase the knowledge acquired in Europe regarding core melt accidents, coordinated by IRSN. Between 2004 and 2008, SARNET consisted of around fifty organisations belonging to 19 European Union countries as well as Switzerland. As well as increasing the scientific knowledge acquired regarding core melt accidents, it has also defined new research programmes and set up the resources needed to ensure the sustainability of the knowledge gained and to transfer the knowledge on a wider level. In 2008, operation of the SARNET network ensured the consistency of the current state of knowledge and of the main remaining uncertainties regarding core melt accidents. As a result, the highest-priority areas for improvement have been identified and new research programmes proposed in order to fill in the remaining gaps [4]. The activities of the network, which include the new proposed subjects of research, have continued as part of the Seventh Framework Programme, as the network has now been joined by the US NRC, Canadian Nuclear Laboratories (CNL, formerly AECL) and two South Korean organisations (KINS and KAERI). This book benefits from the scientific consensus reached in this field [4].

Many international collaborative projects have also been set up with the help of the OECD. The work of the OECD Nuclear Energy Agency Committee for the Safety of Nuclear Installations (CSNI) encourages the kick-off and implementation of research programmes intended to reach a consensus regarding scientific and technical issues of joint interest, notably in the field of core melt accidents [5]. Their subjects are chosen as part of its working groups, which identify questions that have not been fully resolved as well as programmes or facilities that could be the subject of international collaborative projects (for example, see reference [6]). Since the OECD does not have its own budget for this type of action, it relies on contributions from participants.

In the field of simulation tools, CSNI has formed expert working groups with the aim of setting up validation matrices; it also organises International Standard Problems (ISPs), which compare the experimental results obtained by teams using different computer software for a given problem, improving the software concerned as a result [7]. Lastly, State-of-the-Art Reports (SOARs) are produced on subjects of joint interest, such as hydrogen distribution, hydrogen combustion and aerosol behaviour. These SOARs provide the widest possible view of a given problem by reviewing current knowledge and the remaining uncertainties, and may recommend areas for further research [5].

1.3.3. Approach

The objective of core melt accident research is to produce and collect scientific information that enables us to improve our understanding and description of the physical phenomena that take place when such an accident occurs. The characteristics of these physical phenomena are generally rarely experienced and studied outside the nuclear field. They involve specific materials whose chemistry and interactions are complex and must be studied under extreme temperature — and sometimes, radioactivity — conditions. In addition, the physics of core melt accidents combine the disciplines of energy with those of material physics, as well as those of aerosol physics and of fission product physics and chemistry. Couplings between elementary phenomena involving different technical or scientific disciplines must also be taken into account. These special characteristics complicate both the experimental approach and the theoretical approach.

The experimental approach is further complicated by a particular difficulty: accurately reproducing all or part of an accident transient can rarely be envisaged, both for questions of scale as well as for various technological reasons including the radioactivity of the materials involved, which can only be used experimentally in small quantities. As it is impossible to perform full-scale testing in this field and reproduce all accident situations, elementary tests (so-called "analytical" experiments) aimed at providing a detailed understanding of the elementary phenomena contributing to the situation under study must be conducted instead, and more general tests must be performed to confirm that nothing has been forgotten, considering the many interactions between the different physical phenomena. All this must be done at scales that are compatible with the facilities' technical and economic capacities while also maintaining the highest possible level of representativeness, allowing the acquired knowledge to be extrapolated to the full-scale power reactor — often using qualified models.

These characteristics lead us to choose a research approach that combines the following:

- analytical experiments that study the elementary phenomena while limiting the effects of other phenomena as much as possible within a range of parameters that is representative of can be expected in a core melt accident; the obtained results can be used to develop and qualify the models and determine the associated uncertainties;
- the assembly and coupling of all elementary models within computer codes with predictive capabilities;
- more global experiments intended to simulate as accurately as possible the situations that can be met in a power reactor in an actual accident scenario. These global experiments are used to validate the calculation tools in order to ensure that no important phenomena have been forgotten and the coupling of the phenomena has been modelled correctly. If any unexpected behaviour is noticed, the modelling is reviewed or a new campaign of analytical experiments may even be run. Due to their complexity and their generally high cost, few global tests are performed. As each of the tests involves a set of coupled

phenomena, the results are often difficult to interpret. The Phebus FP programme is a notable example of this type of testing, and its lessons are presented in Section 7.3 of this book.

The computer codes contain the knowledge produced by analysing the experimental data. The transposition of the experimental results to the power reactors is therefore based on these codes. Considering the importance of these computer codes, it is essential to assess their ability to correctly describe the accident. This explains the importance attached to physically qualifying the computer codes.

All of the experimental data used (analytical experiments and global experiments) form the experimental basis of the physical qualification of the computer code. Despite the degree of sophistication presently achieved by the computer codes developed in the field of core melt accidents (Chapter 8), these computer tools all still suffer from many uncertainties that must be carefully considered when used in safety studies. These uncertainties are of two main types:

– those resulting from the simplification of the physical models introduced in the calculation software, the representativeness limits of the software experimental qualification base and the lack of precision in the numerical resolution schemes;

– those resulting from the simplification introduced in the simulation tools used to describe an actual facility.

This somewhat theoretical description should enable the reader to form an idea of how core melt accident research operates. The approach described here will be illustrated in Chapter 5 of this book for each of the phenomena involved.

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Chapter 2
Design and Operation of a Pressurised Water Reactor

2.1. General information about reactor operation

The nuclei of some isotopes contained in nuclear fuel, such as ^{235}U and ^{239}Pu, can split up (fission) into two[1] smaller fragments called "fission products". These fragments have large amounts of kinetic energy that is mainly released as kinetic thermal energy in the surrounding fuel material. This release of energy is used to generate electricity in power reactors. Fission into two fragments can either be induced by neutrons (induced fission) or occur spontaneously in the case of heavy isotopes (spontaneous fission). Fission is accompanied by the release of two to three neutrons. Some of these neutrons may in turn initiate other fissions (the principle behind a nuclear chain reaction), be absorbed into the fuel without initiating any nuclear fission, or escape from the fuel.

Neutrons produced by fission from the neutrons of one generation form the neutrons of the next generation. The effective neutron multiplication factor, *k*, is the average number of neutrons from one fission that cause another fission. The value of *k* determines how a nuclear chain reaction proceeds:

– where $k < 1$, the system is said to be "subcritical". The system cannot sustain a chain reaction and ends up dying out;

1. In about 0.4%-0.6% of cases the fission can be into three fission products, this is termed "ternary fission".

– where $k = 1$, the system is "critical", i.e., as many neutrons are generated as are lost. The reaction is just maintained. This situation leads to a constant power level;

– where $k > 1$, the system is "supercritical". For every fission there will be an average of k fissions in the next generation. The result is that the number of fissions increases exponentially.

There are in fact two types of supercritical situation: prompt supercriticality and delayed supercriticality. Nearly all fission neutrons are immediately emitted (for example, 99.3% of neutrons are released as 10^{-7} s for ^{235}U); these neutrons are called "prompt neutrons". However, a small fraction of fission products are de-excited by beta decay (β decay) and subsequently emit what are termed "delayed neutrons". β decay occurs any time from a few tenths of a second to several tens of seconds after the fission event. The fraction of delayed neutrons is typically less than 1% of all the neutrons generated at any time in a chain reaction. During the interval between $k = 1$ and $k = 1/(1-\beta) \approx 1+\beta$, supercriticality is referred to as "delayed"; when $k > 1/1(1-\beta) \approx 1+\beta$, supercriticality is referred to as "prompt". The value of the fraction of delayed neutrons representing the interval between delayed and prompt supercriticality is defined as a "dollar" and depends on the isotope.

To produce energy, nuclear reactors operate in the region of delayed supercriticality for it is in this region that, thanks to the presence of delayed neutrons, changes in reaction rates occur much more slowly than with prompt neutrons alone. Without delayed neutrons, these changes would occur at speeds much too fast for neutron-absorbing systems to control.

The order of magnitude commonly used to express system departure from criticality is known as "reactivity" ρ, $\rho = 1 - 1/k$. Positive ρ values correspond to supercritical states and negative values correspond to subcritical states.

Chain reactions in nuclear reactors must be controlled, i.e., zero or negative reactivity must be maintained with the aid of neutron-absorbing elements. In pressurised water reactors, these elements are either placed inside mobile devices called control rods (containing chemical elements such as cadmium and boron) or dissolved in the cooling water (boron).

In some low-probability accidents, the reactivity of the reactor may reach high positive values that cause the chain reaction to become supercritical. If the measures taken are insufficient to bring the reactor back to a safe condition, such accidents could lead to an uncontrollable power increase that could result in severe reactor damage like that which occurred during the Chernobyl accident (Section 7.2).

The reactivity of a reactor is affected primarily by the temperature of both the fuel and the coolant and by the coolant void fraction. The influence of each of these parameters is characterised by a reactivity coefficient, which is the derivative of the reactivity with respect to the parameter considered. In the case of fuel, an increase in power results in an increase in fuel temperature and an increase in neutron capture by ^{238}U. The reactivity coefficient, called the temperature coefficient or the Doppler coefficient, is therefore negative. In the case of coolant, the reactivity coefficient is related to changes in the coolant density (temperature coefficient) or void fraction (void coefficient). These coefficients are negative in pressurised water reactors[2] to ensure reactor stability and limit the maximum power that could be reached during an accident.

Some fission products formed are radioactive. This radioactivity results in, even after the chain reaction stops, energy being released in the form of heat (called "decay heat"). This heat decreases over time and, one hour after reactor shutdown, amounts to approx. 1.5% of its level during operation[3].

The energy released by fissions and fission products must be continuously removed to avoid an excessive rise in reactor temperature. In pressurised water reactors, this energy is removed during normal conditions by three successive loops whose main purpose is to prevent the radioactive water exiting the core from leaving the plant (Figure 2.3):

- the first loop is the reactor coolant system (RCS). It cools the core by circulating water at an average temperature of around 300 °C and a pressure of 155 bar;
- the secondary loop extracts the heat from the RCS by means of steam generators, which supply steam to the turbine generator to produce electricity;
- the tertiary system consists of a condenser and rejects the remaining heat to a river or the sea or to the atmosphere by means of cooling towers.

This brief description of the operation of a nuclear reactor identifies the basic safety functions that must be ensured at all times:

- reactivity control;
- heat removal;
- containment of fission products and, more generally, radioactivity (some activation products in the RCS[4] are also radioactive).

2.2. The pressurised water reactors in France's nuclear power plant fleet

Various types of nuclear reactor are used to generate electricity in France. They use different fissile materials (natural uranium, uranium enriched in uranium-235, plutonium, etc.) and different neutron moderators (graphite, water, heavy water, etc.)[5]. They

2. Water is used as the moderator in pressurised-water reactors. It decelerates neutrons produced by fission (these neutrons lose their kinetic energy by colliding with the nuclei of the water's hydrogen atoms) and increases fission product yields. As the temperature inside the reactor core increases, the water expands. This reduces the water's ability to slow down neutrons and results in fewer fission reactions. The temperature coefficient of the water is thus negative.
3. One hour after reactor shutdown, a 900 MWe reactor generates 40 MW of heat and a 1300 MWe reactor generates 58 MW of heat. One day after shutdown, this heat output drops to 16 MW for a 900 MWe reactor and 24 MW for a 1300 MWe reactor.
4. Radioactive substances may be formed under irradiation by activation of the metal components in the RCS and be entrained into the reactor coolant by corrosion mechanisms.
5. The moderator reduces the velocity of the neutrons, thereby increasing their likelihood of producing a fission reaction.

are also characterised by the type of coolant (ordinary water in liquid or vapour form, heavy water, gas, sodium, etc.) used to remove heat from the core (where fission reactions occur) and transfer it either to the loops supplying the turbine generators or to the turbine generators directly.

The nuclear power plants currently in operation in France use enriched uranium in oxide form that may be mixed with plutonium oxide recovered from the reprocessing of spent fuel. They use ordinary water as the heat-transfer fluid. This water is maintained under high pressure (155 bar) so that it remains in liquid form at its operating temperature (300 °C). They are known as pressurise water reactors (PWRs) and belong to what is commonly known as the second generation of nuclear power reactors[6].

A distinctive feature of France's reactor fleet is its standardisation. The technical similarity of many of the country's reactors justifies the generic overview given in this chapter. The 19 nuclear power plants in operation in France have two to six PWRs, giving a total of 58 reactors. This reactor fleet consists of three series: the 900 MWe series, the 1300 MWe series, and the 1450 MWe (or N4) series (Figure 2.1).

The thirty-four 900 MWe reactors are split into two main types:

- CP0, which consists of the two reactors at Fessenheim and the four reactors at Bugey;
- CPY (consisting of types CP1 and CP2), which encompasses the 28 other reactors (four reactors at Blayais, four at Dampierre, six at Gravelines, four at Tricastin, four at Chinon, four at Cruas-Meysse and two at Saint-Laurent-des-Eaux).

The twenty 1300 MWe reactors are split into two main types:

- the P4, which consists of eight reactors: two at Flamanville, four at Paluel and two at Saint-Alban;
- the P'4, which consists of 12 reactors: two at Belleville-sur-Loire, four at Cattenom, two at Golfech, two at Nogent-sur-Seine and two at Penly.

Lastly, the N4 series consists of four 1450 MWe reactors: two at the Chooz nuclear power plant and two at the Civaux nuclear power plant.

Despite the deliberate standardisation of France's fleet of nuclear power reactors, technological innovations have been introduced during the design and construction of each plant. The creation of France's fleet occurred in four main stages:

- the CP0 900 MWe "preproduction" series was brought into operation between 1977 and 1979;
- the CPY 900 MWe series was brought into operation between 1980 and 1987;

6. Reactors built before the 1970s make up the first generation. The Generation-I reactors in France were graphite moderated, cooled by carbon dioxide, and fuelled with natural uranium metal. They were a type of gas-cooled reactor (GCR).

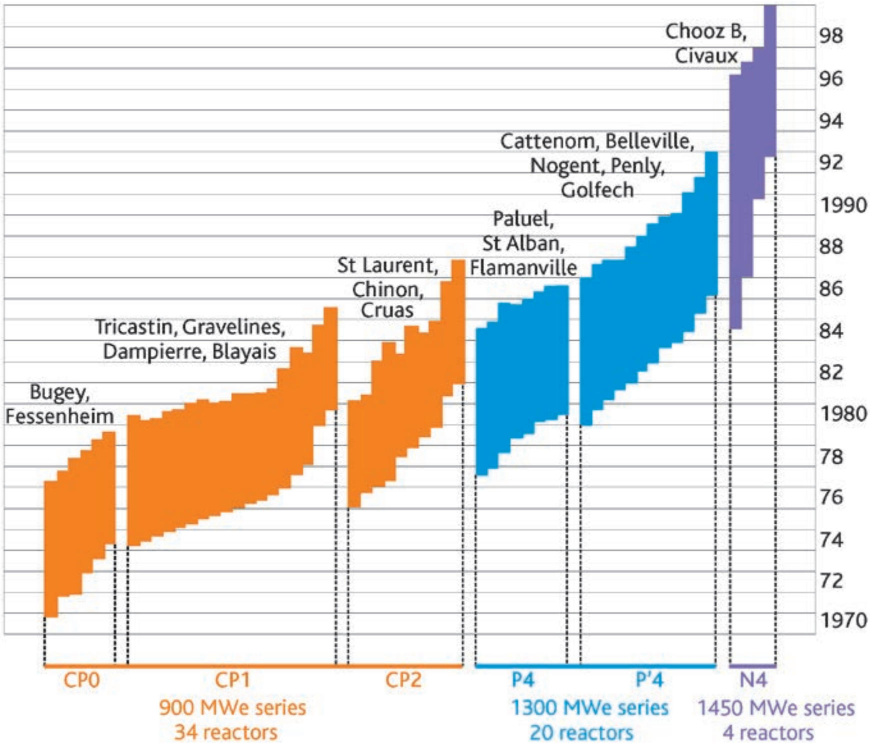

Figure 2.1. Construction periods and distribution of the three series of 900, 1300 and 1450 MWe power reactors in operation in France in 2015.

– the P4 and P'4 1300 MWe series were brought into operation between 1984 and 1993;

– the 1450 MWe (or N4) series was brought into operation between 2000 and 2002.

The CPY reactors benefited from the feedback obtained from the design studies, construction and operation of the CP0 reactors. Unlike the design studies for the CP0 series, which were conducted separately for each site, the design studies for the CPY series were conducted for all the sites. As a result, the CPY series differs from the CP0 series in terms of building design (in particular, the containment building was modified to facilitate operations), siting of the engineered safety systems (which were modified to increase the independence of the systems' trains and increase their reliability) and more flexible reactor control (particularly via the use of control rods and the addition of control rods with less neutron-absorbing capacities[7]). In the case of the CP2 reactors, the orientation of the

7. The control-rod clusters are made up of 24 rods. There are two types of control-rod cluster, "black" and "grey". Black clusters have 24 neutron-absorbing rods (consisting of a silver, indium and cadmium alloy (Ag-In-Cd) or boron carbide [B_4C]). Grey clusters consist of rods made of materials with varying degrees of absorbency (e.g., only eight Ag-In-Cd or B_4C absorbing rods and 18 rods made of steel, which is more transparent to neutrons). Moving these clusters at different rates in the core makes it possible to optimise the spatial power distribution, control changes in reactor power and adjust the mean temperature of the reactor coolant.

control room was shifted by 90 degrees to prevent projectiles generated by rupture of the turbine generator from damaging the reactor containment vessel (Figure 2.2).

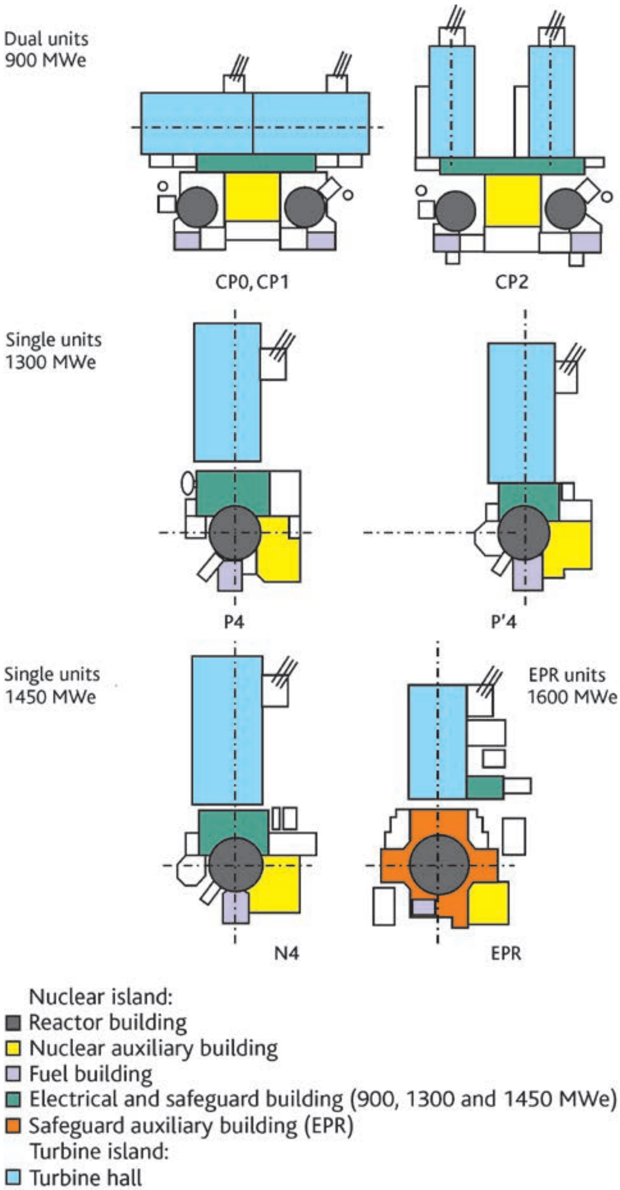

Figure 2.2. Schematic plant layout showing the buildings of the different reactor series in operation in France.

The 1300 MWe reactors differ from the 900 MWe reactors in terms of the design of their core, loops and reactor protection system as well as their buildings. The increase in power was achieved by increasing the size of the reactor. In order to remove the increased heat (from 900 to 1300 MWe), the cooling capacity of the RCS was increased by the installation of an additional cooling loop (thus changing the number of loops from three for the 900 MWe reactors to four for the 1300 MWe reactors) (Figure 2.3). The components of each RCS are also larger than those of the preceding series. In terms of the locations of the buildings, the new series are single-unit plants, whereas the preceding series were dual-unit plants (Figure 2.2). The engineered safety systems and auxiliary systems are located in buildings specific to each unit so as to improve the safety of their operation. In addition, each containment vessel has a double concrete wall (an inner wall of prestressed concrete and an outer wall of reinforced concrete) instead of the single wall of steel-lined prestressed concrete on the 900 MWe reactors. New microprocessor-based instrumentation and control technologies using programmable memory are used. The P'4 series differs from the P4 series in that the installation of the buildings and structures was optimised with the primary goal of reducing costs. The result is a denser plant layout and smaller buildings and structures.

Lastly, the main differences between the 1450 MWe reactors and those of the preceding series are the larger reactor core, smaller steam generators (SG) that delivery steam at higher pressure, the design of the reactor coolant pump (higher flow rate) and the computerised control system.

The next generation of reactor that EDF is planning to put into service in France will consist of a design known as the European Pressurised Water Reactor, or EPR). A reactor with a power output of around 1600 MWe is currently under construction at EDF's Flamanville site, on France's Cotentin Peninsula on the English Channel. These new PWRs incorporate evolutionary improvements over earlier designs. They therefore benefit from extensive operating experience feedback from the current fleet and meet more stringent safety objectives. They also benefit research findings, particularly regarding core melt accidents, which were factored in right from the design phase. Their main differences with the Generation-II PWRs are the design of the loops, the reactor protection system and the site buildings (particularly the containment), which offer a higher degree of protection in the event of an accident.

The design of the RCS and the main components and the configuration of the loops are quite similar to those of the N4 series. The main evolutionary improvements are as follows:

- increase in the volumes of primary and secondary water (particularly in the steam generators) to increase the thermal inertia of the reactor;
- organisation of the engineered safety systems and the support systems (safety injection system [SIS], steam generator emergency feedwater system [EFWS], component cooling-water system [CCWS], essential service-water system [ESWS], emergency power supplies [EPS]) into four independent trains located in physically separate rooms. This physical separation ensures that the engineered safety systems remain available in the event of an internal or external hazard (e.g., fire, earthquake or flood).

Regarding the containment, in addition to the reinforcement of its structure (more specifically the outer concrete wall, see Section 2.3.2.3), the following changes have been made in relation to those of the N4 series:

- placement of the borated-water storage tank inside the containment, hence the name "in-containment refuelling water storage tank" (IRWST). The IRWST feeds the safety injection system and the containment heat-removal system (CHRS);
- installation of a system for containing and cooling molten corium inside a special compartment in the event of a vessel melt-through during a core melt accident. The purpose of this system is to provide long-term protection of the basemat from erosion should such an accident occur;
- installation of a steel liner on the inner wall of the double-wall containment.

Another notable difference with the N4 series is that more rooms are protected by the reinforced-concrete outer wall (airplane crash [APC] shell). In addition to the reactor building, the fuel building and two of the rooms housing the engineered safety systems are covered by the outer concrete wall.

The layout of the buildings (Figure 2.2) was changed so that the four independent trains of the engineered safety systems and support systems could be housed in separate rooms and thus prevent leaks being released directly into the environment from the containment. All the containment penetrations lead into buildings located around the reactor building and equipped with ventilation and filtration systems.

To provide the reader with the information needed to understand the concepts presented in this document, the rest of this chapter provides a relatively generic, summary overview of the main components of the reactors in operation in France and of how these reactors function under normal and accident conditions. The specific features of the EPR are described whenever they relate to core melt accidents.

2.3. Description of a pressurised water reactor and its main loops

2.3.1. Facility overview

Each reactor comprises a nuclear island, a turbine island, water intake and discharge structures and, in some cases, a cooling tower (Figures 2.2 and 2.3).

The main parts of the nuclear island are:

- the reactor building (RB), which contains the reactor and all the pressurised coolant loops as well as part of the loops and systems required for reactor operation and safety (Figures 2.3, 2.6 and 2.7);
- the fuel building (FB), which houses the facilities for storing and handling new fuel (pending its loading into the reactor) and spent fuel (pending its transfer to reprocessing plants). The fuel building also contains the equipment in the

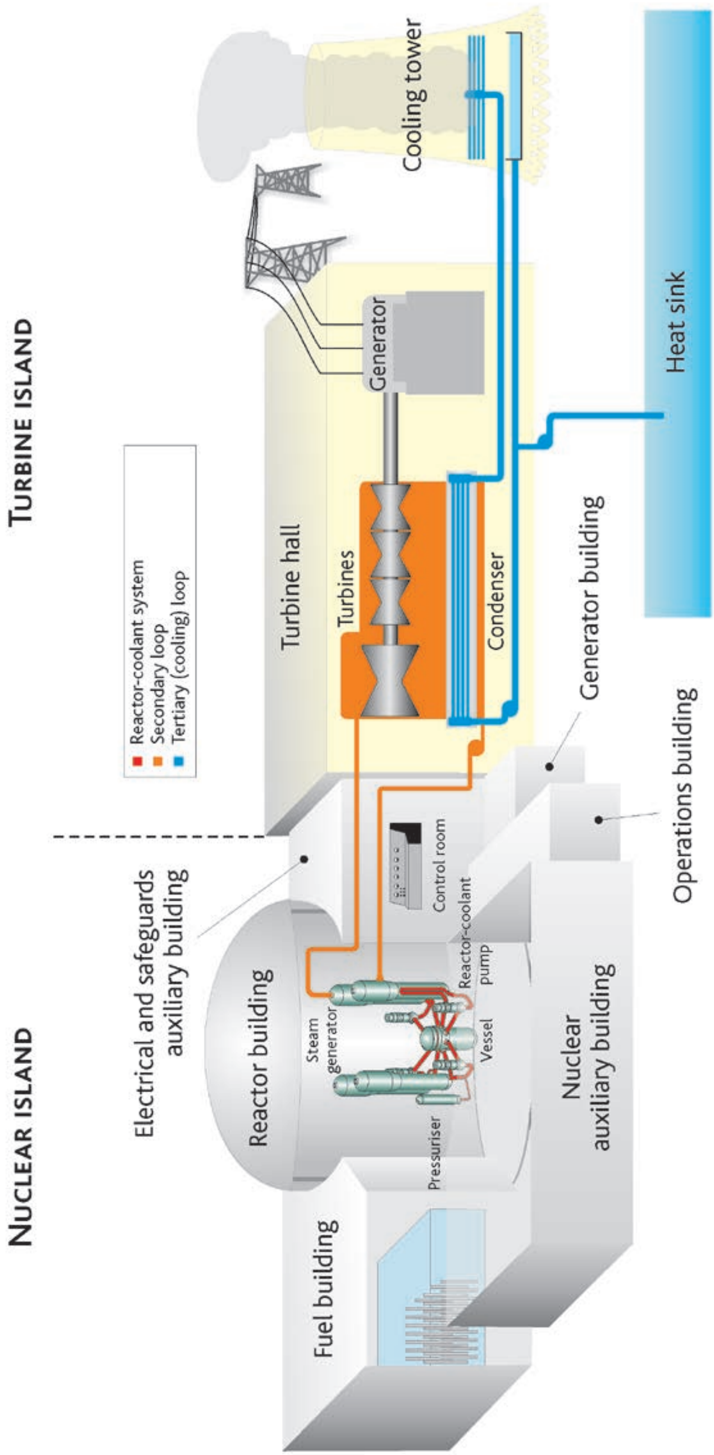

Figure 2.3. Schematic diagram of a PWR (1300 MWe or N4) and its main loops.

fuel pool cooling and purification system (FPCPS) and, for units in operation, the equipment in the steam generator emergency feedwater system (EFWS). The EPR itself has four independent steam generator emergency feedwater trains. Each train is located in one of the four divisions of the safeguard auxiliary building;

– a safeguard auxiliary building (SAB) with electrical equipment rooms. The main engineered safety systems are located in the SAB's bottom half and the electrical equipment rooms are located in its top half. These two halves do not communicate with each other. The rooms in the SAB contain equipment, particularly that of the safety injection system (SIS), the containment spray system (CSS), the component cooling water system (CCWS) and ventilation equipment. The electrical equipment rooms contain all the means for controlling the unit (the control room and operations facilities, electric power supplies, and the instrumentation and control [I&C] system). Note that, in the case of the 900 MWe series, there is only one SAB with electrical equipment rooms for two adjoining units. In the case of the 1300 MWe and N4 series, there is only one building per unit. The EPR has four independent engineered safety systems. Each is located, with its support systems, in a room that is physically separate from the others. These rooms are known as the "divisions" of the SAB. Divisions 2 and 3 of the SAB are protected by the reinforced-concrete outer wall. The control room is located in division 3 of the SAB;

– a nuclear auxiliary building (NAB) housing the auxiliary systems required for normal reactor operation. This building houses the equipment of the chemical and volume control system (CVCS), the gaseous waste processing system, the reactor coolant effluent processing system and the boron recycle system;

– two geographically separate buildings, each housing a diesel generator (emergency power supply). In the case of the EPR, the offsite emergency power supplies consist of two sets of four diesel generators (each set being housed in its own building) and two station blackout (SBO) generators;

– an operations building.

The turbine-island equipment converts the steam generated by the nuclear island into electricity and supplies this electricity to the transmission system. The main parts of the turbine island are:

– the turbine hall, which houses the turbine generator (it receives the steam generated by the nuclear island and converts it into electricity) and its auxiliary systems;

– a pump house to cool the facility under normal operating conditions and provide emergency cooling with the related hydraulic structures;

– a cooling tower in the case of closed-loop cooling.

Some of these items of equipment contribute to reactor safety. The secondary loops are the interface between the nuclear island and the turbine island.

2.3.2. Description of the main components of a PWR

2.3.2.1. Reactor core

The reactor core is made up of fuel assemblies (Figure 2.4). Each assembly consists of 264 fuel rods (Figure 2.4, left), 24 tubes to contain the rods of a control rod cluster and a guide tube. All are arranged in a 17 × 17 square lattice (Figure 2.4, right). The fuel rods are made up of zirconium alloy tubes also known as "cladding" (zirconium has low neutron-absorbing properties and good corrosion resistance). Zircaloy, which contains 98% zirconium, is the alloy most frequently used in France's PWRs. The cladding, which is 0.6 mm thick and 9.5 mm in diameter, is held in place by Zircaloy grids. Pellets made of uranium dioxide (UO_2) or a mixture of uranium and plutonium oxides ($(U,Pu)O_2$, commonly referred to as MOX fuel) and measuring 8.2 mm in diameter are stacked inside the rods. These pellets make up the nuclear fuel. The level of ^{235}U enrichment varies

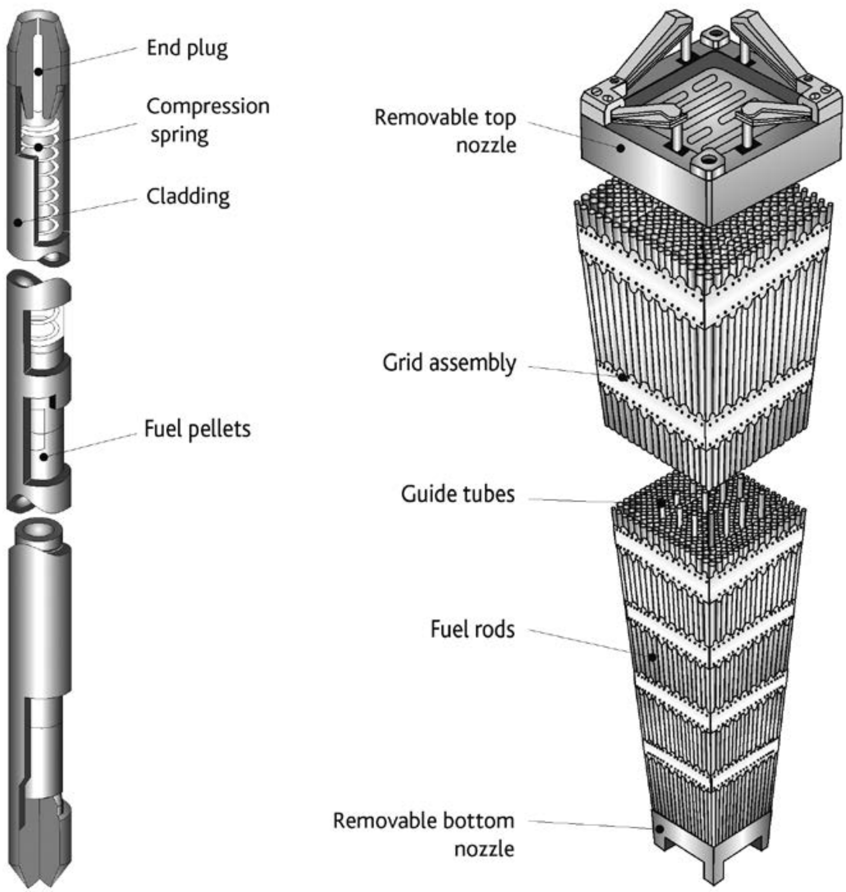

Figure 2.4. Diagram of a fuel rod (left) and of the main components of a fuel assembly (right).

between 3% and 4.5% depending on the method of fuel management[8]. The fuel assemblies are similar for all the series. Only their lengths change. One-third to one-fourth of the fuel is replenished once every 12 to 18 months during reactor outages.

The main characteristics of the fuel and the core are given for each series in Table 2.1.

Table 2.1. Characteristics of the cores of each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Number of fuel assemblies	157	193	205	241
Total height of the fuel pellets in each assembly rod (m)	3.66	4.27	4.27	4.20
Number of control rod clusters Absorbing material	57 Ag-In-Cd	65 Ag-In-Cd + B_4C	73 Ag-In-Cd + B_4C	89 Ag-In-Cd + B_4C
Mass of enriched uranium (t)	72.5	104	110.5	144.2

The core is located inside a vessel made of 16MND5 low-carbon steel fitted with an upper head that is removed for refuelling purposes (Figure 2.5). Inside the vessel are

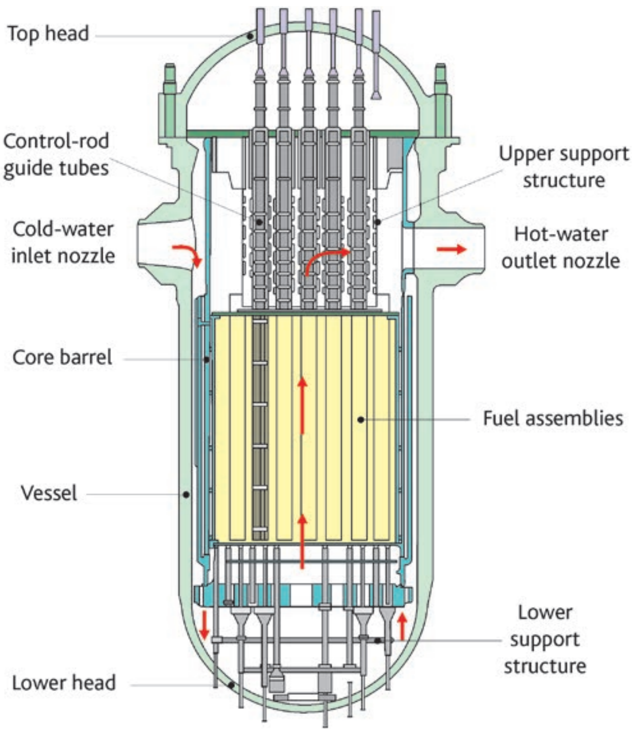

Figure 2.5. Cutaway of the PWR vessel at Fessenheim.

8. During reactor operation, the amount of fissile material in the fuel diminishes, requiring the spent fuel rods to be replaced by new assemblies. The method of managing this replacement depends on the initial enrichment of fissile material within the fuel.

metal structures (known as internals) that can be completely removed to facilitate periodic inspections:

- the lower structures support the core;
- the side structures (core barrel) separate the cold fluid entering the vessel from the hot fluid exiting the core;
- the upper structures are made up of the control rod guide tubes.

The dimensions of the vessels of each series are given in Table 2.2.

Table 2.2. Dimensions of the vessels of each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Inside diameter (m)	4.00	4.39	4.486	4.885
Height (m)	12.3	13.6	13.645	13.105
Cladding thickness at core level (m)	0.20	0.22	0.225	0.25

2.3.2.2. Reactor coolant system and secondary loops

The reactor coolant system (RCS) carries heat away from the reactor core by circulating pressurised water (known as reactor coolant) through the heat transport loops (there are three for a 900 MWe reactor, four for a 1300 MWe reactor, a 1450 MWe reactor or an EPR). Each loop is connected to the reactor vessel, which contains the core, and is equipped with a reactor coolant pump (RCP). This pump circulates the coolant heated through contact with the fuel elements to heat exchangers, called steam generators, where the coolant transfers its heat to the secondary loops and flows back to the reactor (Figures 2.3 and 2.6). The RCPs are fitted with seals that are continuously cooled by pressurised water to prevent reactor coolant from leaking outside the RCS.

The steam generators are evaporators composed of a bundle of U-tubes and a secondary side with integral moisture-separation equipment. The reactor coolant enters the inverted U-tubes and heats the secondary-side water, which flows in through a nozzle located above the tube bundle. The steam generated rises through the moisture separators and exits through the top of the steam generator.

A tank, called a pressuriser, allows the coolant to expand and maintains the RCS pressure at 155 bar so that the coolant (heated to over 300 °C) remains in liquid form. The reactors in operation have three letdown lines, each of which has an isolation valve and a safety valve. In particular, these valves enable emergency blowdown of the RCS to prevent high-pressure core melt.

The upper section of the EPR pressuriser has three letdown lines, each of which has a pilot valve fitted with a position sensor. The EPR also has an emergency RCS blowdown system consisting of a set of motor-operated valves that are actuated to avert high-pressure core melt. This system consists of two parallel letdown lines connected to the same nozzle at the top of the pressuriser. Each line is fitted with two motor-operated valves and is connected to a shared letdown line that leads to the pressuriser relief tank. This system is described in Section 4.3.4.

The normal operating conditions of the RCS for each series are given in Table 2.3.

For each unit, the RCS is completely located inside the containment.

Table 2.3. Normal operating conditions of the RCS for each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Number of loops	3	4	4	4
Nominal absolute RC pressure (bar)	155	155	155	155
Nominal flow rate (m^3/h)	21,250	23,325	24,500	27,195
RCS volume, pressuriser included (m^3)	271	399	406	460
Nominal temperature of the water at the vessel inlet (°C)	286	293	292	296
Nominal temperature of the water at the vessel outlet (°C)	323	329	330	330

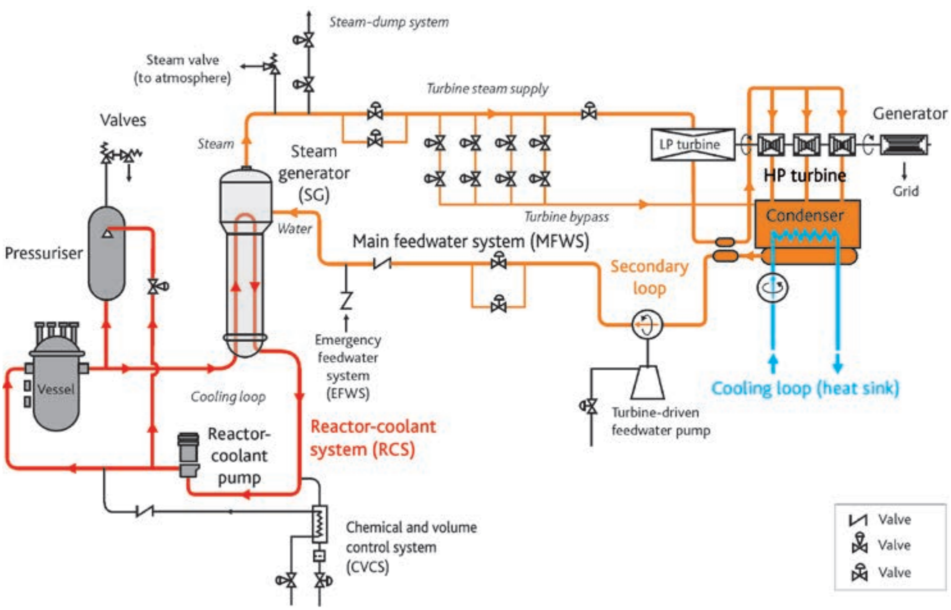

Figure 2.6. Schematic diagram of the main components of the RCS and the secondary loops.

During normal operation, the secondary loops convert the thermal energy produced by the core into electrical energy. To prevent radioactive coolant from leaving the containment, the secondary loops are separated from the RCS by the pipes of the steam generators. The reactor coolant flows through these pipes, where its heat is transferred to the water in the secondary loops. This water is vaporised then expands in the steam turbine connected to the generator (Figure 2.6). The steam is generated in these loops at a pressure of 58 bar absolute (900 MWe reactors), 65 bar absolute (1300 MWe reactors), 73 bar absolute (1450 MWe reactors) or 77 bar absolute (EPR). It exits the turbine and flows into a condenser that is cooled by water from a river or sea. In some instances, the water is cooled by contact with air inside a cooling tower.

The upper sections of the steam generators are connected to the turbine's steam chest *via* three or four lines[9] (one per steam generator) (Figure 2.6). Each line has:

– a flow restrictor inside the outlet pipe of the steam generator;
– a steam dump system equipped with an isolation valve and a control valve;
– seven (two for the N4 series and the EPR) safety valves with steam dump pipes;
– an isolation valve that closes in a matter of seconds.

The flow restrictor slows down the rate of cooling and depressurisation of the secondary loop and reduces the forces exerted on the tube bundle in the event of a steam line break. The valves protect the loop against overpressure if the steam can no longer be dumped. The bypass is used to temporarily send steam directly to the condenser without passing through the turbine or activating the valves. It is used especially to remove heat from the core during startup, hot shutdown or cold shutdown of the reactor and until the residual heat removal system (RHRS) is turned on (Figure 2.7). The steam dump system discharges the residual heat, thus cooling the reactor core if it can no longer be cooled by the normal systems, and avoids having to open the safety valves in the event of rupture of one or more steam generator lines. This system consists of one line per steam generator for the 900 and 1300 MWe series, two lines per steam generator for the N4 series, and only one line for the EPR. Each line has a dump valve and an isolation valve.

The characteristics of the secondary loops are given for each series in Table 2.4.

Table 2.4. Characteristics of the secondary loops for each series.

Series	900 MWe	1300 MWe	1450 MWe	EPR
Number of steam generators (SG)	3	4	4	4
Secondary-side steam pressure at the SG outlet (bar absolute)	58	65	73	77
Heat-transfer area in an SG (m^2)	4746	6940	7308	7960
Steam flow rate (t/h) per SG	1820	1909	2164	2197
Steam temperature at the SG outlet (°C)	273	281	288	293

9. In the EPRs, each of the four lines is located in a separate room.

2.3.2.3. Containment

The containment is made up of the reactor building, which houses the RCS, a portion of the secondary loops (including the steam generators), and a number of auxiliary operating and safety systems. The reactor building is a concrete cylinder topped by a concrete dome. It forms a strong barrier that offers the specified level of integrity (see Chapter 6 for more details), prevents radioactive substances from escaping into the outside environment, and protects the reactor from external hazards. The reactor buildings of PWRs currently in operation are designed to withstand the pressure (4 to 5 bar absolute) expected during a loss-of-coolant accident (LOCA with a double-ended guillotine break of a main coolant pipe) or rupture of a steam line inside the containment. They ensure a satisfactory level of integrity should either situation occur. The containment of the EPR is designed to withstand a higher pressure of approx. 6.5 bar absolute.

Whatever the reactor type, the concrete walls of the containment rest on a foundation, or basemat, which is also made of concrete. The walls are topped by a concrete dome that forms the roof of the building. The reactor building is designed to withstand the effects of a seismic margin earthquake (SME) (the magnitude of the SME is determined based on the magnitudes of the maximum historically probable earthquakes [MHPE] and by taking into account a safety margin that covers uncertainties, amongst other aspects) and environmental hazards (extreme weather conditions, aircraft crashes, explosions, etc.).

The reactor building penetrations are distinctive points of the containment. Pipes, electrical wiring and ventilation ducts are routed through orifices in the containment walls. There are also access hatches, or locks, for personnel and large items of equipment. Lastly, there is a canal, or pipe, for transferring fuel assemblies between the reactor building and the fuel building. Some water and steam pipes, particularly the portions of the secondary loops inside the reactor building and the outer portions leading to the isolation valves, are an extension of the containment. The secondary shell of the steam generators and the tube bundles on the primary side are also an extension of the containment.

All these penetrations have a specified level of integrity (see Chapter 6 for more details). With the exception of the water and steam penetrations on the secondary loops, these penetrations are fitted with isolation devices located inside the containment. These isolation devices, which are closed before or during an accident, are located on the fluid inlets and outlets. The isolation valves for the water and steam penetrations on the secondary loops are located inside the containment and after the safety valves (see the description of the secondary loops in the preceding section).

Before the reactor is first brought online, the containment is inspected and tested to determine its overall integrity and its resistance to forces under normal and accident conditions. All these aspects are explained in Chapter 6 of this document.

Internal components (known as internals) support equipment, provide biological shielding of personnel, and physically separate the loops (particularly the coolant loops) and some items of equipment.