IRSN
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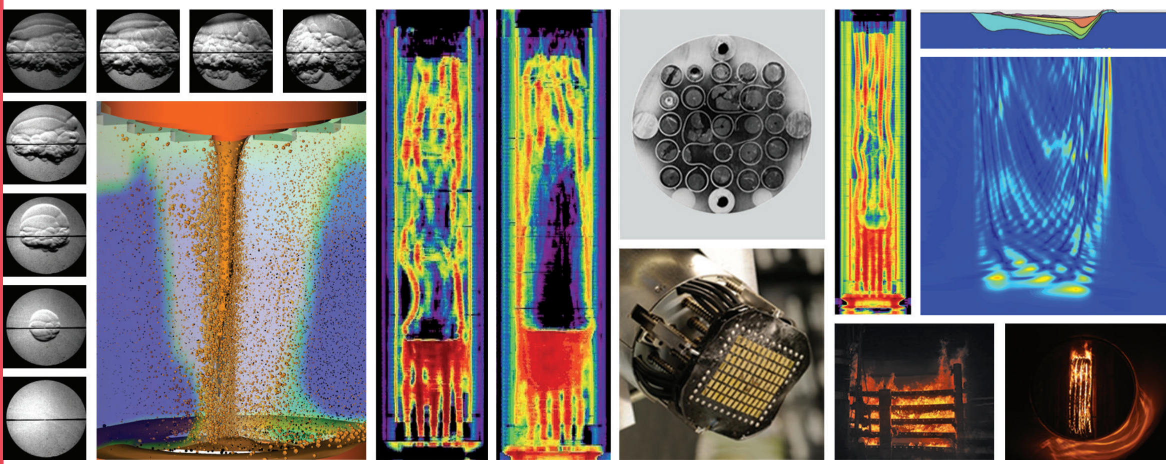
Enhancing nuclear safety

Jean Couturier, Michel Schwarz

Current State of Research on Pressurized Water Reactor Safety



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Science and Technology Series

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Cover illustrations: photographs showing research and development in pressurized water reactor safety at IRSN. From left to right: the hydrogen risk in nuclear reactors © IRSN Institut Icare; image from a 3D simulation by the MC3D code of corium dispersion in the water in a reactor pit in the event of a central reactor vessel failure © IRSN; states of degradation of fuel assemblies from the Phébus-PF program © IRSN; a prototype of the conformable transducer – IRSN/CEA patent © IRSN/CEA (bottom); Phébus-LOCA – section view (post-mortem) of a test fuel cluster following a temperature transient typical of a LOCA © IRSN (top); an experiment involving a fire in a stack of cable raceways © Florent–Frédéric Vigroux/IRSN (bottom); 2D model of the Nice basin; below, amplification of seismic waves calculated by a numerical simulation of seismic wave propagation in this model © Fabien Peyrusse/Inria; fire test on electrical cables © Florent–Frédéric Vigroux/IRSN (bottom).

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Preface

This publication sets out to provide an overview of much of the research and development work carried out over the past forty years in the field of pressurized water reactor safety, in particular by the Institut de Protection et de Sûreté Nucléaire (IPSN), later becoming the Institut de Radioprotection et de Sûreté Nucléaire (IRSN), whether alone or in collaboration with other organizations. This work – and the lessons that can be learned from it – is set in front of the safety issues raised during assessments of French nuclear power plants, or following incidents and accidents that have occurred, such as those affecting reactor 2 at the Three Mile Island nuclear power plant in 1979, reactor 4 at the Chernobyl nuclear power plant in 1986 and, more recently, at the Fukushima Daiichi nuclear power plant in 2011.

The safety of nuclear facilities is part of a continuous improvement process that is based on:

- national and international operating experience feedback,
- and knowledge acquired through research and development, with assessment and research activities fueling each other.

Since the 1980s, safety reviews have been carried out at nuclear power plants operated by Électricité de France (EDF), the national power utility, as part of a ten-yearly outage program. These provide an opportunity to ensure that lessons learned from operating experience feedback and R&D work flow down to the operational level.

The research and development themes relating to pressurized water reactor safety in which IRSN has been – and for the most part still is – particularly involved are part of the general initiative to prevent and mitigate the impact of postulated events. These include internal events, such as the loss of coolant or an uncontrolled increase of reactivity in the core until it melts, and "hazards" such as earthquakes, external flooding, and fires inside or outside facilities. The knowledge acquired through research on core melt accidents is fed into ASTEC, an IRSN computer code used for simulation purposes, and internationally recognized as a reference tool.

The authors also wished to provide a panorama of studies and research work relating to human and organizational factors, a theme that has been of concern to the international community since the accidents at Three Mile Island and Chernobyl. The safety issues raised by the growing tendency of licensees to work with subcontractors have become major themes of studies and research today. Furthermore, since the accident at the Fukushima Daiichi nuclear power plant, and the conclusions drawn from the event – in particular those of the National Diet of Japan Fukushima Nuclear Accident Independent Investigation Commission, according to which various social factors contributed to the accident – studies and research initiatives have covered topics in the human and social sciences.

For each of the research and development themes addressed, the remaining uncertainties and the new knowledge required are highlighted. Research in new areas, such as topics relating to new reactor designs (passive safety, the possibility of retaining the corium in the reactor vessel in the event of a core melt accident, etc.) is also discussed.

The research and development work discussed in this publication illustrates IRSN's policy of opening up to the international community and the many ties it has developed with it. Examples include direct ties with similar organizations (such as the United States NRC or GRS in Germany) research operators (in particular the Commissariat à l'énergie atomique et aux énergies alternatives [CEA] and universities) as well as involvement in knowledge sharing organizations (such as the OECD Nuclear Energy Agency) and the many projects funded in France by the Agence nationale de la recherche (ANR) and at the european level by the European Commission through its multi-year Framework Programs (FP) for research and development. In addition, IRSN is closely involved in defining European research strategies within the European Sustainable Nuclear Energy Technology Platform (SNETP) and, since 2012, as a member of the NUGENIA association.

I would especially like to thank the two main contributors, Jean Couturier and Michel Schwarz, for this impressive summary (representing more than three years' work) and, more generally, all the IRSN experts for their invaluable assistance in this work.

Jean-Christophe Niel
IRSN Director-General

The authors

Jean Couturier works at the office of IRSN's Director General, supporting the roll-out of a knowledge management program. He is also a senior expert (safety policies, risk analyses). He began his career working on the design of fast reactors. From 1982, his work focused on nuclear safety not only for this type of reactor (PHENIX, SUPERPHENIX) but also for research reactors, pressurized water reactors, etc. He is a member of the Standing Group of experts for Reactors and the Standing Group of experts on Nuclear Pressurized Equipment.

Michel Schwarz retired from IRSN in 2012. He spent his career in nuclear safety research. In particular, he ran the Phébus-PF international research program on core melt accidents in light water reactors. He was director of major accident prevention and then IRSN's scientific director. He is a member of ASN's Scientific Committee.

List of abbreviations

Glossary of institutions

AEC: Atomic Energy Commission, USA (forerunner of the U.S.NRC)

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

AFPS: Association française du génie parasismique (French Earthquake Engineering Association)

ANDRA: Agence nationale pour la gestion des déchets radioactifs (French National Radioactive Waste Management Agency)

ANL: Argonne National Laboratory, USA

ANR: Agence nationale pour la recherche (French National Research Agency)

AREVA: French nuclear operator

ARMINES-SPIN: Institut Carnot M.I.N.E.S – Centre des Sciences des processus industriels et naturels de l'École des Mines de Saint-Étienne, France (Carnot M.I.N.E.S Institute – Center for Industrial and Natural Processes of the École des Mines in Saint-Etienne)

ASME: American Society of Mechanical Engineers, USA (commonly used to refer to the design and construction rules drawn up by this American society and used by American designers [Westinghouse, etc.])

ASN: Autorité de sûreté nucléaire (French Nuclear Safety Authority)

BelV: Belgian Federal Agency for Nuclear and Radiological Inspections of nuclear installations

BETCGB: Bureau d'études techniques et de contrôle des grands barrages (French Technical and Inspection Office for Large Dams)

BRGM: Bureau de recherche géologique et minière (French Geological Research Mining Bureau)

C3R: Laboratoire de cinétique chimique, combustion et réactivité (Laboratory of Chemical Kinetics, Combustion and Reactivity is a French joint research laboratory established by IRSN, CNRS and the University of Lille 1 Science and Technology)

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

CEBTP: Centre d'essais du bâtiment et des travaux publics (the French Construction and Public Works Test Center is a center of expertise in soil engineering, materials and structural engineering, and construction)

CEGB: Central Electricity Generating Board, UK

CEMAGREF: Centre national du machinisme agricole, du génie rural, des eaux et des forêts (French National Center for Agricultural Machinery, Rural Engineering, Water and Forestry, research institute)

CERIB: Centre d'études et de recherches de l'industrie du béton (French Concrete Industry Study and Research Center)

CETMEF: Centre d'études techniques maritimes et fluviales (French Technical Maritime and River Study Center)

CNL (formerly AECL): Canadian Nuclear Laboratories

CNPP: Centre national de prévention et de protection (French National Center for Prevention and Protection)

CNR: Compagnie nationale du Rhône (French National Company of the Rhone)

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

CNSC: Canadian Nuclear Safety Commission

CORIA: Complexe de recherche interprofessionnel en aérothermochimie (French Aerothermochemistry Research Complex)

CRL: Chalk River Laboratories, Canada

CSNI: Committee on the Safety of Nuclear Installations, OECD

CSN: Consejo de Seguridad Nuclear (Spanish Nuclear Safety Council)

CSO: Centre de sociologie des organisations (French Center for Sociology of Organizations, a joint research unit of Sciences Po and CNRS)

CSTB: Centre scientifique et technique du bâtiment (French Construction Science and Technology Center)

CTICM: Centre technique industriel de la construction métallique (French Industrial Technology Center for Construction in Metal)

DCNS: French hi-tech company specializing in defense naval systems

DGA: Direction générale de l'armement (Ministère de la Défense) (French General Directorate for Armament, [French Ministry of Defence])

DGPR: Direction générale de la prévention des risques (French General Directorate for Risk Prevention, a government department reporting to the Ministry of Ecology, Sustainable Development and Energy)

DMT: Département de mécanique et thermique (Mechanical and Thermal Engineering Department at CEA's research platform in Saclay, France)

EDF: Électricité de France (French power utility)

ENEL: Ente Nazionale per l'Energia Elettrica, Italian power utility

ENS Cachan: École nationale supérieure de Cachan (a prestigious French higher National School)

ENSI: Eidgenössisches Nuklearsicherheitsinspektorat (Swiss Federal Nuclear Safety Inspectorate)

ENSREG: European Nuclear Safety Regulators Group

EPRI: Electric Power Research Institute, USA

ETIC: Laboratoire d'étude des incendies en milieux confinés (Laboratory for the Study of Fire in a Contained Environment, IRSN, France)

Euratom: European Atomic Energy Community

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

Framatome: nuclear steam supply system manufacturer, France

GDF: Gaz de France (French gas utility)

GeM: Institut de recherche en génie civil et mécanique, unité mixte CNRS/École centrale de Nantes/Université de Nantes (French Civil and Mechanical Engineering Research Institute, a joint CNRS/École Centrale de Nantes/University of Nantes unit)

GFH: Groupe facteurs humains (Human Factors Group, EDF, France)

GrDF: Gaz, réseau, distribution France (French Gas, Network management, Distribution Company)

GRS: Gesellschaft für Anlagen – und Reaktorsicherheit (reactor safety organization, Germany)

HAMMLAB: Halden Man-Machine Laboratory (Norwegian laboratory researching man-machine interactions and control processes)

HSE: Health and Safety Executive, UK

IAEA: International Atomic Energy Agency, Vienna, Austria

IBRAE: Nuclear Safety Institute of the Russian Academy of Sciences, Russia

ICARE: Institut de combustion aérothermique réactivité et environnement (Institute for Combustion, Aerothermal Engineering, Reactivity and Environment, CNRS, Orléans, France)

IFE: Institutt for energiteknikk (Institute for Energy Technology, Norway)

IFREMER: Institut français de recherche pour l'exploitation de la mer (French Research Institute for Exploitation of the Sea)

IFSTTAR: Institut français des sciences et technologies des transports, de l'aménagement et des réseaux (French Institute of Science and Technology for Transport, Development and Networks)

INERIS: Institut national de l'environnement industriel et des risques (French National Institute for the Industrial Environment and Risks)

INL: Idaho National Laboratory, USA

Inria: Institut national de recherche dédié au numérique (French National Research Institute for the Computational Sciences)

INQUA: International Union for Quaternary Research (organization responsible for promoting international collaboration in the earth sciences)

INSA: Institut national des sciences appliquées (French National Institute of Applied Sciences)

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

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

IRSN (formerly IPSN): Institut de radioprotection et de sûreté nucléaire (French Institute for Radiological Protection and Nuclear Safety)

ISTC: International Science and Technology Center

IUSTI: Institut universitaire des systèmes thermiques industriels (French University Institute for Industrial Thermal Systems, a joint research unit of CNRS/Universities of Provence and the Mediterranean)

JAEA: Japan Atomic Energy Agency

JAERI (formerly JAEA): Japan Atomic Energy Research Institute

KAERI: Korea Atomic Energy Research Institute, South Korea

KEPCO: Kansai Electric Power Company, Japan

KfJ: Forschungszentrum Jülich (Jülich Research Center, Germany)

KfK: Kernforchungszentrum Karlsruhe (Karlsruhe Institute of Technology, Germany)

KIT (formerly FzK and KfK): Karlsruhe Institut für Technology (Karlsruhe Institute of Technology, Germany)

KWU: Kraftwerk Union (German nuclear power plant construction company)

Labra: Laboratoire des rayonnements appliqués (Laboratory for Applied Radiation, CEA, France)

LaMCoS: Laboratoire de mécanique des contacts et des structures (Laboratory for the Mechanics of Contacts and Structures, a joint research unit of INSA in Lyon and CNRS, France)

LCPC: Laboratoire central des ponts et chaussées (French Central Laboratory for Bridges and Roads)

LEFH: Laboratoire d'étude des facteurs humains (Laboratory for the Study of Human Factors, IPSN, France)

LEMTA: Laboratoire d'énergétique et de mécanique théorique et appliquée (Laboratory of Energy and Theoretical and Applied Mechanics, jointly run by the University of Lorraine and CNRS, France)

LEPMI: Laboratoire d'électrochimie et de physico-chimie des matériaux et des interfaces (Laboratory of Electrochemistry and Physico-chemistry of Materials and Interfaces in Grenoble, France)

LMA: Laboratoire de mécanique et d'acoustique, (Mechanics and Acoustics Laboratory, France)

LMDC: Laboratoire matériaux et durabilité des constructions (Construction Materials and Durability Laboratory, France)

LNE: Laboratoire national de métrologie et d’essais (French National Laboratory for Metrology and Testing)

LSHS: Laboratoire des sciences humaines et sociales (Human and Social Sciences Laboratory, IRSN, France)

LVEEM: Laboratoire Vellave sur l’élaboration et l’étude des matériaux (Vellave Laboratory on the Development and Study of Materials in Le Puy-en-Velay, France)

MAI: Materials Ageing Institute, France (an international research and development institute set up by EDF, specializing in research into the ageing of materials used in power plants)

MIST: Laboratoire de micromécanique et intégrité des structures, laboratoire “ sans mur ” commun au CNRS et à l’IRSN (Micromechanics and Structural Integrity Laboratory, jointly funded by the CNRS and IRSN, could be described as a laboratory “without walls”, France)

NEA: Nuclear Energy Agency, OECD

NEI: Nuclear Energy Institute, USA

NIIAR: Scientific Research Institute of Atomic Reactors, Russia

NIST: National Institute of Standard and Technology, USA

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

NRA: Nuclear Regulation Authority, Japan

NUGENIA: NUclear GENeration II & III Association (international association dedicated to the safety of generation II and III reactors)

NUPEC: Nuclear Power Engineering Center, Japan

OECD: Organization for Economic Co-operation and Development

NGO: Non-Governmental Organization

ORNL: Oak Ridge National Laboratory, USA

PC2A: Laboratoire de physico-chimie des processus de combustion et de l’atmosphère (Laboratory for the Physical Chemistry of Combustion Processes and the Atmosphere, Lille, France)

PNNL: Pacific Northwest National Laboratory, USA

PROMES: Procédés, matériaux et énergie solaire (Processes, Materials and Solar Energy, laboratory in Perpignan, France)

PSI: Paul Scherrer Institute, Switzerland

RESIF: Réseau sismologique & géodésique français (French Seismologic and Geodesic Network)

ROSATOM: Federal Agency on Atomic Energy, Russia

SHOM: Service hydrographique et océanographique de la marine (French Naval Hydrographic and Oceanographic Service)

SNCF: Société nationale des chemins de fer (French rail operator)

SNL: Sandia National Laboratory, USA

STUK: Radiation and Nuclear Safety Authority, Finland

U.S.NRC: United States Nuclear Regulatory Commission, USA

VTT: Technical Research Center, Finland

WGAMA: Working Group on Analysis and Management of Accidents, OECD/NEA/CSNI

WGELE: Working Group on Electrical Power, OECD/NEA/CSNI

WGEV: Working Group on External Events, OECD/NEA/CSNI

WGFCS: Working Group on Fuel Cycle Safety, OECD/NEA/CSNI

WGFS: Working Group on Fuel Safety, OECD/NEA/CSNI

WGHOF: Working Group on Human and Organizational Factors, OECD/NEA/CSNI

WGIAGE: Working Group on Integrity and Ageing of Components and Structures, OECD/NEA/CSNI

WGRISK: Working Group on Risk Assessment, OECD/NEA/CSNI

Technical glossary

AAR: Alkali-Aggregate Reactions

ABI: name of a program of tests conducted as part of research into reactor vessel failure and basemat erosion due to corium in a reactor core melt accident

ACHILLE: name of an experimental program to study the behavior of fuel rods

ACRR: Annular Core Research Reactor (experimental reactor run by SNL to study core temperature rise and core melt in a reactor vessel)

AGORAS: *Amélioration de la gouvernance des organisations et des réseaux d'acteurs pour la sûreté nucléaire* (Improving Governance of Organizations and Networks involved in Nuclear Safety, research project)

AIC: *Argent-Indium-Cadmium* (Silver-Indium-Cadmium)

ALPHA: name of an experimental facility run by JAERI to study steam explosions in a reactor

ALPS: Advanced Light water reactor Performance and Safety (international research program)

APHRODITE: name of a series of analytical tests to study two-phase thermal hydraulics

ARTEMIS: name of a test program conducted as part of research into reactor vessel failure and basemat erosion due to corium in a reactor core melt accident

ARTIST: Aerosol Trapping in a Steam Generator (experimental facility run by PSI – study of aerosol retention in a steam generator)

ASTEC: Accident Source Term Evaluation Code (system of simulation codes for evaluating the physical phenomena that occur during a core melt accident in a pressurized water reactor)

ASTRID: Advanced Sodium Technological Reactor for Industrial Demonstration (demonstration sodium-cooled fast reactor project)

AZALEE: name of a CEA shake table

BALI: name of a series of tests conducted as part of research into reactor core melt accidents

BALISE: name of a test program conducted as part of research into reactor vessel failure and basemat erosion due to corium in a reactor core melt accident

BETA: name of a KIT experimental facility used to study corium/concrete interaction as part of research into reactor core melt accidents

BETHSY: name of a CEA experimental facility – tests used to check the ability of the CATHARE code to predict satisfactorily the behavior of a nuclear steam supply system in an accident situation

BFC: Bottom of Fissile Column

BILLEAU: name of tests to study steam explosion in a reactor (study of the dispersion of a jet of solid spheres in a test section filled with water)

BIP: Behavior of Iodine Project (international program to study iodine behavior in a reactor containment)

BK: fuel building

BWR: Boiling Water Reactor

CABRI: name of a CEA test reactor used to study accident situations in reactors (PWR, FNR)

CAD: Computer-Aided Design

CADUCEE: name of an experimental apparatus on the GALAXIE platform for research into controlling fire risks in nuclear facilities

CALIST: Characterization and Application of Large and Industrial Spray Transfer (IRSN experimental facility for studying spray mechanisms)

CANON: name of a series of analytical tests to study two-phase thermal hydraulics

CARAIDAS: name of an IRSN experimental facility – study of the behavior of radioactive products in a reactor containment

CAIMAN: name of a CEA experimental facility – study of iodine behavior in a reactor containment

CARINEA: name of an experimental facility on the GALAXIE platform used for research into controlling fire risks in nuclear facilities

Cast3M: name of a finite-element computer code used for the mechanics of structures and fluids

CATHARE: *Code avancé de thermohydraulique pour les accidents de réacteurs à eau* (Advanced Thermohydraulics Code for Water Reactor Accidents, a simulation code used for safety analyses)

CATHODE: name of some analytical test programs conducted as part of research into controlling fire risks in nuclear facilities

CAV: Cumulative Absolute Velocity

CCI: Corium Concrete Interaction

CCM: Cold Crucible Melting (test program to study the fragmentation of molten mixtures)

CCWS: Component Cooling Water System

CEOS.fr: *Comportement et évaluation des ouvrages spéciaux* (Behavior and Evaluation of Special Structures, a French research project)

CESAR: name of a module of the ASTEC code

CIP: CABRI International Program (international program to study the behavior of nuclear fuel rods and their cladding in a reactivity injection accident in PWR)

CIR: Cooperative Irradiation-assisted stress corrosion cracking Research (an international research and development program on the corrosion of stainless steels under stress and under irradiation)

CIVA: name of a numerical simulation platform for non-destructive testing

CFD: Computational Fluid Dynamics

CFR: Code of Federal Regulations

CHIP: *Chimie de l'iode dans le circuit primaire* (Iodine Chemistry in the Reactor Coolant System, an IRSN facility or research program contributing to better evaluation of the quantity of iodine that can be released in a reactor core melt accident)

CHRS: Containment Heat Removal System

CIRCE: name of a research program on the corrosion of nickel-based alloys under stress

CLARA: name of a test program conducted as part of research into reactor vessel failure and basemat erosion due to corium in a reactor core melt accident

CODAZIR: research program to study the fuel rod behavior in a LOCA

COMET: name of an experimental facility and a concept developed by KIT to research reactor core melt accidents

COPAT: *Centre opérationnel de pilotage des arrêts de tranche* (EDF), see OCC

COPO: Corium Pool Facility (Finnish experimental facility – research into the potential for retaining the corium in the reactor vessel in a reactor core melt accident)

CORA: name of a research program on core temperature rise and core melt in the reactor vessel

CORDEB: Corium-Debris (experimental program of research into reactor core melt accidents)

CROCO: name of a code for simulating corium spreading

CSD: *Combustible sévèrement dégradé* (Severely Degraded Fuel)

CSS: Containment Spray System

CVCS: Chemical and Volume Control System

DA: *diagnostic automatique* (automatic diagnostics)

DANAIDES: *Dispositif analytique pour l'étude, en cas d'incendie, du dysfonctionnement électrique par les suies* (Analytical Equipment for Studying Electrical Malfunctions caused by Soot during a Fire, an experimental facility on the GALAXIE platform)

DBE: Design Basis Earthquake (used during the design of nuclear facilities)

DCH: Direct Containment Heating

DDF: *durée de fonctionnement* (Operating Life [EDF project])

DDT: Deflagration–Detonation Transition

DEBORA: name of a series of analytical tests for research into two-phase thermal hydraulics

DEF: Delayed Ettringite Formation

DELTA: name of some analytical test programs run as part of research into controlling fire risks in nuclear facilities

DENOPI: *Dénoyage piscines* (Spent Fuel Pool Water Uncovery, a research program on accidental water uncovery of a nuclear fuel storage pool)

DEVAP: *Dépôt en phase vapeur des produits de fission volatils sur les surfaces des circuits* (Volatile Fission Product Deposits in the Vapour Phase on Reactor System Surfaces, an analytical test program to study the transfer of radioactive products in reactor systems during a core melt accident)

DF: Damage Fuel (a research program to study core temperature rise and core melt in a reactor vessel)

DISCO: DISpersion of simulated COrium (KIT experimental facility to study airborne contamination/corium dispersion, using inert powders)

DIVA: *Dispositif incendie ventilation et aérocontamination* (Fire, Ventilation and Airborne Contamination Device, IRSN experimental facility for conducting fire tests in laboratories and factories or in a pressurized water reactor)

DNB: Departure from Nucleate Boiling

DRACCAR: *Déformation et renoyage d'un assemblage de crayons de combustible pendant un accident de refroidissement* (Deformation and Reflooding of a Fuel Rod Assembly during a Loss-Of-Coolant Accident, simulation code)

DRIVER: name of a KIT experimental facility for studying hydrogen risk

ECO: Experiments on energy COnversion during a steam explosion

ECOBA: *Étude du confinement des ouvrages en béton armé* (Study of Reinforced Concrete Containment Structures, a research project to improve the assessment of containment integrity in pressurized water reactors during a core melt accident)

ECS: Complementary Safety Evaluation

EDGAR: name of a CEA experimental facility – study of fuel rod behavior

ELISA: name of an experimental loop for research into cooling water recirculation under accident conditions

EMAIC: *Émission de l'argent, de l'indium et du cadmium* (Silver, Indium and Cadmium Emission, a series of tests to study the release of products from Ag-In-Cd control rods during accident transients)

EMIS: *Émission de produits de fission* (Fission Product Emission, a code for simulating the release of fission products [a forerunner of ELSA])

ENACCEF: *Enceinte accélération de flamme* (Flame Acceleration Enclosure, an experimental facility run by CNRS/ICARE, Orléans, France)

ENISTAT: Experimental and Numerical Investigation of Shear wall reinforced concrete buildings under Torsional effects using Advanced Techniques (a European program of experimental and numerical studies)

EPICUR: *Études physico-chimiques de l'iode confiné sous rayonnement* (Physical and Chemical Studies of Contained Iodine under Radiation, an IRSN facility or research program to validate models of iodine chemistry in the containment of a pressurized water reactor in an accident situation)

EPR: European Pressurized Water Reactor

ERCOSAM-SAMARA: Containment thermal-hydraulics of current and future LWRs for Severe Accident Management (a research program on hydrogen risk)

ESSOR: *Essai orgel* (ORGEL Reactor Test, research reactor at the Ispra Joint Research Centre, Italy)

EVA: *étude du vieillissement des aciers* (Study of the Ageing of Steel)

FABIME: name of a series of tests on thermal fatigue and their test equipment

FALCON: name of a program of analytical tests to study the transfer of radioactive products in reactor systems during a core melt accident

FARO: name of a test program to study interactions between fuel and coolant (steam explosions)

FAT3D: name of a series of tests on thermal fatigue and their test equipment

FDS: Fire Dynamics Simulator

FEBA: Flooding Experiments with Blocked Arrays (experimental program to study fuel rod behavior in a loss-of-coolant accident)

FGD: Fission Gas Dynamics (analytical tests to study reactivity accidents)

FITS: Fully Instrumented Test Series (Sandia National Laboratories experimental facility used to study steam explosion in a reactor)

FLECHT-SEASET: Full-Length Emergency Core Cooling Heat Transfer–Separate Effects tests And System-Effects Tests (experimental program – study of fuel rod behavior in a loss-of-coolant accident)

FLHT: Full Length High Temperature (research program on core temperature rise and core melt in a reactor vessel)

FLIP: *Feux de liquide en interaction avec une paroi* (Interaction of Liquid Fires with a Wall, research program on solvent fires for fuel reprocessing plant safety)

FLUENT: name of a numerical fluid mechanics code

FNR: Fast Neutron Reactor

FP: Fission Products

FP: Framework Program, European Commission

FPCPS: Fuel Pool Cooling and Purification System

FPT: Fission Product Test (acronym associated with the tests run as part of the Phébus-PF program)

GALAXIE: name of an IRSN platform of experimental facilities for research into controlling fire risks in nuclear facilities

GONDOLE: name of a research program – swelling of steel under irradiation

GMR: Giant Magnetoresistance

GPS: Global Positioning System

H2-PAR: Hydrogen Passive Autocatalytic Recombiners (test program)

HEPA: High Efficiency Particulate Air

HEVA: *Hélium, vapeur* (Helium, Steam – experimental program)

HI: Horizontal Induction (tests as part of a study of fission product emission by fuel when its temperature rises)

HOF: Human and Organizational Factors

HRO: High Reliability Organizations

HRP: HALDEN Reactor Project (research aimed at improving nuclear power plant safety)

HT: High Temperature

HYCOM: integral large scale experiments on HYdrogen COMbustion for severe accident code validation (European project)

HYDRA: name of an experimental device on the GALAXIE platform for research into controlling fire risks in nuclear facilities

HYDRAZIR: research program to study fuel rod behavior in a loss-of-coolant accident

ICE: *Interaction corium-eau* (Corium-Water Interaction, experimental program)

IGR: Impulse Graphite Reactor (Russian research reactor – study of reactivity accidents)

IMPACT: name of an experimental program as part of research and development on the behavior of engineered structures subject to an impact

INB: *installation nucléaire de base* (Basic Nuclear Installation)

INCEFA: INcreasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment, European project

INSAG: International Nuclear Safety Group

InSa: Interferometric Synthetic aperture radar (technique used in geodesy and remote sensing)

IRIS 2010: Improving Robustness assessment of structures Impacted by missileS (international benchmarking on the behavior of engineered structures subject to an impact)

IRMA: *Irradiation matériaux* (IRradiation of MAterials, a CEA experimental facility for studying radiation/material interaction mechanisms)

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

ISAFES: Interactive Seismic Analysis of Fragilities of Equipment and Structures

ISIS: name of a software system for analysing the ventilation of a fire and airborne contamination

ISP: International Standard Problem

ISR: *ingénieur de sûreté-radioprotection* (Safety and Radiation Protection Engineer)

ISTP: International Source Term Program

ITER: International Thermonuclear Experimental Reactor

IVANA: name of a VUEZ experimental facility used for research into cooling water recirculation in accident conditions

IVMR: In-Vessel Melt Retention

IVR: In-Vessel Retention

KALI H2: name of a CEA experimental facility for studying hydrogen risk

KARISMA: KAshiwazaki-Kariwa Research Initiative for Seismic Margin Assessment (international benchmarking carried out following an earthquake affecting the Kashiwasaki Kariwa nuclear power plant in Japan)

KMS: Russian experimental facility – hydrogen risk

KONVOI: name of a pressurized water reactor of German design

KROTOS: name of a CEA experimental facility for studying the interactions between a molten mixture and a coolant (steam explosions)

LDV: Laser Doppler Velocimetry

LII: Laser Induced Incandescence

LOBI: name of an experimental facility at the Ispra Joint Research Centre, used for studying the thermohydraulic behavior of a nuclear reactor in an accident situation

LOCA: Loss-of-Coolant Accident

LOFT-FP: Loss Of Fluid Tests–Fission Product (research project on core temperature rise and core melt in a reactor vessel due to a loss of coolant)

LS-DYNA: name of a rapid dynamic simulation code for studying structures

LSTF: Large Scale Test Facility (Japanese experimental facility – study of the thermohydraulic behavior of a nuclear reactor in an accident situation)

MAAP: Modular Accident Analysis Program (software [or software system] for the simulation of physical phenomena that occur during a core melt accident in a pressurized water reactor)

MACE: Melt Attack and Coolability Experiments (test program on corium-concrete interaction as part of research into reactor core melt accidents)

MAEVA: *Maquette échange vapeur air* (Steam Air Exchange Model, EDF experimental model for studying on a large scale the behavior of a containment in a loss-of-coolant accident)

MAFFé: name of an oven on the EDF experimental platform used for research into controlling fire risks in nuclear facilities

MANON: name of an experimental loop used for research into cooling water recirculation in accident conditions

MARC: *Masse, amortisseur, raideur, critère* (Weight, Damper, Stiffness, Criterion)

MASCA: name of a test program conducted as part of research into reactor core melt accidents

MC3D: name of a 3D multi-phase thermal hydraulics code for simulating the interaction between molten materials and a coolant

MCCI: Molten Core-Concrete Interaction

MELCOR: name of a software (or software system) for simulating the physical phenomena that occur during a core melt accident in a pressurized water reactor

MFPR: Module for Fission Product Release (code for simulating fission product behavior)

MILONGA: name of an experimental platform developed by EDF for research into controlling fire risks in nuclear facilities

MIRE: *Mitigation des rejets à l'environnement en cas d'accident nucléaire* (Mitigation of Releases into the Environment in the Event of a Nuclear Accident, research program to study and improve the mitigation of radioactive releases in the event of a reactor core melt accident)

MISS3D: *Modélisation de l'interaction sol-structure en trois dimensions* (3D Modelling of Ground-Structure Interaction, a simulation software)

MISTRA: name of a CEA experimental facility – studies of hydrogen risk

MITHYGENE: *Mitigation hydrogène* (Hydrogen Mitigation, improving knowledge of hydrogen risk and how to manage it in a reactor core melt accident)

MIT3BAR: *Évaluation et mitigation du risque de percement de la troisième barrière de confinement des centrales nucléaires* (Assessment and Mitigation of the Risk of Third Containment Barrier Breakthrough at Nuclear Power Plants)

MOCKA: name of a KIT experimental facility – study of corium-concrete interaction in a reactor core melt accident

MOBY DICK: name of a series of analytical tests conducted as part of the study of two-phase thermal hydraulics

MOX: Mixed Oxide Fuel ($UO_2 + PuO_2$)

MOZART: *Mesure de l'oxydation du zirconium par l'air en température* (Measurement of Zirconium Oxidation by Air at Temperature, a program of analytical tests to study the oxidation of fuel rod cladding in the presence of air)

MPE: Maximum Probable Earthquake

MRBT: Multi-Rod Burst Test (ORNL experimental facility – study of fuel rod behavior)

NORS: NOkia Research Simulator

NPPs: Nuclear Power Plants

NUREG: Nuclear Regulatory Report, U.S.NRC

NUSMOR: NUgenia Small Modular Reactor with passive safety features, European research project

NRU: National Research Universal (Canadian research reactor)

NSRR: Nuclear Safety Research Reactor, Japan

NYX: name of an experimental device on the GALAXIE platform, used for research into controlling fire risks in nuclear facilities

OCC: Outage Control Centers

ODOBA: *Observatoire de durabilité des ouvrages en béton armé* (Monitoring Center for the Durability of Reinforced Concrete Structures, a research project to study the ageing of structures and the pathologies affecting them)

OLHF: OECD Lower Head Failure (research program to study failure of the lower part of a reactor vessel)

OMEGA: name of a series of analytical tests conducted as part of studies of two-phase thermal hydraulics

OSIRIS: name of a CEA research reactor

PACTEL: Parallel Channel Test Loop (Finnish experimental facility for studying the thermohydraulic behavior of a nuclear reactor in an accident situation)

PANDA: name of a PSI experimental facility used for research into hydrogen risk

PARIS: name of an experimental program conducted as part of the study of radioactive iodine behavior in a reactor core melt accident

PASSAM: Passive and Active Systems on Severe Accident source term Mitigation (multi-partner research project on passive and active systems to mitigate releases in a reactor core melt accident)

PATRICIA: name of a research program on reactivity accidents

PBF: Power Burst Facility, INL, USA

PCCV: Prestressed Concrete Containment Vessel (SNL experimental model – assessment of leaks of air and steam through cracks in conditions representative of a reactor containment)

PCMI: Pellet-Cladding Mechanical Interaction

PEARL: name of an IRSN experimental facility for testing debris bed reflooding

PERFECT: name of a European research program – corrosion of stainless steel under stress and under irradiation

PERFORM60: Prediction of the Effects of Radiation FOR Pressure Vessel and in-core Materials using multi-scale Modelling – 60 years foreseen plant lifetime (irradiation European research and development project on metallic components subject to irradiation)

PERFROI: *Étude de la perte de refroidissement* (Loss-of-Coolant Study, an experimental project aimed at filling the gaps in knowledge of reactor core cooling in a loss-of-coolant accident)

PERICLES: name of a series of analytical tests conducted as part of a study of two-phase thermal hydraulics

PGA: Peak Ground Acceleration

PHEBUS: name of a CEA experimental reactor

Phebus-CSD: international research program to study severe fuel degradation, using tests carried out in the PHEBUS reactor

Phebus-FP: international research program to study the behavior of fission products, using tests carried out in the PHEBUS reactor

PHENIX: name of a CEA prototype nuclear power reactor (and experimental reactor), a fast neutron reactor using liquid sodium as a coolant

PICSEL: *Propagation de l'incendie de combustibles solides dans un environnement laboratoires et usines* (Propagation of Solid Fuel Fires in Laboratories and Factories, research programs into fires in electrical cabinets, looking at safety in fuel reprocessing plants)

PIA: *plan d'investissement d'avenir* (Investment in the Future Program)

PIV: Particle Image Velocimetry

PKL: Primärkreislauf (reactor coolant system, German research projects and large-scale test facility for studying the thermohydraulic behavior of a nuclear reactor in an accident situation)

PLUTON: name of an experimental facility on the GALAXIE platform used for research into controlling fire risks in nuclear facilities

PRELUDE: *Préliminaire sur le renoyage expérimental d'un lit de débris* (Preliminary Study on the Experimental Reflooding of a Debris Bed, an IRSN experimental facility for conducting feasibility studies and qualification studies of the instrumentation used by the PEARL program on reflooding debris beds)

P2REMICS: name of a simulation code for studying hydrogen risk

PREMIX: name of a KIT experimental facility for studying steam explosion in a reactor

PRENOLIN: *Amélioration de la prédiction des effets non linéaires induits par les mouvements sismiques forts* (Better Prediction of Non-Linear Effects Induced by Strong Seismic Motion, benchmarking process)

PRISME: *Propagation d'un incendie pour des scénarios multi locaux élémentaires* (Spread of a Fire for Multi-Room Elementary Scenarios, international research program)

PROGRES: *Program expérimental analytique sur le renoyage de lits de débris* (Analytical Experimental Program on Debris Bed Reflooding)

PROMETRA: *Propriétés mécaniques en transitoire* (Mechanical Properties in a Transient, research program to study the mechanical behavior of fuel cladding in a reactivity accident)

PSB-VVER: Russian experimental facility used to study the thermohydraulic behavior of a nuclear reactor in an accident situation

PSA: Probabilistic Safety Assessments

PTR: *système de traitement et refroidissement de l'eau des piscines* (FPCPS water tank)

PWR: Pressurized Water Reactor

QUENCH: name of a KIT experimental facility used for integral core reflooding tests

RADIOSS: name of a fast dynamic code for analysing structures

RASPLAV: name of a test program run as part of research into reactor core melt accidents

RB: Reactor Building

RCC-G: *règles de conception et de construction des ouvrages de génie civil* (Rules on the Design and Construction of Civil Engineering Structures)

RCC-M: *règles de conception et de construction des matériels métalliques* (Rules on the Design and Construction of Metallic Equipment)

RCCV: Reinforced Concrete Containment Vessel (SNL experimental model – assessment of leaks of air and steam through cracks in conditions representative of a reactor containment)

RCS: Reactor Coolant system

REBEKA: name of a KfK experimental facility (Germany) – study of fuel rod behavior in a LOCA

RECI: *Recombineur et iode* (Recombiner and Iodine, tests to quantify experimentally the conversion rate of metal iodides into iodine on the basis of temperature)

REKO: name of a KfJ experimental facility – study of hydrogen risk

REPASS: Reliability Evaluation of Passive Safety Systems (European research project)

RESOH: *Recherche en sûreté, organisation et hommes* (Research on Safety, Organization and Humans, a chair devoted to safety management in hazardous industries, particularly nuclear)

RG: Regulatory guide, USA

RHRS: Residual Heat Removal System

RIA: Reactivity Injection Accident

RJH: *réacteur Jules Horowitz* (Jules Horowitz Reactor, France)

ROSA: Rig Of Safety Assessment (research projects to study the thermohydraulic behavior of a nuclear reactor in an accident situation)

ROSCO: name of a series of analytical tests to study two-phase thermal hydraulics

RSE-M: *règles de surveillance en exploitation des matériels mécaniques* (Rules for the Monitoring of Mechanical Equipment in Operation)

RSNR: *Recherche en matière de sûreté nucléaire et de radioprotection* (Research Projects in the Field of Nuclear Safety and Radiation Protection)

RT: Release of Transuranics

RTF: Radioiodine Test Facility (AECL experimental facility – study of radioactive iodine behavior in a reactor containment during a core melt accident)

RUT: name of a Russian experimental facility – study of hydrogen risk

RWST: Refueling Water Storage Tank

SAFEST: Severe Accident Facilities for European Safety Targets

SARNET: Severe Accident Research NETwork of excellence (international network of excellence on core melt accidents)

SATURNE: name of an experimental facility on the GALAXIE platform used for research into controlling fire risks in nuclear facilities

SCANAIR: *Système de codes pour l'analyse d'accidents d'insertion de réactivité* (System of Simulation Software for Analysing Reactivity Injection Accidents)

SCARABEE: name of a CEA research reactor (used for studying accident situations in FNRs)

SEFLEX: (Fuel Rod) Simulator Effects in Flooding Experiments (experimental program to study fuel rod behavior in a LOCA)

SERENA: Steam Explosion REsolution for Nuclear Applications (research program to study and develop simulation tools for steam explosion)

SETH: SESAR THermalhydraulics (international research program on hydrogen risk)

SFD: Severe Fuel Damage (experimental program on core temperature rise and core melt inside a reactor vessel)

SFP: Spent Fuel Pool

SFR: Sodium-cooled Fast neutron Reactors

SG: Steam Generator

SIGMA: Seismic Ground Motion Assessment (research and development program)

SILOE: name of a CEA research reactor

SIMIBE: name of tests conducted on leaks through a crack in a reactor containment

SINAPS@: *Séisme et installations nucléaires, améliorer et pérenniser la sûreté* (Earthquakes and Nuclear Facilities, Improving and Protecting Safety, a research project on seismic hazards and the vulnerability of nuclear components and structures)

SIROCCO: name of an oven on the GALAXIE platform, used for characterizing malfunctions of electrical equipment subjected to thermal stress

SIS: Safety Injection System

SMA: Seismic Margins Assessment

SMART: name of an instrumented CEA model used for assessing seismic motion transferred to equipment

SMD: SUPER MOBY DICK (name of a series of analytical tests conducted as part of the study of two-phase thermal hydraulics)

SME: Seismic Margin Earthquake

SMR: Small Modular Reactors

SNETP: Sustainable Nuclear Energy Technology Platform

SOA: State-Oriented Approach

SOAR: State-of-the-Art-Report, OECD/NEA

SOFIA: *Simulateur d'observation du fonctionnement incidentel et accidentel* (Observation Simulator for Incidental and Accidental Operation, an engineering simulator to improve the nuclear safety of pressurized water reactors)

SPARK: software for simulating the operation of passive autocatalytic recombiners

SPERT: Special Power Excursion Reactor Tests (American research reactor used for studying reactivity accidents)

SPLASH: name of a series of tests on thermal fatigue and their test equipment

SPOT: name of a Russian experimental facility – study of hydrogen risk

SSI: Soil-Structure Interaction

SSG: Specific Safety Guide, IAEA

SSWICS: Small Scale Water Ingression and Crust Strength (ANL experimental platform for research into reactor vessel failure and basemat erosion due to corium)

STARMANIA: name of an IRSN experimental facility for conducting research into controlling fire risk in nuclear facilities

STEM: Source Term Evaluation and Mitigation (research program on the behavior of radioactive products likely to be released into the environment during a core melt accident)

STORM: name of an Ispra Joint Research Centre experimental facility used for studying the transfer of radioactive products into reactor systems in a core melt accident

STL: *sonde tournante longue* (Long Rotating Probe)

STT: *sonde tournante transversale* (Transverse Rotating Probe)

STYX: name of an experimental facility on the GALAXIE platform used for research into controlling fire risks in nuclear facilities

SUPERCANON: name of a series of analytical tests conducted as part of the study of two-phase thermal hydraulics

SUPERPHENIX: name of an EDF nuclear power reactor, a fast neutron reactor using liquid sodium as a coolant

SUW: Scale-Urania-Water (Winfrith experimental facility – study of steam explosion in a reactor)

SYLVIA: *Système de logiciels de simulation pour l'étude de la ventilation, de l'incendie et de l'aérocontamination* (Software System for Analysing the Ventilation of a Fire and Airborne Contamination)

TAGCIS: *Trempe en APRP de gaine de combustible à irradiation simulée* (Quench during a LOCA of fuel rod cladding that has undergone simulated irradiation, a research program)

TAGCIR: *Trempe en APRP de gaine de combustible irradiée* (Quench during a LOCA of irradiated fuel rod cladding, a research program)

TAMARIS: *Tables et moyens d'analyses des risques sismiques* (Tables and Means of Analysing Seismic Risks, a CEA experimental platform)

TANDEM: Tsunami in the Atlantic and the English ChaNnel: Definition of the Effects through numerical Modeling

THAI: Thermal-hydraulics, Hydrogen, Aerosols and Iodine (experimental installation of Becker Technologies)

THETIS: name of an experimental program to study fuel rod behavior

THINS: Thermal-hydraulics of Innovative Nuclear Systems (European research project)

TIB: Total Instantaneous Blockage

TMI: Three Mile Island, USA

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

TOFD: Time Of Flight Diffraction

TONUS: name of a simulation code for assessing hydrogen risk in core melt accident conditions

TORPEDO: name of a KIT experimental facility – study of hydrogen risk

TOSQAN: name of an IRSN experimental facility used for simulating the thermohydraulic conditions in a nuclear reactor containment during a core melt accident

TRANSAT: name of a program of analytical tests conducted as part of research into the transfer of radioactive products into reactor systems during a core melt accident

TREPAM: name of an experimental program to study corium-concrete interaction

TRIGA: Training, Research, Isotopes, General Atomics (pool-type research reactor)

TROI: Test for Real cOrium Interaction with water (KAERI experimental facility used to study the interaction between fuel and coolant)

TSO: Technical Safety Organization

TUBA: name of a program of analytical tests to study the transfer of radioactive products into reactor systems during a core melt accident

ULPU: An IVR-related full-scale boiling heat transfer facility at University of California, Santa Barbara (a UCSB experimental facility for research on the possibility of retaining the corium in the reactor vessel during a reactor core melt accident)

UMR: *unité mixte de recherche* (Joint Research Unit)

UNGG: *uranium naturel-graphite-gaz* (Natural Uranium-Graphite-Gas, gas-cooled reactor type)

VD: *visite décennale* (Ten-Yearly Outage Program)

VEGA: Verification Experiments of radionuclides Gas/Aerosol release (tests to study the emission of fission products by fuel when its temperature rises)

VERCORS: CEA facility in Grenoble, used to study the release of fission products by irradiated fuel subjected to a temperature increase

VERCORS: *Vérification réaliste du confinement des réacteurs* (Realistic Verification of the Containment of Reactors, an EDF model and a series of tests to assess leaks through a containment in an accident situation)

VERDON: name of the CEA experimental facility that replaced VERCORS in Grenoble

VI: Vertical Induction (tests to study the emission of fission products by fuel when its temperature rises)

VIKTORIA: name of an experimental loop for conducting research into cooling water recirculation in accident conditions

VITRA: name of an experimental loop for conducting research into cooling water recirculation in accident conditions

VULCANO: Versatile UO_2 Laboratory for COrium ANalysis and Observation (CEA experimental facility for research into reactor vessel failure and basemat erosion due to corium in a reactor core melt accident)

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

VWU: VULCANO Water-Uranium

WUMT: tests to study steam explosion in a reactor

XRD: X-Ray Diffraction

Foreword

This summary on the current state of research in the field of pressurized water reactor safety is a collective effort by a team of authors from IRSN, the French national Institute for Radiological Protection and Nuclear Safety. Jean Couturier and Michel Schwarz (retired from IRSN) are the main authors.

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Chapter 1
Introduction

Preventing accidents liable to affect a nuclear facility and, in keeping with a defense-in-depth approach, mitigating their impact, call for robust methods based on the state of the art, operating experience feedback and regulations, and the demonstration by the licensee that the provisions and measures it implements are effective. This demonstration and its assessment, carried out in France by IRSN as part of its assessment activities[1], draw on a body of scientific knowledge that has grown considerably after nearly forty years of research and the related development work, particularly in the field of simulation codes.

This publication describes the current state of this research and development conducted by IRSN, alone or in collaboration with other organizations, the lessons learned and how they have contributed to safety at French nuclear power reactors and, lastly, the work that is still underway or planned to consolidate and push back the boundaries of knowledge. This summary does not claim to be exhaustive, and some R&D work is only quoted or mentioned briefly at the end of this publication. Also, some of the research discussed here, such as the work on seismic, flooding and fire hazards, is of benefit to all nuclear facilities, not only to nuclear power reactors.

Nuclear reactors are large, complex machines. Many complex physical phenomena interact as an accident unfolds. Studying these phenomena requires knowledge in many fields of physics, including neutronics, heat transfers, fluid mechanics, structural mechanics, metallurgy, radiation chemistry, etc. That being the case, it is clear that the results of experimental research, which is generally carried out at a reduced scale and is simplified to varying degrees, cannot be used directly for demonstration purposes. Physical models

1. Defined in Decrees No. 2002-254 of February 22, 2002 and No. 2016-283 of March 10, 2016.

must therefore be developed, based on academic knowledge or targeted experiments, also called separate effects experiments (see the Focus at the end of this introduction). These models must then be integrated in computer codes used to simulate all the phenomena considered essential for a clear understanding of how the accident unfolds at reactor scale.

More complex experiments simulating real conditions as closely as possible, but carried out at an intermediate scale, also known as integral experiments (see the Focus at the end of this introduction) are conducted to assess the relevance of the calculations performed using these simulation codes. In some cases, the experiments highlight areas where further research is required to supplement and deepen knowledge. Accidents such as those that occurred in reactor 2 of the Three Mile Island nuclear power plant in the United States on March 28, 1979, and in the reactors of the Fukushima Daiichi nuclear power plant in Japan on March 11, 2011, have also provided unique data. Despite gaps due to lack of instrumentation, this data nonetheless helps to assess the performance of simulation codes in terms of their ability to predict how real accidents unfold. The accident that occurred in reactor 4 of the Chernobyl nuclear power plant on April 26, 1986 raised questions as to the suitability of models – and therefore the ability of computer codes – for predicting radioactive release from western nuclear power plants in the event of core melt, as well as the validity of the safety criteria adopted until then for reactivity-initiated accidents.

In spite of the considerable progress made over the past forty years in understanding the phenomena involved, more in-depth knowledge is still required in some areas relating to nuclear safety because of changes in reactor operating conditions, the use of innovative technology, and the search for more accurate scientific approaches, all in an environment of increasingly fierce industrial competition. Assumptions thought to be prudent have often been made at the design stage, sometimes making it necessary to check and determine how conservative they are and what margins they really allow.

Furthermore, the possible occurrence of core melt was taken into account right from the design stage of new-generation reactors, like the European Pressurized water Reactor (EPR), which was not the case for reactors commissioned during the 20th century, even if they did subsequently benefit from improvements aimed at taking this risk into consideration. Consequently, extending the operating life of these reactors beyond forty years cannot be contemplated without a significant improvement in safety, in terms of preventing core melt accidents and mitigating their impact. This heightens the need for research in this area. The complementary safety evaluation (ECS) carried out in the wake of the accident at the Fukushima Daiichi nuclear power plant underscored the need for further research on external hazards that could potentially lead to core melt accidents, the risk and consequences of hydrogen explosions, and radioactive release filtering.

In France, the main organizations involved in nuclear safety research are IRSN, the French Alternative Energies and Atomic Energy Commission (CEA), EDF – French power utility – and, to a lesser degree, AREVA (Framatome before 2006), the designer. Academic research organizations, such as universities, engineering schools and CNRS[2], the

2. French National Center for Scientific Research.

French National Center for Scientific Research, are also involved in promoting a fuller understanding of basic phenomena.

While it is for the licensee to provide the French Nuclear Safety Authority (ASN) with the technical bases used in its safety demonstration, including the related research results, it is essential for IRSN to also carry out its own research. It is this work that allows it to develop and maintain its expertise in complex scientific themes that must be perfectly mastered in order to perform relevant, independent assessments of the licensee's safety demonstrations, and make a positive contribution to advancing safety. IRSN conducts experimental research programs in its laboratories, or has them carried out at partner laboratories (CEA's in particular), benefits from partners' work and, whenever possible, develops its own models and simulation codes. It validates the simulation codes (or has them validated) that are vital to the research work and studies carried out in support of its assessment activities.

The knowledge thus acquired and the developed and validated simulation codes also enable IRSN to provide the public authorities with scientific and technical support in the event of emergency. It was thus able to provide the authorities, general public and media with precise updates on events at the Fukushima Daiichi nuclear power plant following the accident.

Some research allows IRSN to clarify or consolidate its technical examination of topics that could help to significantly enhance safety, in nondestructive testing of reactor systems and components for example.

One specific feature of nuclear safety research is the period of time required before results can be used for assessment purposes. In general, this is relatively long, about ten years, for example, for research involving the use of a nuclear reactor, or requiring the design of a technologically innovative experimental setup. This is due to the complexity of the experimental facilities to be designed and implemented, particularly those that must be installed in nuclear research reactors, and to the time required for post-mortem (or post-test) examinations at specialized laboratories, especially when testing involves the use of radioactive material. It is therefore important to plan ahead to ensure that the required knowledge and simulation codes are available when needed. For licensees and IRSN experts alike, such forward planning is crucial if they are to be ready for major milestones such as reactor safety reviews during periodic outage programs, or examinations of safety conditions for extending the operating life of reactors beyond forty years (EDF's DDF project on "operating life").

With regard to its cost, a great deal of this research – especially experimental programs – is carried out under cooperation agreements between industry and IRSN and its foreign counterparts, with each partner being free to use the results of the research for its own purposes.

IRSN is a partner in more than a dozen projects selected by the French National Research Agency (ANR), notably in the framework of the call for projects "Nuclear Safety and Radiation Protection Research" launched in the wake of the Fukushima Daiichi accident. But the cooperative research is not limited to France. It will be seen later that the main countries that designed and built nuclear power reactors – the United States, Canada, Japan, Germany, the United Kingdom, Switzerland and Russia – carried out research programs on nuclear safety and continue to do so. IRSN receives funding from international partners under various bilateral and multilateral agreements for most of its research programs and, in exchange, has access to the results of foreign programs to which it makes a financial contribution.

The Organization for Economic Co-operation and Development/Nuclear Energy Agency[3] (OECD/NEA) plays a very important role by assisting "*its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes*". It works to achieve a consensus among experts on the current state of knowledge (e.g. State-Of-the-Art Reports [SOAR]), gaps to be filled, and future research priorities. It organizes international benchmark exercises to compare various simulation codes with experimental results (International Standard Problems [ISP]). These exercises always provide a great deal of information. It also helps to structure international research projects – referred to as Joint Projects, proposed by one of its members – by bringing together other partner members to obtain the required funding, and thus advance knowledge in areas where it considers this is necessary. With regard to OECD/NEA activities, IRSN is particularly involved in the work of the Committee on the Safety of Nuclear Installations (CSNI), whose mission is to assist member countries in maintaining and further developing the scientific and technical knowledge base required to assess the safety of nuclear reactors and fuel cycle facilities and, more directly in its various working groups:

- Working Group on Risk Assessment (WGRISK),
- Working Group on Fuel Safety (WGFS),
- Working Group on Analysis and Management of Accidents (WGAMA),
- Working Group on Integrity and Ageing of Components and Structures (WGIAGE),
- Working Group on Human and Organizational Factors (WGHOF),
- Working Group on Fuel Cycle Safety (WGFCS),
- Working Group on External Events (WGEV),
- Working Group on Electrical Power Systems (WGELE).

The European Commission also makes a substantial contribution to the funding of international research and development projects relating to nuclear safety. Calls for projects on what are considered as priority themes are issued in the Euratom part of the multi-year research and development Framework Programs (FP), which began in 1984. These are carried out within a cooperative framework, and generally involve industrial partners, nuclear reactor operators, assessment organizations, and research laboratories.

3. As of January 1, 2015, NEA membership included 31 countries in Europe, North America and the Asia-Pacific region.

It should also be noted that IRSN is involved in European bodies whose task is to steer prestandardization research in various fields (metal structural mechanics, civil engineering, instrumentation and control software).

The European Sustainable Nuclear Energy Technology Platform (SNETP), which brings together representatives from the nuclear industry, research, safety organizations, associations, and NGOs[4], was set up in 2007 to formulate a collective vision of the contribution that nuclear fission could make towards the transition to a low-carbon energy mix by 2050. Within this context, the SNETP has published various documents, including a "Strategic Research Agenda" (January 2009) and a "Strategic Research and Innovation Agenda" (February 2013), which addresses a number of safety issues relating to Generation II and III water reactors, as well as a special document entitled "Identification of Research Areas in Response to the Fukushima Accident" (January 2013). These documents are used to help the European Commission to define its research and development Framework Programs. Note that in 2012, SNETP stakeholders (except for the NGOs) working on Generation II and III reactors joined with other existing organizations or networks (including SARNET (Severe Accident Research NETwork of excellence) for core melt accidents, NULIFE, and ETSON), to form NUGENIA (NUclear GENeration II & III Association), an association under Belgian law. Two publications of this association are worthy of mention: "NUGENIA Roadmap – Challenges & Priorities" (October 2013) and "NUGENIA Global Vision" (April 2015). IRSN is closely involved in both SNETP and NUGENIA.

There now follows an overview of the main safety-related research programs, in particular those carried out by IPSN then IRSN, as well as an outline of the lessons learned from results. The main foreign research programs in the same field are also presented, although by no means exhaustively.

The first chapter of this publication is devoted to research on the design-basis accident taken into account for engineered safety equipment used in pressurized water reactors (PWR), namely a main pipe break in the reactor coolant system (loss-of-coolant accident [LOCA]). The first research programs on nuclear safety, which started in the 1970s, focused on this area with the aim of improving knowledge and developing computer tools for studying this accident.

For each of the research and development themes discussed in this publication – positioned in the "deterministic" safety demonstration in Figure 1.1 on the next page – the safety objectives of the safety demonstration are indicated, together with the issues or difficulties that the demonstration might have raised, or still raises today given the current state of knowledge.

First, a brief description is provided of some of the research instruments favored by IRSN in the field of nuclear power reactor safety and which will be mentioned later in the publication.

As a general rule, it was decided to refer only to a few documents regarded as the most significant, or as having the advantage of providing an overview of the knowledge acquired in some of the areas of interest (e.g. some OECD State-Of-the-Art Reports), and which generally include references to many scientific documents.

4. Non-Governmental Organization.

► Research carried out as part of strategic guidelines and programs

The research themes and topics referred to in this publication concern strategic guidelines and scientific programs aimed at ensuring that first IPSN, then IRSN, has had access to the best available knowledge for their assessments and other activities. These scientific guidelines and programs have, of course, been determined by issues raised during safety analysis, by changes in reactor design, lesson learned from operating experience feedback (from incidents in particular), the accidents that have occurred (TMI, Chernobyl and, more recently Fukushima Daiichi), as well as research results. IRSN's scientific strategy can now be consulted by the public. Pressurized water reactor safety is directly addressed in reference [1], published in October 2015, which considers it in terms of seven "priority scientific issues", including one concerning cross-cutting human and organizational aspects, corresponding to the themes and topics discussed in the present publication.

#FOCUS

Separate-effects or Analytical Experiments *versus* Integral Experiments

Separate-effects experiments are very well-instrumented experiments used for the detailed study of a particular physical phenomenon to precisely determine the physical laws that govern it. The "experimental designs" allow systematic variation of the important physical parameters. For example, the oxidation kinetics of fuel rod cladding is determined by placing a cladding sample in a test section with steam flowing through it inside a controlled temperature oven. The sample is hung from scales and the required information is obtained by measuring the increase in its weight at different temperatures over time.

Integral experiments, on the other hand, are more concerned with reproducing the complexity of the phenomena under study, even if this often means that the measurement of the physical variables is broader and less precise. For example, the damage to a fuel assembly following loss of cooling was studied by irradiating a bundle of 20 fuel rods in the PHEBUS reactor, in a heat-insulated test section with steam flowing through it. The physical phenomena involved are: nuclear fission reactions that cause the fuel to overheat; cladding oxidation beyond a certain temperature, which has the result of consuming steam, releasing hydrogen, and generating heat; and heat transfers in the bundle through conduction, convection and radiation. The phenomena can only be studied by measuring the reactor power, the flow of steam passing through the test section, local temperatures in the fuel and the test section, as well as total hydrogen production.

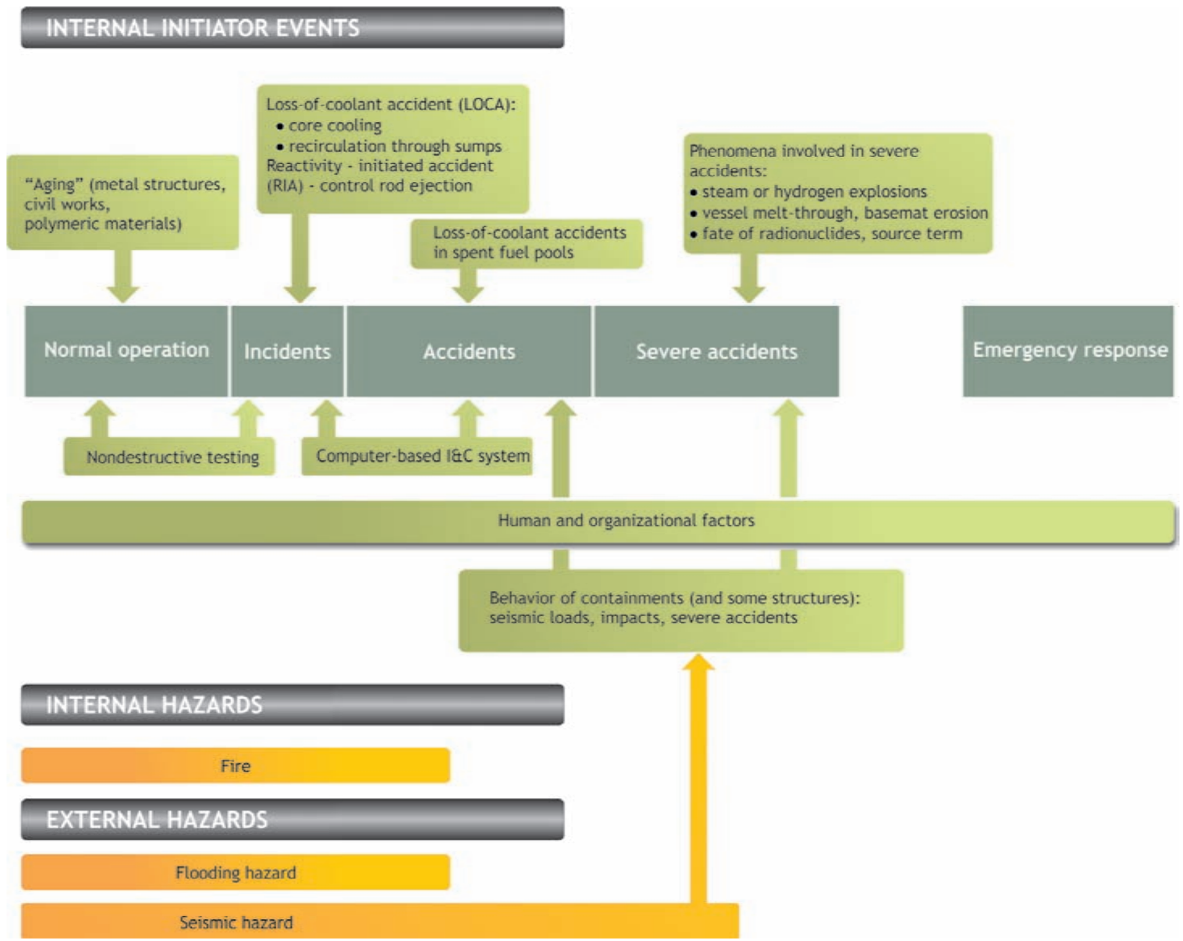

Figure 1.1 Research and development themes discussed (or simply mentioned) in this publication. © Georges Goué/IRSN.

Reference

[1] *La stratégie scientifique de l'IRSN*, October 2015: http://www.irsn.fr/FR/IRSN/presentation/Documents/IRSN_Strategie-scientifique_2015.pdf

Chapter 2
Some of the Research Facilities Favored by IRSN in the Field of Nuclear Power Reactor Safety

2.1. The CABRI reactor

The CABRI safety test reactor, together with SCARABEE, its former twin reactor now disused, is the basic nuclear installation (INB 24) at the Cadarache research center. CABRI is operated by CEA for IRSN.

CABRI was commissioned in 1964. It is a pool-type reactor consisting of a 25 MW, water-cooled driver core[5] that uses slightly enriched uranium oxide (UO_2) fuel since the 1970s.

It is used to study the impact on fuel of reactivity-initiated accidents that may occur in various types of reactors. The experiments consist in very rapidly increases of reactor power above its normal operating level. The power peaks, which last only a few milliseconds, may reach 20 GW and are achieved by the more or less rapid and sequential emptying of control rods filled with a neutron-absorbing gas, the helium-3 isotope. The

5. The driver core produces neutrons for conducting experiments on the test fuel, which is placed in a loop that contains its own coolant, and crosses the core along its central axis.

65 cm width and 80 cm high reactor core is made up of 40 fuel rod assemblies designed to withstand the impact of these power peaks.

From 1978 to 2001, CABRI was used for research on reactivity-initiated accidents in sodium-cooled fast neutron reactors (SFR), using single fuel pins (or rods) placed in a sodium test loop. From 1993 to 2000, the sodium loop was also used for carrying out tests to simulate the first phase of a reactivity-initiated accident in pressurized water reactors. These were known in France as REP-Na (i.e. PWR-sodium) tests. CABRI is now equipped with a pressurized water loop for the purposes of a new research program comprising ten tests on pressurized water reactors, under conditions that are more representative of a PWR[6]. This program, called the CABRI International Program (CIP), was launched in 2000 (Figure 2.1).

The CABRI reactor includes a device called a hodoscope, which consists of several tens of fission and recoil proton ionization chambers positioned at the end of a collimator passing through the core. The hodoscope is designed to detect and precisely measure the solid or molten fuel movements in the test rods during the experiment.

SCARABEE, which is no longer used, shared its cooling systems with CABRI. Unlike CABRI, it was not used to produce rapid power transients. Equipped with a sodium loop with a larger diameter than that of the CABRI loop, it was used mainly for the study of hypothetical accidents involving fuel assembly blockage and

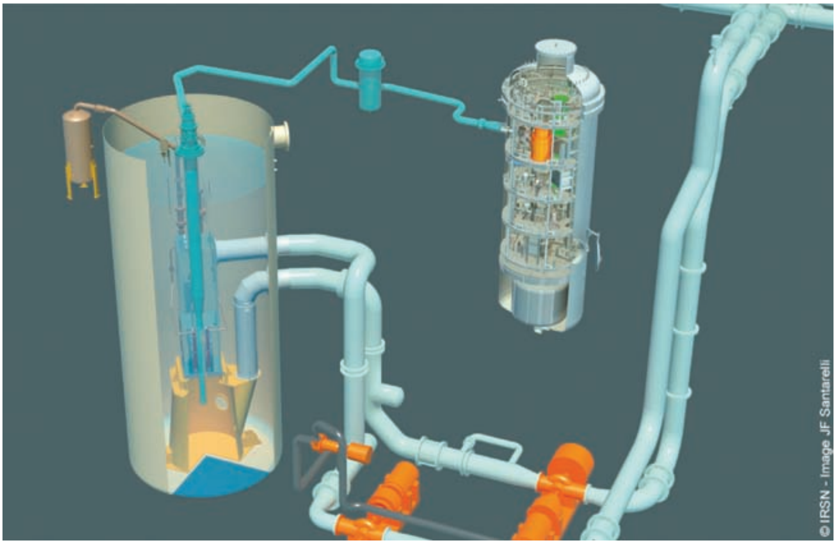

Figure 2.1 Diagram of the CABRI safety test reactor with its pressurized water loop.

6. The pressure and temperature in the water loop are respectively 155 bar and 350° C.

meltdown in high-power fast neutron reactors (SUPERPHENIX, "RNR 1 500" project), from 1983 to 1989. The tests were performed on small assemblies comprising up to 37 fuel pins.

2.2. The PHEBUS reactor

PHEBUS was an experimental nuclear reactor (INB 92) also located at the Cadarache center and operated by CEA. Commissioned in 1978, it was intended for research on accidents liable to affect pressurized water reactors. It has not been used since 2010.

PHEBUS was a pool-type reactor consisting of a 40 MW drive core, equipped with a cooling tower that enabled it to operate at high power for several days, unlike CABRI, whose operating time cannot exceed tens of minutes. It was used to provide neutron flux to heat the test fuel that was inserted into the pressurized test loop at the center of the core and produce radioactive substances, whose behavior was studied during a subsequent phase of the experiment.

It was mainly used as the experimental tool for a Fission Product program (Phebus-FP), which was aimed at studying what became of fission products from a pressurized water reactor core in the event of core melt. This program was launched by IPSN in 1988 under a partnership with EDF and the European Commission, and with a number of other countries (United States, Canada, Japan, South Korea and Switzerland). It lasted from 1988 to 2010. PHEBUS had previously been used for the Phebus-LOCA and Phebus-CSD programs.

CEA plans to file an application with the French Nuclear Safety Authority (ASN) in 2017 for the final shutdown of PHEBUS.

2.3. The GALAXIE experimental facility

GALAXIE is an IRSN-operated experimental facility that is also located at the Cadarache center. It groups together experimental facilities with various capabilities, designed for conducting experimental research on fire risk control in nuclear facilities (see details in Chapter 7). A brief description of the two main GALAXIE test facilities is given below.

► DIVA (Fire, Ventilation and Airborne Contamination Device)

The DIVA facility (Figure 2.2) is used for performing tests on fires that may occur at fuel cycles facilities as well as pressurized water reactors. The tests are performed in configurations involving several ventilated rooms. DIVA consists of three 120 m^3 rooms, a 150 m^3 corridor, another room with a volume of 170 m^3 on the second floor, and a ventilation network. It is very well-instrumented, with up to 800 measuring channels. Its reinforced concrete structure and its equipment have been designed to withstand a gas pressure range of – 100 hPa to + 520 hPa. Leakage between rooms and ventilation configurations can be adjusted to study fire in situations involving several confined and ventilated rooms.

Figure 2.2 The DIVA facility © Jean-Marie Huron/Signatures/IRSN.

▶ SATURNE

The SATURNE calorimeter cone (Figure 2.3) is a facility used to determine characteristic variables of a fire, such as its heat release rate and mass flow. The capacity of the hood and ventilation network (maximum capacity of 30,000 m^3/h) is suitable for studying fires with heat release rates of up to 3 MW.

2.4. Other facilities

IRSN uses five other experimental facilities in the field of pressurized water reactor safety.

▶ The CALIST experimental facility (Saclay)

The CALIST (Characterization and Application of Large and Industrial Spray Transfer) experimental facility was designed to characterize the size and velocity of drops from one or more spray nozzles and for the study of air entrainment by the sprays.

▶ The TOSQAN facility (Saclay)

The TOSQAN facility is used to simulate the thermal-hydraulic conditions inside the containment building of a nuclear reactor in the event of a core melt accident. This facility is designed for the analytical study of physical phenomena influencing hydrogen

Figure 2.3 The SATURNE facility. © Olivier Seignette/Mikaël Lafontan/IRSN.

distribution in a reactor containment: wall condensation, exchanges induced by the containment sump or the containment spray system.

► The CHIP facility (Cadarache)

The CHIP[7] facility is used for physical-chemical studies. It consists of ovens in which chemical reagents are placed, and temperature-controlled axial profile tubes through which these reagents and a carrier gas are injected. It is used in the CHIP research program, which focuses on iodine chemistry far from thermodynamic equilibrium (impact of chemical kinetics) in the reactor coolant system of a water reactor during a core melt accident.

► The EPICUR facility (Cadarache)

The EPICUR[8] facility is a panoramic irradiator composed of cobalt sources (^{60}Co). It emits γ radiation characterized by an average dose rate of 10 kGy/h. It consists of an experimental chamber used to study the effect of radiation emitted by the radioactive substances found in the containment of a pressurized water reactor during a core melt accident. It is used for the program of the same name, which studies iodine chemistry in a reactor containment in an accident situation.

7. Iodine Chemistry in the Reactor Coolant System.
8. Physical and Chemical Studies of Contained Iodine under Radiation.

► The IRMA Irradiator (Saclay)

IRMA[9] is a panoramic irradiator used to irradiate materials. It consists of cobalt sources (^{60}Co) and emits γ radiation characterized by an average dose rate of 20 kGy/h. Various scientific and technical experiments are carried out at the facility. By nature, they concern the study of radiation/material interaction mechanisms and, more especially, the response of materials and components to gamma radiation and any related degradation. IRMA is also used in the study and design of biological shielding and protection for various types of nuclear facilities. Its research potential (high γ flux) is also compatible with accident studies.

9. IRradiation of MAterials.

Chapter 3
Research on Loss-of-coolant Accidents

Loss-of-coolant accidents (or LOCAs) are among the design-basis operating conditions to be considered for pressurized water reactors, in line with the deterministic safety approach. They are hypothetical postulated accidents (Plant Condition Category 4[10]), in which the initiating event is a break in the reactor coolant pressure boundary. A break of this type leads to a pressure drop, of varying suddenness, and a loss of water. This inexorably causes the fuel rods to heat up, owing to the residual heat given off by the fuel, and in spite of the automatic control rod drop stopping any further nuclear fission reaction. This heating phenomenon must remain limited to ensure that fuel damage does not adversely affect reactor core cooling and does not lead to meltdown. Loss-of-coolant accidents are design-basis accidents considered, in particular, for the safety injection system (SIS), which is designed to inject water at varying flow rates, some mechanical components in the reactor coolant system, and the reactor containment building.

In the 1970s, safety criteria were defined for fuel rod cladding (the first barrier to ensure containment), based on the knowledge available at that time. They appear in United States regulations, in particular in 10 CFR (*Code of Federal Regulations*) 50.46 and its Appendix K, issued in 1974, and were adopted in France for the construction of the first nuclear power reactors under license to Westinghouse. These regulations were the result of long years of discussions between the Atomic Energy Commission (AEC), the forerunner to today's United States Nuclear Regulatory Commission (U.S.NRC), and the licensees of American nuclear power plants. Since 1974, however, reactor operating

10. See "Elements of Nuclear Safety", J. Libmann, EDP Sciences, 1996, Chapter 3.

conditions and fuels have changed (increased burnup, new fuel rod cladding materials, etc.), and this has led to the emergence of various research and development programs, described in the following pages.

Loss-of-coolant accidents involve complex phenomena in the reactor in three areas:

– thermal-hydraulics in the reactor coolant system,
– mechanical behavior of structures inside the reactor vessel,
– fuel rod thermomechanics.

The mechanical behavior of internal structures is an especially complex aspect of LOCAs, owing to the dynamic and dissymmetrical nature of water decompression in the reactor coolant system. The pressure drop, which ranges from 50 to 80 bar, propagates as a decompression wave through the reactor coolant system at the speed of sound in water (about 1000 m/s), and reaches the vessel *via* a single nozzle. This imposes significant mechanical loading on the vessel internals and fuel assemblies that must keep their geometry to ensure that rod drop can be made to shut down the reactor, and that the core cooling function remains operational. Ever since pressurized water reactors have been designed in France, the double-ended guillotine break[11] on a reactor coolant pipe (known as a 2A break) has been systematically postulated to study some of the consequences of a LOCA (core cooling capability, containment integrity, radiological consequences), but not others (mechanical strength of vessel internals and fuel assemblies, for example), for which limited guillotine breaks are considered. In this respect, specific devices to limit pipe deflection in the event of breaks are fitted on all pressurized water reactors currently in operation. In addition, for the break postulated at the reactor coolant pump outlet, where there is no device for restrain a pipe deflection, the break cross-section is limited by the rigidity of the cold leg.

Research on loss-of-coolant accidents has focused on two main areas:

– the study of two-phase thermal-hydraulic phenomena encountered during reactor coolant system draining, core reflooding, and fuel rod rewetting;
– the study of cladding and fuel behavior under accident conditions of this type.

Knowledge of these phenomena has been considerably improved over the past 30 years. This has led to the development of sophisticated simulation software codes used to study these accidents at reactor scale, and to determine whether safety criteria are met with sufficient margins as regards remaining uncertainties and, where possible, even improve these margins.

11. This is a complete (360°) pipe break where the two separated sections are pulled apart, thus maximizing the leak rate.

3.1. Two-phase thermal-hydraulics

French research in this area, carried out for the most part through cooperation between EDF, CEA, Framatome and IPSN, resulted in the development of the CATHARE[12] simulation code [1]. This code provides detailed behavioral models of the water flowing through the reactor coolant and secondary systems of a pressurized water reactor, from normal operating conditions up to the limits of standard design-basis conditions, i.e. fuel damage. In order to cover the widest possible range of thermal-hydraulic conditions, the liquid and vapor phases are processed separately, using a set of six equations (conservation of mass, momentum and quantity of energy). Initially, modeling was mostly one-dimensional, although in some parts of the reactor, such as the reactor core, flows may not be unidirectional, owing to non-uniform power distribution or changes in geometry during the accident. In the 2000s, the code was upgraded to include the possibility of modeling multidimensional flows and describing the behavior of droplets during core reflooding separately from that of the continuous liquid and vapor phases [2]. As it travels at high speed along the cladding, the vapor produces droplets due to a shear phenomenon; these circulate within the vapor film and contribute to heat transfers.

Designing this type of software to obtain a detailed description of thermal-hydraulic phenomena calls for precise knowledge of the physical laws governing mass, momentum and energy transfers at the interfaces between each phase and between these phases and the walls. For this reason, many analytical tests were performed as part of the development of this code. Most of them took place between the 1980s and 1990s in special, highly instrumented facilities built by CEA on its Grenoble site. The following is a non-exhaustive list of experimental facilities:

- CANON and SUPERCANON, for the study of depressurization and vaporization of water, first in a circular configuration, then in a geometrical configuration representative of a fuel assembly;
- MOBY DICK and SUPER MOBY DICK (SMD), for the study of two-phase flows passing through openings representative of the breaks studied in the reactor coolant system;
- OMEGA and APHRODITE (EDF/Chatou), for the study of film boiling around the fuel rods;
- DEBORA, for the study of flows under boiling conditions;
- SMD, for the study of friction at the liquid-vapor interface;
- PERICLES-2D, for the study of core uncovery and reflooding; in particular, it implements a mockup of three assemblies of different power levels;
- ROSCO, for the study of the very first moments of core reflooding, which are characterized by an unstable flow rate, as observed under experimental conditions during large-scale experiments on reactor models (see further on).

12. Advanced Thermohydraulics Code for Water Reactor Accidents.

In the 1980s, CEA, with support from EDF, Framatome and IPSN, designed the BETHSY facility on its Grenoble site, with the aim of checking whether the CATHARE code was able to reliably predict the behavior of a reactor in an accident situation. BETHSY was a mock-up of the reactor coolant system of a 900 MWe reactor. It was a full-scale model in terms of the height of the different components, while volumes were represented on a 1/100 scale (Figure 3.1). It consisted of three loops, each equipped with a pump and a steam generator, together with the secondary system components considered essential for thermal-hydraulic studies. The facility was designed for pressures of 17.2 MPa in the reactor coolant system and 8 MPa in the secondary system. The reactor core was represented on a 1/100 scale by 428 electrically heated rods with stainless steel cladding. Their power output was 3 MW, i.e. roughly 10% of the nominal power output of a reactor on the scale considered, thus allowing simulation of the residual heat in the core just after rod drop. All the engineered safety systems were reproduced, including the high- and low-pressure injection systems, the accumulators and the secondary system safety valves. Breaks could be simulated at various points in the reactor coolant system: in the cold leg, in the hot leg, at the top of the pressurizer, and in the steam generator. More than 1000 measuring channels monitored changes in key parameters during the tests (temperature, pressure, rate and direction of flow, void ratio, etc.).

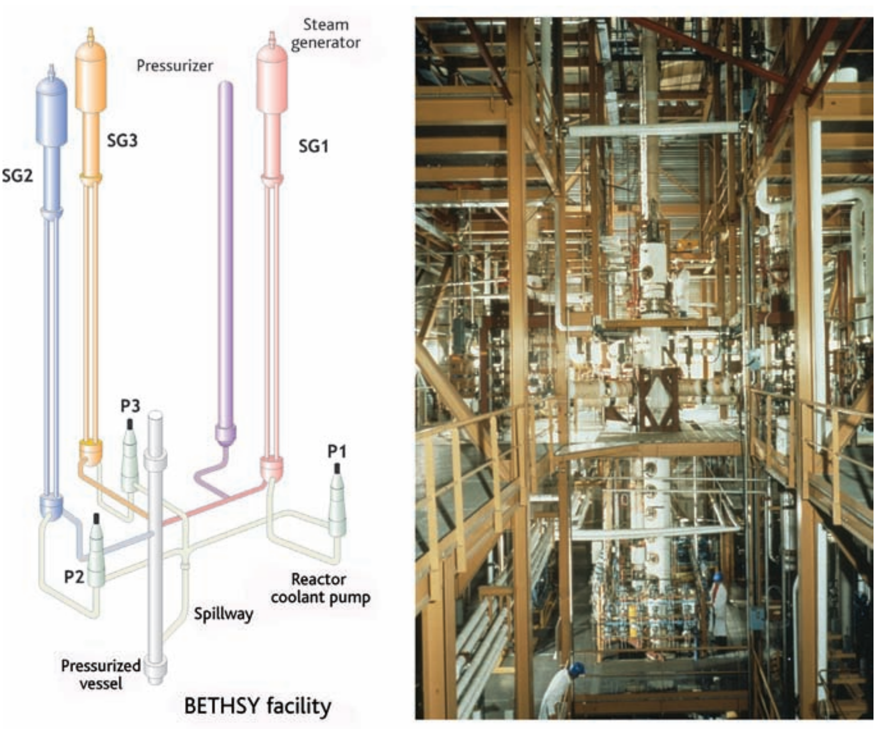

Figure 3.1 The BETHSY loop (900 MWe PWR, three loops). © Georges Goué/IRSN-Source CEA (left), CEA (right).

In all, more than 80 tests were performed between 1987 and 1998. They did not focus only on large-break LOCAs (complete pipe break). Other accident scenarios were studied, including those involving small or intermediate breaks, or nitrogen injection in the reactor coolant system after total drainage of the accumulators, or loss of cooling during an outage when the reactor coolant system is partially drained. The facility was also used to perfect control procedures (as part of the new state-oriented approach) to be implemented by licensees during the different phases of an accident to bring the reactor to a safe state.

In 1995, the international community selected test 6.9c as the basis for the OECD/NEA international benchmark exercise on simulation software, called "International Standard Problem (ISP) 38". This involved the study of a loss-of-coolant scenario during an outage, when pressurizer and the steam generator outlet plenum manways are open for maintenance work. ISP 38 brought together research organizations and licensees from 18 countries and compared the results of five different software codes with experimental results. It revealed that the main physical phenomena were effectively reproduced by the codes considered, although variations were observed due to faults in modeling multidimensional effects, particularly in the low-pressure range.

Similar facilities were built overseas in the 1970s and the early 1980s to study the thermal-hydraulic behavior of various nuclear reactor types in accident situations. The most important of these are listed below:

- Loss-of-Fluid Test, or LOFT, United States. A scale model of a 1000 MWe reactor (volume and power 1/50, height 1/2), comprising two loops, the only facility of its type with a nuclear core for heating water;
- Primärkreislauf, or PKL, Germany. A scale model of a 1300 MWe KWU reactor (volume and power 1/145, height full scale), comprising four loops and 314 electrically heated rods;
- Large Scale Test Facility, or LSTF, Japan. A scale model of a 1100 MWe reactor (volume and power 1/48, height full scale), comprising two loops;
- LOBI, European Community Joint Research Center, Ispra, Italy. A scale model of a 1300 MWe KWU-Siemens reactor (volume and power 1/700, height full scale). It comprised two loops and 64 electrically heated rods;
- PSB-VVER[13] (Russia). A scale model of a 1000 MWe VVER-type reactor (volume and power 1/300, height full scale) comprising four loops;
- Parallel Channel Test Loop, or PACTEL, Finland. A scale model of a 400 MWe VVER-type reactor (volume and power 1/305, height full scale), comprising three loops and 144 electrically heated rods.

International agreements, generally negotiated in connection with research work supported by OECD/NEA, gave CATHARE development teams access to many experimental results obtained during various research programs carried out at these facilities.

13. *Vodo-Vodianoï Energuetitcheski Reaktor*.

In all, nearly 300 tests were analyzed and used to qualify the simulation code on a broad spectrum of cooling accidents.

Most of these facilities were shut down and decommissioned. Some uncertainties remained, however. In some configurations, for example, water slugs containing no boric acid can form as a result of steam condensing in the steam generator tubes during a LOCA, for example. They can then be transported to the core where they are liable to initiate a criticality accident. In addition, performance checks had to be carried out on the passive cooling systems, based on natural convection, that were contemplated for use in some Generation III reactors. The studies aimed at eliminating these uncertainties called for the use of Computational Fluid Dynamics (CFD) codes, which are multidimensional codes that provide very detailed flow descriptions. It was, of course, important to check the predictive capabilities of these advanced tools through large-scale tests using systems that were as realistic as possible. For this purpose, the OECD/NEA backed new research projects in the PKL and LSTF facilities, in most cases with French involvement. These included:

- the PKL (2004–2007), PKL-2 (2007–2011) and PKL-3 (2012–2015) projects aimed at studying phenomena such as boric acid dilution in various situations, and natural convection in the event of loss of cooling during an outage, or to reproduce beyond-design-basis situations involving delayed safety injections for assessing safety margins;
- the ROSA[14] (2005–2009) and ROSA-2 (2009–2012) projects for studying thermal stratification and natural convection phenomena, and testing new cooling procedures in accident situations.

Tools simulating the behavior of the nuclear steam supply system were needed in order to train teams likely to be required in the event of nuclear emergency. The SIPA simulator was developed by IPSN in the 1990s using CATHARE modules. It has since been replaced by the SOFIA[15] simulator, which was jointly developed by AREVA and IRSN. It is used by IRSN in particular to produce accident scenarios used during national emergency exercises.

3.2. Fuel rod behavior

Research on fuel rod behavior during a loss-of-coolant accident has mainly focused on the following phenomena [3,4,5]:

- steam-induced oxidation of Zircaloy (zirconium alloy) cladding that modifies the mechanical properties of the cladding and generates hydrogen and heat;
- cladding swelling and rupture;

14. Rig Of Safety Assessment.
15. Observation Simulator for Incidental and Accidental Operation.

– the mechanical behavior of oxidized cladding with regard to the thermal shock induced by reflooding and to other phenomena during core cooling over the longer term;

– fuel pellet behavior inside the ballooned cladding, as the ceramic breaks up due to the stress produced by reactor operation.

The earliest research work on zirconium oxidation due to steam at high temperatures was carried out in the 1950s by Bostrom and Lemmon in the United States. The physical process is quite complex. It involves the adsorption and dissociation of water molecules at the cladding surface, the formation of O_2^- ions in the zirconia (zirconium oxide) layer that forms around the edge of the cladding, with the ions diffusing to the oxide-metal interface, where they contribute to the formation of zirconia or continue to diffuse through the metal. At constant temperature, the zirconia layer builds up according to a law that is parabolic with time ($m^2 = Kt$, where m is the mass of oxidized zirconium by unit area and t the time), which indicates that ion diffusion in the zirconia is the predominant phenomenon. The activity of the process increases with temperature, and the reaction rate K is temperature-dependent according to an Arrhenius equation ($K = e^{-c/RT}$, where c is a constant, R the ideal gas constant, and T the temperature in Kelvin).

Baker and Just at Argonne National Laboratory (ANL, United States) reviewed the experimental results and completed them by performing other experiments at higher temperatures, using hot filaments. Their correlation giving reaction rate *versus* temperature has been considered as a reference ever since.

A great deal of other research carried out in the 1970s backed up their work using different methods and zirconium alloys: Cathcart and Powell at Oak Ridge National Laboratory (ORNL, United States), Brown and Healey of the Central Electricity Generating Board (CEGB, United Kingdom), Urbanic and Heidrick of Atomic Energy of Canada Limited (AECL, Canada), Lestikow and Schanz of Kernforchungszentrum Karlsruhe (KfK[16], Germany), and Prater and Courtright at Pacific Northwest National Laboratory (PNNL, United States). Other more precise correlations of the oxidation reaction rate were obtained making a distinction between the various crystal systems of zirconia (monoclinic, tetragonal and cubic) according to temperature.

Depressurizing the reactor coolant system and emptying the reactor core causes the cladding to heat up and leads to an increase in the difference between the pressure inside and outside the fuel rods. At around 700 °C, the Zircaloy sees a sharp decline in its mechanical strength, but remains very ductile, which causes strain of up to nearly 50% before rupture. The surface area of cladding exposed to steam oxidation can therefore grow considerably, whereas the cladding becomes thinner. After rupture, the inner surface of the cladding is exposed to steam and can also be subject to oxidation.

The mechanical phenomenon was studied in the 1980s in CEA's EDGAR facility at the Saclay center. Some 500 tests were performed on tubes made of various zirconium alloys and directly heated by the Joule effect. This heating method ensured that temperature was uniformly distributed throughout the tube. These tests determined the creep and

16. Became after FzK (Forschungzentrum Karlsruhe) and then KIT (Karlsruher Institut für Technologie).

rupture elongation laws according to tube temperature and the temperature and pressure ramps. In particular, the results highlighted the effect of heating kinetics on the metallurgical phase change (from phase α to phase β) in the zirconium, which occurs between 800 °C and 1000 °C, and significantly modifies rupture elongations. These laws are used in the CATHARE code module for calculating the mechanical behavior of fuel rods during a loss-of-coolant accident.

Other research was also carried out on this phenomenon in the 1970s at the REBEKA facility of KfK in Germany and ORNL's MRBT[17] facility in the United States. The heating method was the main difference between these tests and those carried out at the EDGAR facility. In these tests, the zirconium alloy tubes were equipped with internal electrical heaters placed inside unheated cases. The heating element was not centered inside the tube, as a result of which tube temperature was not completely uniform. This reflects reality more closely, at least for low-burnup fuel rods, where a gap remains between the fuel pellets and the inner surface of the cladding. The tests showed that a slight azimuthal temperature gradient (around 10 degrees) could have a significant impact on cladding strain, with creep developing mainly at the hottest point. This leads to local ballooning and lower total rupture elongation.

Further tests were carried out at these facilities with a geometry involving the use of several rods arranged in a square-pitch configuration (5 × 5 and 7 × 7 assemblies in REBEKA, 4 × 4 and 8 × 8 in MRBT), with some unheated tubes. This was because some assemblies in a reactor are equipped with guide tubes through which the absorber rods in the control rod assembly can slide into the core. Similar tests were performed in Japan during the same period. They revealed that once the rods came into contact with one another, cladding swelling spread in an axial direction. They also produced the highest fuel assembly blockage rates (up to 90% during an MRBT test).

The shape of the blocked areas in the assemblies depends on the degree of coplanarity of cladding circumferential strain and the axial extent of this strain. There is reason to fear that this shape may not be compatible with correct cooling of these areas after core reflooding. This phenomenon was studied at various facilities under experimental conditions at the end of 1970s and in the early 1980s. Examples include the FLECHT-SEASET[18] program conducted by Westinghouse Electric Corporation Nuclear Energy Systems (United States), the FEBA[19] and SEFLEX[20] programs carried out by Forschungzentrum Karlsruhe (Germany – FzK) and the THETIS and ACHILLES programs of the United Kingdom Atomic Energy Authority in Winfrith (United Kingdom). The experimental facilities consisted of assemblies of several electrically heated rods (up to 163 rods of normal length in the FLECHT-SEASET program). A group of these rods included sections with a larger diameter in some places to reproduce the clad swelling anticipated in the event of a LOCA. The main parameters studied were the locally blocked

17. Multi-Rod Burst Test.
18. Full-Length Emergency Core Cooling Heat Transfer–Separate Effects tests And System-Effects Tests.
19. Flooding Experiments with Blocked Arrays.
20. (Fuel Rod) Simulator Effects in Flooding Experiments.

fraction of the straight section, the reflooding rate and the type of reflooding (forced, or passive gravity-driven).

Multidimensional thermal-hydraulic computer tools, designed to take into account local changes in geometry, were required for the analysis of these different tests. It was found that in areas subject to significant blockage (90% of the straight section blocked), the cladding could be cooled during reflooding, provided that the height of the blockage was limited (less than 10 cm). Note that all these experiments were carried out with a stationary heating element inside the rods. The conclusions of these studies could be contested because, in reality, nuclear fuel breaks up after several months of operation and, as will be seen in some of the studies mentioned further on in this document, fuel fragments may fill some of the gap inside the ballooned cladding. In France, IRSN has been developing the DRACCAR[21] multidimensional computer code since 2007 for studying local blockage rates and all the above phenomena. The code has already been used in the examination of the safety analysis reports submitted by EDF in connection with the planned update of the LOCA study baseline[22].

Many experimental studies were carried out in the early 1970s on the mechanical behavior of oxidized cladding with regard to the thermal shock produced during fuel rod reflooding. Argonne National Laboratory and Oak Ridge National Laboratory in the United States were prominent in this area. These studies produced the experimental data on which the maximum temperature of 1204 °C and the maximum equivalent oxidized cladding fraction of 17% in U.S. regulations (10 CFR 50.46 and its Appendix K) were based. The aim of these criteria was to provide a minimum requirement regarding cladding ductility at temperatures above 135 °C, which is the water saturation temperature during core reflooding.

As the experimental studies concerned new cladding, it was naturally necessary to investigate the possible impact of in-service corrosion on the above values. While it is inside the reactor core, the cladding oxidizes, and the hydrogen that results from the dissociation of the water molecules migrates through the thickness of the cladding. Beyond certain concentrations, the hydrogen in the solution reacts with zirconium to form hydride precipitates that embrittle the cladding material.

These phenomena were the subject of experimental studies in France as part of the research programs conducted at the CEA Grenoble center from 1991 to 2000, in partnership with EDF and IPSN. The TAGCIS[23] program studied the effect of a zirconia layer around the edges of the cladding. Some 400 new cladding samples were placed inside a pressurized water loop until a representative outer layer of zirconia had formed. The samples were then heated by steam up to high temperatures, before being plunged into a water tank. Various parameters were studied, such as the temperature increase rate, the maximum temperature reached, the initial zirconia thickness, and simultaneous oxidation of the outer and inner cladding surfaces. Some tests were reproduced using

21. Deformation and Reflooding of a Fuel Rod Assembly during a Loss-Of-Coolant Accident.
22. Study method including, in particular, the criteria to be met.
23. Quench during a LOCA of fuel rod cladding that has undergone simulated irradiation.

samples from cladding that had been used in a reactor as part of the TAGCIR[24] program, with some cladding coming from rods with a maximum burnup of 60 GWd/tU.

In order to study the effect of hydride formation separately, new cladding samples undergo a treatment in order to load them with hydrogen. They were then oxidized by steam at high temperature (CODAZIR program) and some of the samples were quenched (HYDRAZYR program).

Analysis of all these results showed that the effects of in-service corrosion on the cladding were slight, in terms of both oxidation kinetics and mechanical strength during rod reflooding.

However, other research carried out by the Japan Atomic Energy Research Institute (JAERI[25]) in Japan since then found that the secondary hydriding phenomena occurring at high temperature on the inner surface of cladding after rupture considerably reduced cladding ductility during reflooding. Furthermore, this research highlighted the importance of reproducing the axial stress applied to the rods as a result of their bowing and blocking in the grids when the core is reflooded. IRSN and JAEA[26] continue to study these phenomena, with research focusing particularly on the types of stress to be considered. The results of these studies and research will be used to adjust the values of these criteria, especially for cladding that has undergone significant in-service corrosion.

Meanwhile, industrial research has focused on developing new zirconium alloys that offer improved resistance to in-service corrosion, consistent with longer reactor operating lives. These are zirconium and niobium alloys with additional elements destined to replace the "traditional" Zy-4 alloy. The new alloys are called Zirlo™, E110 and M5™, respectively developed by Westinghouse in the United States, by Russia, and by AREVA in France. Some of the safety tests mentioned above were, of course, repeated using these new materials for an accurate assessment of the margins obtained with regard to postulated events.

Given the complexity of the phenomena studied, it was important to conduct experiments that reproduced the conditions of a large-break LOCA on a large scale and as realistically as possible. The PHEBUS experimental reactor mentioned earlier was built in France in 1970. The Phebus-LOCA program was designed to study the loss-of-coolant accident caused by an instantaneous double guillotine break in the largest pipe in the reactor coolant system, as well as the effectiveness of emergency cooling. Tests were performed from 1979 to 1984, three with a single rod, and 22 with 25 rods that had never been used in a reactor, arranged in a square-pitch configuration, and with a fissile height of only 80 cm. The tests were aimed at observing fuel behavior in the limit cases of engineered safety system operation. The results showed that damage was limited, even under pessimistic conditions, and that cladding ballooning did not prevent core cooling in the case of fresh fuel (Figure 3.2). This can be explained by the fact that as cladding

24. Quench during a LOCA of irradiated fuel rod cladding.
25. JAEA (Japan Atomic Energy Agency) is the result of the merging of JAERI and JNC (Japan Nuclear Cycle Development Institute) in 2005.
26. Japan Atomic Energy Agency.

Accident conditions were reproduced based on pessimistic assumptions, in accordance with the conservative approach for the reference accident (large-break loss-of-coolant accident with operation of emergency cooling systems). It can be seen that the cladding (made of Zircaloy) has undergone creep-induced ballooning and ruptured following a temperature transient of up to some 1200°C (the fuel rods are pressurized under normal operating conditions). The cluster configuration remains consistent with cooling.

Figure 3.2 Phebus-LOCA – sectional view (post-mortem) of a test fuel cluster after a characteristic LOCA temperature transient. © IRSN.

temperatures were not uniform on the horizontal plane, balloon size was less than that observed during the experiments carried out in the EDGAR facility mentioned earlier.

Phebus-LOCA test 218 was used as the reference for a computer code benchmark test organized by the OECD/NEA (ISP 19).

Other integral experiments were carried out in foreign reactors, some of which involved portions of rods removed from a power reactor:

– PBF-LOC tests performed in the Idaho National Laboratory (INL) Power Burst Facility (United States) on a single rod irradiated up to 16 GWd/tU;

– tests performed in the FR2 reactor at the KfK center (Germany), on a single rod irradiated up to 35 GWd/tU;

– tests carried out in the ESSOR[27] reactor at the Joint Research Center (Ispra, Italy), using a single, unirradiated rod;

– FLASH tests performed in the SILOE reactor in Grenoble, with a single rod, which was irradiated up to 50 GWd/tU in one test;

– MT tests carried out in the NRU (National Research Universal) reactor at AECL's Chalk River center (Canada) using assemblies of 32 unirradiated rods that had preserved their original length.

On analysis, the results of these experiments revealed that the fuel in the rods removed from power reactors breaks up and tends to relocate and fill the gap left by cladding ballooning. These observations were recently confirmed by tests carried out in

27. ORGEL Reactor Test.

the reactor at the Halden center in Norway. The tests in question were carried out as part of the HRP[28] LOCA program, under the aegis of OECD/NEA from 2003 to 2012. In all, 13 tests were performed with a single rod, under conditions representative of a large-break LOCA. The rod samples studied came from Western and Russian pressurized water reactors, and from boiling water reactors. The burnups studied ranged from 50 to 90 GWd/tU. Results analyses confirmed that at the time of cladding rupture, fuel fragments were relocating due to the effect of rod depressurization. Much of the most highly irradiated fuel (90 GWd/tU) had even spread outside the rod through the break. Research on this phenomenon is in progress at the Studsvik laboratory in Sweden, where separate-effects tests are carried out on irradiated fuel to study post-rupture fuel dispersion. At Halden, research work continues, focusing in particular on rods from EDF nuclear power plants that have undergone five or six irradiation cycles.

So far, no integral experiment has been carried out inside a reactor using an irradiated rod assembly. The conclusions of the blocked area cooling tests mentioned earlier could be called into question if the balloons are filled by some of the fuel contained in the rods.

In 2013, IRSN, with EDF support and the participation of two CNRS research laboratories (LEMTA[29] in Nancy and INSA-LamCoS[30] in Lyon), launched a research program called PERFROI[31]. Scheduled to last six years, the program is jointly funded by the French National Research Agency (ANR), as part of the "Investment in the Future" program and, more specifically, the call for research projects in the field of nuclear safety and radiation protection, issued in the wake of the Fukushima accident in 2012.

Research in this area focuses on the study of dreaded blockages and the possibility of cooling them under reactor core reflooding conditions. It includes experimentation and modeling work to validate the DRACCAR simulation code by 2020. The program is built around two main topics: cladding mechanical properties and two-phase flow.

The first part of the program comprises creep and rupture tests performed in a temperature range of 600 °C to 1100 °C, using specimens of various zirconium alloys, some of which are pre-oxidized and pre-hydrided to simulate the different stages of in-service corrosion. In addition to these tests, swelling and rupture experiments will be carried out on segments of cladding arranged in a more realistic, but finely instrumented, configuration. These will include several rods used to study the effect of contact between neighboring rods on rupture, together with the extent of ballooning, particularly in the axial direction.

The second part comprises thermal-hydraulic tests that implement instrumented assemblies of 49 rods, simulating as realistically as possible anticipated blockage and fuel relocation phenomena. The technological challenge consists in developing electrical heating elements capable of realistically simulating nuclear power distribution in deformed rods partly filled with fuel fragments. The main parameters studied are:

28. Halden Reactor Project.
29. Laboratory of Energy and Theoretical and Applied Mechanics.
30. French National Institute of Applied Sciences - Laboratory for the Mechanics of Contacts and Structures.
31. Loss-of-Coolant Study.

balloon geometry, balloon overpower temperature and flow rate of injected water, and pressure. Some geometries studied will be similar to those already covered in earlier research programs (in particular the THETIS program mentioned earlier) for the purpose of comparison.

References

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Chapter 4
Research on Reactivity-initiated Accidents

The most severe design-basis reactivity-initiated accident (RIA) considered for a pressurized water reactor in terms of uncontrolled nuclear reaction is the control rod ejection accident. This accident is in plant condition category 4. In terms of safety, the aim is to ensure core coolability and avoid a core melt accident and, at the same time, check that this accident would not impair the integrity of the reactor vessel and coolant system were it to occur.

In the event of control rod drive mechanism failure, the control rod ejection accident is due to the difference in pressure between the reactor coolant system (155 bar) and the containment (at atmospheric pressure). This violent ejection causes a local runaway effect in the nuclear reaction for several tens of milliseconds (power pulse), leading to a rapid increase in fuel temperature. Neutron feedbacks limit the power transient before the reactor trip (intact control rod drop), which occurs in a second phase.

A specific safety criterion relative to rod ejection was defined in the 1970s based on American tests, limiting the deposited energy (enthalpy) in the fuel (generally expressed in cal/g) during the reactivity transient. In the early 1990s, however, following the Chernobyl accident and, more especially, in view of the gradual increase in fuel assembly burnups, the international scientific community began to question the validity of this criterion, which was defined for moderate burnups.

It was within this context that Japan and France developed research programs, including IPSN's tests in the CABRI reactor. These were principally aimed at improving understanding of the physical phenomena that could lead to cladding leaks, and the ejection of fuel fragments into the reactor coolant system, which could be detrimental to core cooling.

The OECD report cited in the reference section [1] is a State-Of-the-Art Report on the knowledge of RIAs as of 2010.

Two types of cladding failure can occur during an RIA:

- failure due to pellet-cladding mechanical interaction (PCMI), which can occur during the very first instants of the power excursion, with the heated fuel suddenly expanding more rapidly than the cladding, which is still cold;
- this can be followed (after a few hundred milliseconds) by post-DNB[32] failure, when the cladding bursts due to a degradation in the cladding-coolant heat exchange coefficient, followed by a sharp increase in cladding temperature, as well as rising pressure inside the rods as the fission gases initially trapped inside the fuel are released.

In view of the risks induced by burst cladding and the dispersal of fragmented fuel [2] – which could lead to water vaporization and rod cooling blockage – the U.S.NRC adopted the "conservative maximum limit" of 280 cal/g of UO_2 for the energy deposited[33] (enthalpy) during an RIA-type power transient in its Regulatory Guide 1.77. The aim was to ensure that "core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired". The adopted value was deduced from experiments performed using unirradiated and slightly irradiated fuel (up to 32 GWd/tU) as part of the Special Power Excursion Reactor Tests (SPERT [1969–1970] in stagnant water and at ambient temperature). In the early 1980s, however, in light of continued experiments conducted under more realistic conditions at the INL Power Burst Facility reactor (PBF [1978–1980], circulating water, representative temperature and pressure conditions, but with burnup only reaching 6 GWd/tU), it was found that a value of 280 cal/g was not sufficiently conservative. Criteria were therefore revised downwards: in Europe, values of 220 cal/g and 200 cal/g were adopted for fresh and irradiated fuel respectively.

Increasing burnup to 52 GWd/tU or beyond, which is the value considered[34] by virtually all nuclear power plant licensees in the world, including EDF, can impair the mechanical properties of Zircaloy cladding (see Section 3.2), while the fuel contained inside the rod has undergone significant changes induced by reactor operation, such as fragmentation, and an increased quantity of trapped fission gases.

IPSN carried out a program of 14 tests in the CABRI reactor (Cabri REP-Na, 1993–2002) to validate this value. The tests used fuel from nuclear power plants at burnups ranging from 33 to 76 GWd/tU. Four of the tests used MOX[35] fuel, a mixture of uranium and plutonium oxides.

32. Departure from Nucleate Boiling.
33. Average radial value.
34. Adopted and authorized in some countries.
35. Mixed Oxide Fuel (UO_2 + PuO_2).

As mentioned in Chapter 2, the CABRI reactor is able to generate power peaks representative of those that could be encountered during a reactivity-initiated accident in a PWR. It achieves this through the depressurization of rods that have been pre-filled with a neutron-absorbing gas (^{3}He). The test fuel rod is placed in a test device and inserted in a cell at the center of the reactor. Almost all the meter-long test rods used for the Cabri REP-Na program were made from fuel rods removed from EDF-operated reactors. An instrument called a hodoscope for observing the fuel, and more conventional instruments (for taking flow rate, temperature, pressure, and acoustic measurements) were used to determine the precise moment when cladding burst occurred, estimate the quantity of fuel dispersed and measure the resulting pressure wave. Until 1992, the CABRI facility was used to carry out safety tests on fuel used in fast neutron reactors, and was equipped with a sodium loop. The tests were therefore performed with sodium flowing around the test rods. This was considered acceptable for the study of the predominantly mechanical phenomena occurring during the first few tens of milliseconds of the power excursion, when there is little effect on cladding temperature.

IPSN then launched a new program of experiments calling for a complete overhaul of the facility to study the phenomena occurring after the first few hundred milliseconds (cladding dryout and rupture), as well as the consequences on reactor structures in terms of pressure wave caused by the possible dispersal of fuel in the coolant. This was the OECD/NEA project called CABRI International Program (CIP, 2000–2015), carried out in partnership with EDF and many foreign safety and industrial organizations. Twelve tests are scheduled under the program, two of which were performed in 2002 in the previous facility using very high-burnup fuel (75 GWd/tU). The facility was then extensively overhauled. The overall seismic resistance of the reactor was increased, a new cell was installed and connected to a pressurized water system designed to reproduce representative thermal-hydraulic conditions, and many components were replaced.

Two other CIP tests (CIP3-1 and CIP3-2) are scheduled to focus on the study of post-rupture phenomena. Further tests are planned with low deposited energy on rods whose cladding has been embrittled before the tests as part of a post-CIP program. Tests such as these would help to confirm deposited energy criteria by examining whether fuel dispersal in the coolant would not be detrimental to core coolability[36], even if the cladding is embrittled or already flawed.

CEA performed a series of analytical tests for IPSN and EDF on the mechanical behavior of cladding samples taken from fuel rods removed from nuclear power reactors (as part of the PROMETRA[37] program at the Saclay center) and on critical heat flux phenomena during rapid wall heating (determination of critical heat flux under transient conditions as part of the PATRICIA program at the Grenoble center).

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36. IPSN had already considered this question in a similar way in the early 1980s during technical assessments in connection with the SUPERPHENIX reactor, for the specific case of inadvertent control rod withdrawal, especially as this event was classed in plant condition category 2 at that time.
37. Mechanical Properties in a Transient.

Other foreign programs with high-burnup fuels worthy of mention here include:

- the tests performed in Russian reactors: IGR (Impulse Graphite Reactor), 47 to 49 GWd/tU, from 1990 to1992; BIGR, 47 to 60 GWd/tU, from 1997 to 2000;
- the sixty or so tests performed in the JAEA Nuclear Safety Research Reactor[38] (NSRR) in Japan, using PWR, BWR (Boiling Water Reactor) and MOX fuels over a burnup range of 20 to 77 GWd/tU, from 1975 to 2011.

In all, nearly 140 tests were performed in reactor on high-burnup fuels. Analysis of the results shows that the fuel can be dispersed in the coolant at deposited energy levels well below the 200 cal/g of UO_2 criterion with a value of around 120 cal/g once the burnup exceeds 40 GWd/tU. This dispersal occurs after sudden cladding rupture due to:

- stresses applied by the fuel, the volume of which tends to increase with heating (thermal expansion and internal pressurization by fission gases), bearing in mind that the initial internal gap between the fuel and cladding is filled as soon as burnup exceeds a few tens of GWd/tU;
- impaired mechanical properties of cladding caused by the formation of zirconium hydrides that embrittle it; the hydrides are formed when some of the hydrogen released by the decomposition of water at the cladding surface in the reactor is diffused inside the cladding.

However, the tests also showed that results were highly sensitive to the type of cladding material. The worst results were obtained with Zircaloy-4. The rod clad with this alloy used during the Cabri REP-Na 1 test (64 GWd/tU) exhibited traces of spalling[39] of the outer zirconia layer that formed while the rod was inside the reactor. It ruptured with a rise in enthalpy of only 30 cal/g of UO_2, and about 2% of the fuel was dispersed in the sodium. More recently developed alloys (low-tin Zircaloy, Zirlo^{TM}, and M5^{TM}) are less sensitive to hydriding and are more resistant, even at high burnup.

Only 13 tests have been performed in the world using MOX fuel. The results of Cabri REP-Na tests indicate that, energy levels being equal, the cladding would be exposed to greater stress and, in the event of rupture, greater quantities of fuel would be dispersed. The mechanisms that might explain this difference in behavior have not yet been clearly identified and further experiments are planned to study them as part of the CABRI International Program.

In 2010, JAEA also launched an international research program called “Advanced Light Water Reactor Performance and Safety-II” (ALPS-II) in the NSRR reactor to improve

38. The TRIGA (Training, Research, Isotopes, General Atomics) pool-type reactor, designed and built by General Atomics. It has been in use since 1975. The power excursion is produced by the rapid ejection of neutron-absorbing rods. The fuel, a uranium-zirconium hydride alloy, enriched to about 20% with uranium-235, is designed to rapidly halt the excursion. The power peaks are not very wide (from 4.4 to 7 ms at mid-power), unlike in the CABRI reactor, where the transient rod depressurization valves can be adjusted to vary the peak width from 10 to a few hundred milliseconds.
39. Spalling is the localized loss of some of the zirconium oxide (or zirconia) layer that forms at the surface of the rod while it is in the reactor.

knowledge of high-burnup and MOX fuels. This program is a sequel to the first ALPS program conducted from 2002 to 2010, with 14 tests on high-burnup fuels (67–77 GWd/tU) and MOX fuels (45–59 GWd/tU).

The tests performed in the NSRR use an instrumented experimental capsule that accepts reconstituted rods filled with fuel up to a height of 120 mm. A single rod is tested during a test, with the cladding surrounded by stagnant water, which was initially at ambient temperature and pressure in all tests before those of the ALPS program. The instrumentation is used to measure cladding and coolant temperature, detect the moment of cladding rupture, and measure the mechanical energy developed during fuel dispersal and water vaporization.

For the ALPS program, JAEA designed a high-temperature (HT) capsule capable of operating at 280 °C and 7 MPa, values that are closer to real conditions. Six tests were performed in this type of capsule. Results clearly revealed the effect of initial cladding temperature. The cladding, which is made more brittle by the presence of zirconia hydrides in particular, bursts at low temperature at significantly lower energy levels.

The ALPS-II program should comprise 12 to 14 tests performed with samples of rods that have been used in European reactors (including, for France, a UO_2 rod with a burnup of 76 GWd/tU from reactor 5 of the Gravelines NPP, and a MOX rod with a burnup of 61 GWd/tU from reactor B3 of the Chinon NPP, both with M5^{TM} alloy cladding). Half the tests will be performed in an HT capsule. The program also comprises four to six analytical tests, called the Fission Gas Dynamics (FGD) tests, carried out by JAEA in partnership with IRSN, which contributes to the design of the experimental device. These tests are aimed at measuring the fission gas quantities released during a power excursion by fuels from French reactors, and an experimental fuel irradiated to 130 GWd/tU.

This research program came to a halt when Japan's reactors were shut down in the wake of the Fukushima accident.

As in many other fields, simulation codes are needed to acquire a full understanding of phenomena and transpose experimental results for the study of reactivity-initiated accidents at reactor scale. The SCANAIR[40] simulation code developed by IRSN is designed to calculate temperature and stress fields in fuel and cladding. It also calculates pressures due to fission gases inside the fuel. Meanwhile, more academic research is conducted with CNRS (MIST[41] laboratory, a laboratory "without walls", jointly funded by IRSN, CNRS and the University of Montpellier) to determine crack propagation laws inside hydrided cladding, and define rupture criteria.

Most countries operating pressurized water reactors consider that fuel-related criteria should be updated with regard to RIAs, and jointly fund experimental programs aimed at learning more about fuel behavior (CIP [IRSN] and ALPS [JAEA]). In France, talks between IRSN, the French Nuclear Safety Authority (ASN) and EDF have been ongoing since the early 2010s to discuss EDF's proposal for a "decoupling domain" aimed at precluding the occurrence of cladding rupture due to pellet-cladding mechanical

40. System of Simulation Software for Analysing Reactivity Injection Accidents.
41. Micromechanics and Structural Integrity Laboratory.

interaction in the event of a reactivity-initiated accident, with or without cladding spalling. The following parameters are considered:

– average rod burnup,
– maximum thickness (azimuthal average) of the external oxide layer,
– enthalpy variation,
– mid-height pulse width,
– maximum cladding temperature.

References

[1] Nuclear Fuel Behavior under Reactivity-initiated Accident (RIA) Conditions. State-Of-the-Art Report, OECD 2010, NEA No. 6847, 2010.
[2] Nuclear Fuel Safety Criteria Technical Review, *Second Edition* – OECD 2012, NEA No. 7072, 2012.

Chapter 5
Research on Recirculation Cooling under Accident Conditions

If a leak occurs in a PWR reactor coolant system and is not compensated by the chemical and volume control system (CVCS), reactor core cooling is ensured by injecting borated water. This is done by the safety injection system (SIS). In order to remove residual heat and preserve the integrity of the reactor containment building, the containment spray system (CSS) may also be required. Sodium hydroxide is added to the spray water to favor retention inside the reactor building of radioactive substances such as iodine-131.

The borated water required for these operations is initially drawn from the tank of the treatment and cooling system of the water of the spent fuel storage pools (RWST[42]), which has an approximate capacity of 1600 m^3 in a 900 MWe pressurized water reactor. When the low level threshold is reached in this tank, the SIS and CSS automatically switch over to recirculation mode. In this mode, they take in water that has been collected in the sumps located at the bottom of the reactor building (Figure 5.1). This recirculation mode may be required over very long periods of time to cool the fuel assemblies. It must be highly reliable as it is fundamental in avoiding fuel assembly damage and preventing a core melt accident.

The sumps at the bottom of the reactor building are equipped with a strainer system designed to ensure that the quality of water downstream of the strainers is compatible with the operation of SIS and CSS components, and with fuel assembly cooling. Debris can be generated as a result of the break (destruction of equipment by shock wave or jet) or

42. Refueling Water Storage Tank.

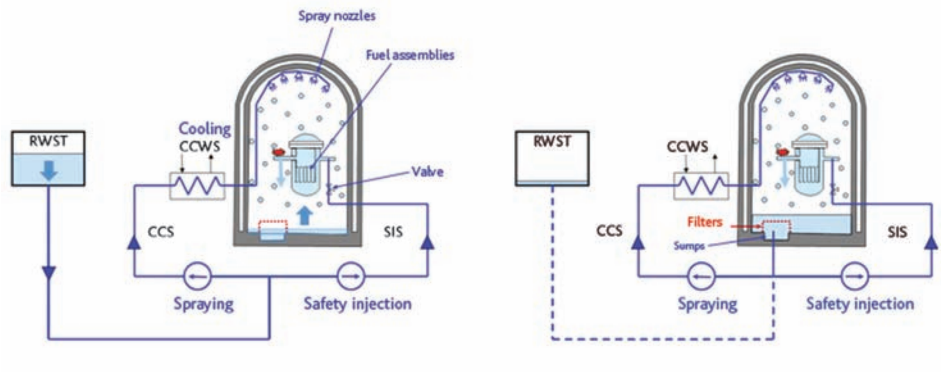

Figure 5.1 Simplified diagram of SIS and CSS operation in direct injection mode (left) and recirculation mode (right) – the component cooling water system (CCWS) is also responsible for cooling some pumps in the systems shown. © IRSN.

ambient conditions in the reactor building (temperature, irradiation, etc.), or it may already be present in the reactor building (dust, etc.). The flow of water (from the break in the reactor coolant system or CSS washdown) can then transport this debris to the sumps.

Depending on its characteristics (size and weight), some of this debris may be carried as far as the strainers and form a debris bed, which can lead to strainer clogging as a result of physical or chemical effects. The main risk involved is the impeding or obstruction of water recirculation.

Strainers must be sized (in terms of surface area and mesh size) to:

- prevent SIS and CSS pump cavitation failure. This risk is reduced by the use of sufficiently large screen surface areas;
- ensure the operation of components located downstream of the strainers, as well as fuel assembly cooling. This can be guaranteed by a suitable strainer mesh size.

5.1. Operating experience feedback and research topics

Risks relating to the loss of recirculation cooling capability in light water reactors were identified in the 1970s, and were the subject of the U.S.NRC Regulatory Guide RG 1.82 issued in 1974[43].

43. In 1974, the U.S.NRC published its "Regulatory Guide 1.82", which describes the methods and practices acceptable to U.S.NRC staff for addressing all the issues generally related to emergency recirculation cooling after a loss-of-coolant accident and applicable to PWR- and BWR-type light water reactors. The guide was gradually updated to reflect operating experience feedback and new knowledge resulting from research and development. This initial version of the Regulatory Guide was applied to French nuclear power plants designed by Westinghouse. The version currently in force is Revision 4 released in March 2012 [1].

During the early 1990s, however, several incidents occurred in boiling water reactors (at Sweden's Barsebäck plant, and at the Perry and Limerick plants in the United States), which raised new questions on the risk of strainer clogging. The incident that affected reactor 2 at the Barsebäck BWR plant in Sweden [2] on July 28, 1992 was caused by the inadvertent opening of a valve. The resulting steam jet produced 200 kg of fibrous debris, composed mostly of insulating material. Half of it was carried to the suppression pool or wetwell[44], leading to a significant increase in head loss across the emergency core cooling system strainers after 70 minutes. Given the modest size of the equivalent break (corresponding to the valve diameter), the quantity of debris appeared far greater than could have been expected based on RG 1.82 in its first revision in November 1985.

In view of the incidents mentioned above, international investigations focused especially on boiling water reactors. Research results concerning this type of reactor revealed that the quantity of debris generated by a pipe break in the reactor coolant system could exceed by far the quantities considered until then, based on earlier research. They also showed that the debris could be finer (and therefore more easily transportable) and that some combinations of debris (fibers with particles, for example) could augment the risk of strainer clogging. These results can be found in many documents, including the report NUREG[45] 6224 issued in 1995 [3] on the study of strainer blockage in BWR engineered safety systems.

This led to new studies being initiated in the late 1990s by industry, laboratories and research organizations, with active involvement on the part of IPSN.

Research work also revealed further issues relating to the chemical effects associated with the pH and temperature of the solution, and their impact on the risk of strainer blockage, as well as the effect of debris passing through the strainers on the functional performance of SIS and CSS engineered safety system components, and on fuel assembly cooling.

Research carried out since 1974, and incident analysis have raised many questions, for the most part in studies on the following topics.

► Characterization of debris likely to reach the strainers

It is essential to characterize the debris (type, quantity and size) likely to be transported to the strainers in order to assess: the risk of strainer clogging; the operation of components located downstream of the strainers; and fuel assembly cooling.

By far, the greatest amount of debris is directly produced by the shock wave generated by the break. In particular, it comes from the destruction of insulating material (glass fiber, Microtherm, etc.) on equipment located near the break (such as piping and steam generators), as well as paint and concrete. The quantity of debris depends especially on the pressure field induced by the break and the strength of the material

44. Pool located at the bottom of the reactor building from which the engineered safety systems draw water.
45. Nuclear Regulatory Report.

of which the debris is composed. Size and shape varies (fibers a few millimeters in length, particles a few micrometers in diameter).

Other debris is generated by ambient conditions (temperature, humidity, irradiation) in the reactor building. It may be composed of chips (a few mm^2) or particles (a few µm in diameter) of worn or damaged paint.

Lastly, some debris, such as dust or grease, is found in the reactor building before the accident. Referred to as latent debris, it can be entrained in streams of water and must be taken into consideration in strainer design.

► Debris transport to strainers

Some debris can be transported (Figure 5.2) to the bottom of the reactor building in the water flowing through the break and in containment spray washdown. Other debris can be trapped in retention zones – for example on floors and gratings. This is called vertical debris transport.

During the switchover to recirculation mode, the debris transferred to the bottom of the reactor building can, depending on its size or weight, be transported to the SIS and CSS sump strainers. This is referred to as horizontal transport.

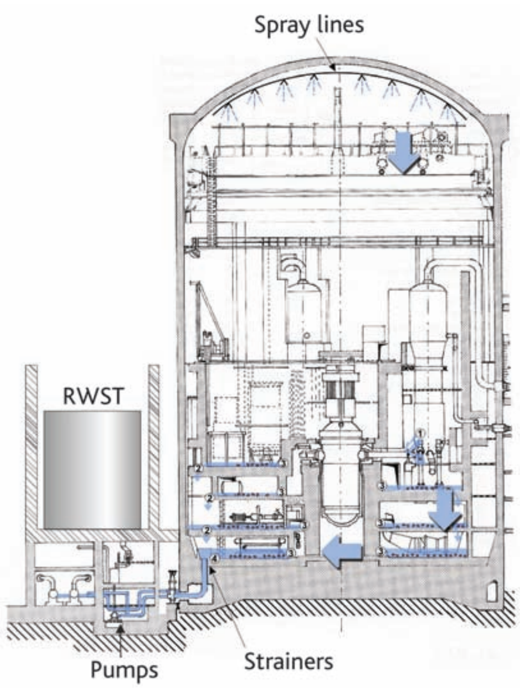

Figure 5.2 Illustration of debris transport to sump strainers (900 MWe PWR containment building). © IRSN-source EDF.

► Assessment of strainer head loss and clogging risk

The debris transported to the strainers may either agglomerate in the strainer mesh and form a debris bed over the entire screen surface area (Figure 5.3), thereby increasing the head loss across the strainer terminals, or, depending on its size or the permeability of the debris bed formed, be transported downstream of the strainer, and induce "downstream" effects. Strainer head loss is determined using correlations, such as the NUREG 6224 correlation, that are used to check that there is no risk of pump cavitation failure.

In this respect, research has demonstrated that for some strainers, just a small quantity of mixed debris (fibers and particles) can lead to significant head loss across strainer terminals in recirculation mode (Figure 5.4), and thereby increase the risk of pump cavitation. This is known as the thin film effect.

► Impact of chemical effects on strainer head loss

Research work, in particular that carried out by IRSN, has shown that physical-chemical phenomena can lead to the formation of precipitates (crystals, gels) in the debris bed. The precipitates can build up and cause an increase in strainer head loss. These phenomena depend not only on water pH and temperature, but also on the characteristics of the debris in the containment, especially in the water at the bottom.

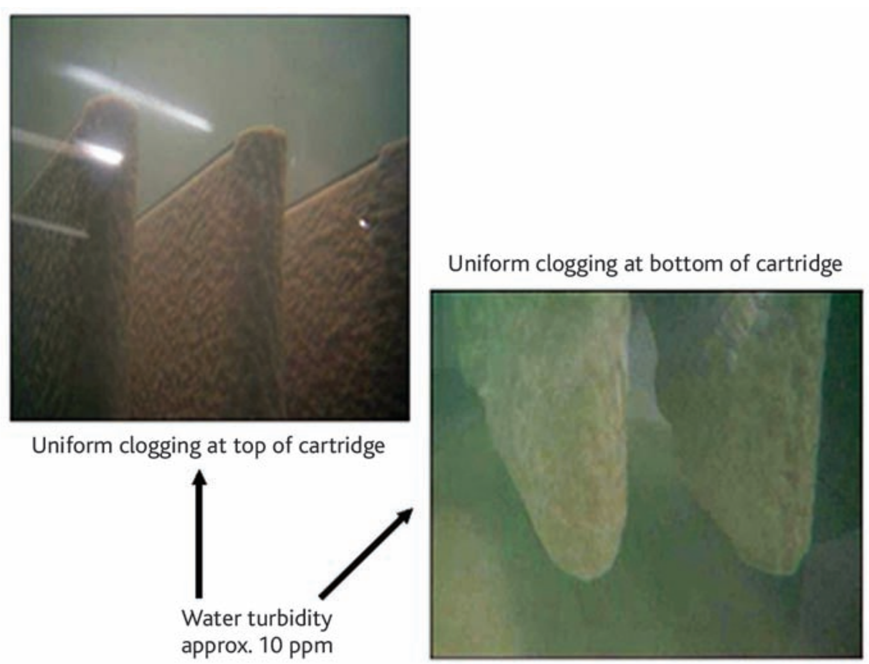

Figure 5.3 Examples of fibrous and particulate debris covering screen surfaces (qualification tests on SIS strainers for the EPR). © IRSN.

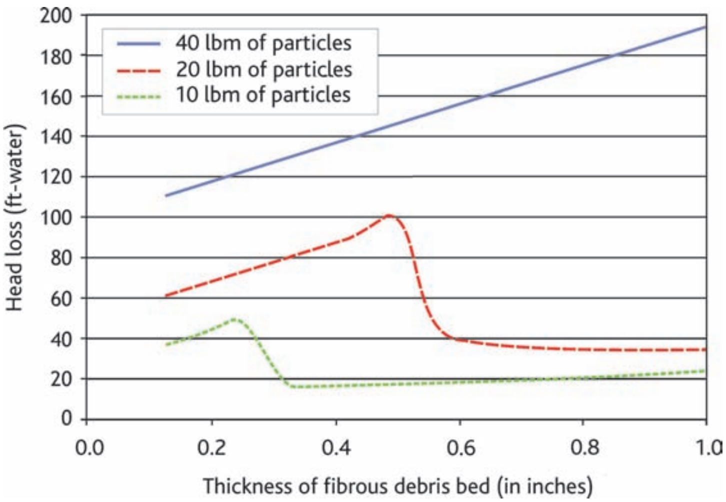

Figure 5.4 Diagram of strainer head loss peak or thin film effect [6].

► Downstream effects

Depending on strainer system characteristics (mesh, screen surface area), the debris passing through the strainers, and the chemical substances in the water, can lead to physical and chemical phenomena, such as soiling, clogging and erosion. This can impair the performance of SIS and CSS components, such as pumps, diaphragms, heat exchangers, valves and safety valves, and compromise fuel assembly cooling in the core (partial blockage of fuel assembly grids).

5.2. Past research programs and lessons learned

Following the BWR incidents that occurred in the 1990s, a considerable number of studies and research and development programs were conducted around the world. This was particularly the case in France with work carried out by EDF and IPSN (then IRSN), to obtain a fuller understanding of the phenomena involved and assess strainer system design.

As part of a knowledge-sharing initiative, OECD/NEA published a knowledge-base report [4] in 1995, following the incident at the Barsebäck plant. An updated version of the report was issued in 2013 [5]. More than ten laboratories and research organizations worked – or continue to work – on these topics for safety authorities and TSOs[46] (U.S. NRC, IRSN, etc.), designers and industry (e.g. General Electric, AREVA, EDF). Progress in some of this work is reported in a number of documents [6,7,8]. IPSN collaborated with

46. Technical Safety Organization.

VUEZ and the University of Trenčín (Slovakia), and with EREC (Russia) on a series of tests in experimentation loops, described in greater detail below.

EDF took part in the Nuclear Energy Institute (NEI) study and research program on recirculation issues that led to the publication of a guide [6, 7] in 2004 describing methods for evaluating strainer design for PWRs.

In an initial research program, IPSN defined and funded a number of experiments, most of which were carried out by VUEZ and EREC from 1999 to 2003. IPSN supervised the experiments. The following topics were studied:

– efflorescence of debris induced by water flow: 77 tests were performed in the ELISA facility set up by IPSN at VUEZ to study the impact of insulation type and water flow rate, temperature and quality, as well as how solid particles affect head loss due to the presence of insulation material on a strainer screen;

– vertical debris transport and crushing caused by obstacles: 30 tests were carried out in the IVANA facility, also installed at VUEZ, to assess the size of debris generated, based on the initial size of debris, spray water flow rate, and the type of insulating material considered;

– horizontal debris transport rates and debris settling in the containment: 52 tests were performed in the VITRA facility, set up by IPSN at EREC, to study the water flow rates below which debris, depending on its characteristics, settles on the horizontal floors of the containment building, and thus does not contribute to sump strainer clogging;

– strainer obstruction mechanisms: following a campaign of 15 preliminary tests, 11 full-scale tests were performed at the MANON facility, set up by IPSN at VUEZ, to assess the quantities of debris likely to seriously compromise the performance of water recirculation pumps.

In 2003, IRSN shared to the French Nuclear Installations Directorate the lessons learned from international research and, in particular, the tests mentioned above. The findings of this work raised the issue as to whether the possibility of sump strainer clogging could lead to loss of recirculation cooling in PWRs under accident conditions. In 2004, EDF decided to make a number of engineering changes to its nuclear power plants in operation, based on the NEI guide mentioned earlier [6,7] and the results of the ensuing research. These changes were implemented on reactors from 2005 to 2009. The new strainer design is now based on methods and practices developed by EDF that are specific to French reactors but also comply with the design provisions of the latest version of U.S.NRC Regulatory Guide RG 1.82.

Until the early 2000s, the mesh size of the strainers installed in French nuclear power varied with the reactor series. The strainers were vertical panels installed in the sumps in a circumferential arrangement. Based on research and development findings, two changes (decided in 2004) were implemented:

– removing Microtherm insulation, because this material can generate very small particles, making it unacceptable in terms of the risk of sump strainer clogging;

– fitting strainer systems with new strainers with a significantly larger surface area (up to 48 times that of the previous strainers).

Note that for the EPR, the strainer system adopted consists of two[47] lines of strainers installed across the path of the debris:

– basket strainers at the edge of the Internal Reactor Water Storage Tank (IRWST), which provides a reserve of borated water in the containment, at the floor openings in large components;

– strainers in the middle section of the IRWST (In-containment Refueling Water System Tank), where the spray lines leading to the pumps are installed.

Following the studies it carried out from 1999 to 2003, IRSN decided to launch a joint program with VUEZ and the University of Trenčín to study the chemical effects of the fibrous bed deposited on the strainers.

The aim was to assess how the formation of precipitates impacted strainer head loss, with engineered safety systems operating in recirculation mode, following a reactor coolant system break. The program was in two parts. The purpose of the first part was to determine the concentration of precipitates that might form. The second sought to determine the impact of precipitate formation on fibrous bed head loss, based on tests carried out in the ELISA test loop.

The program confirmed that precipitates could form in a fibrous bed and that this significantly affected strainer head loss. Tests also showed that temperature had a considerable impact on the formation, type and development of precipitates. A strainer head loss computational model was developed based on the above results. The model is also designed to provide fibrous bed porosity data.

IRSN initiated another research program to confirm the above model and consolidate knowledge concerning long-term corrosion in different types of insulating material and the precipitates that could form as a result. The program was performed using six test loops called ELISA Babies, derived from the design of the ELISA loop.

5.3. Ongoing research programs

As of 2015, many research programs were still in progress both in France (EDF and IRSN) and abroad. These are mostly concerned with studying the effects of debris on engineered safety system components located downstream of the strainers, and on fuel assembly cooling. Some research programs are also aimed at modeling physical and chemical phenomena more precisely to substantiate some strainer designs.

In 2015, for example, IRSN decided to launch new research programs involving the use of VIKTORIA, a new test loop installed at VUEZ. The aim is to evaluate the design of strainer systems used in currently operating reactors and those intended for use in the EPR, in terms of chemical effects and the effects of debris passing through the strainers.

47. In addition to trash racks installed on the floor of large components.

Figure 5.5 View of the VIKTORIA test loop © Brano Valach/IRSN.

VIKTORIA is an integral loop (Figure 5.5) that can simulate all the key physical and chemical phenomena for analyzing questions relating to strainer systems. It can be used for various types of pressurized water reactors for the following purposes:

- studying effects upstream of strainers including:
  - strainer head loss,
  - impact of chemical effects on this head loss,
  - gas formation,
  - the effect of back-flushing, i.e. injecting water inside the strainers to unclog them;
- studying physical and chemical effects downstream of the strainers, which entails characterizing the debris (quantity, type, size) passing through the strainers, and its impact on fuel assemblies and other components, such as heat exchangers and diaphragms.

Otherwise, research and studies carried out so far have not considered core melt situations. In the EPR, however, a core melt situation would require the use of the Containment Heat Removal System (CHRS), which is equipped with strainers. The types and quantities of debris generated in such situations could differ from those considered in design-basis transients owing to the extreme conditions (temperature, irradiation) encountered in the event of core melt. This could lead to changes in the local chemistry of the radionuclides trapped in the debris (as a result of water radiolysis) and cause precipitates. IRSN has begun an exploratory study on this topic that could lead on to more specific research over the coming years.

5.4. Simulation

The subject of debris, and its impact on recirculation cooling following a break in the reactor coolant system, involves a wide range of generally three-dimensional phenomena and disciplines, such as hydraulics and chemistry. There is a gradual increase [8] in the use of CFD simulation tools for studying issues such as debris transport, the deposition of debris on strainers, variation in head loss through a debris bed, and the risk of air and incondensable gas entrainment.

IRSN has used this type of tool for studying the representativeness of EDF tests carried out to substantiate the design of new strainers. These include tests on chemical effects. Based on the initial results of these tests, EDF considered that chemical effects had no impact on strainer head loss, owing to significant settling observed at the bottom of the test loop, and the fact that a stable debris bed could not be formed on the strainer. IRSN simulations, for example, revealed that debris was drawn down upstream of the loop strainer (Figures 5.6 and 5.7). The representativeness of these tests for the case of a reactor therefore remained to be demonstrated.

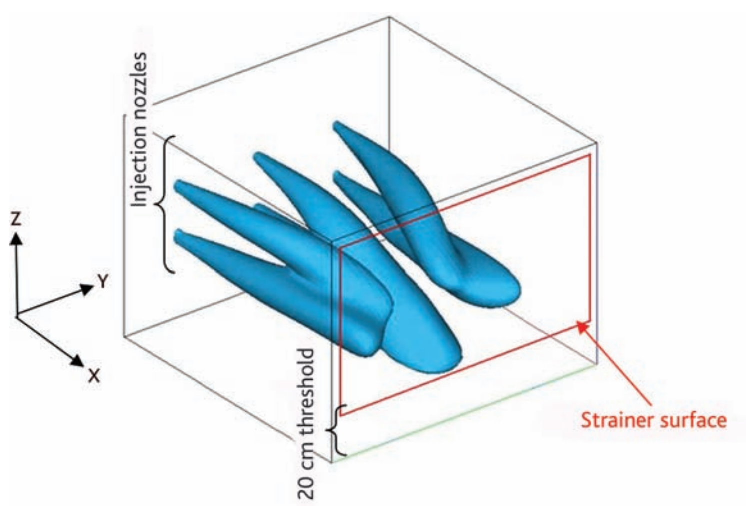

Figure 5.6 View of jets in the EDF/CEMETE model (velocities greater than 6.5 cm/s) where x is the direction of flow. © IRSN.

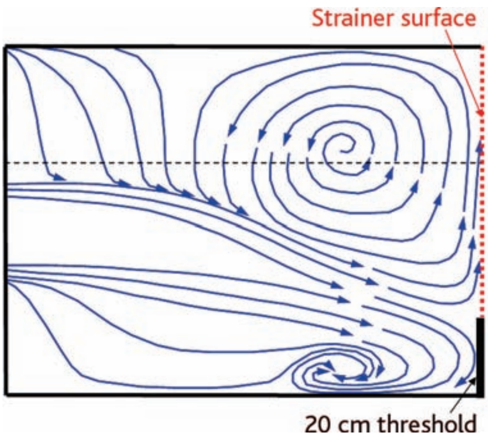

Figure 5.7 View of streamlines in the direction of flow at the middle of the EDF/CEMETE model (in the vertical plan) – the injection nozzles are on the left. © IRSN.

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Chapter 6
Research on Spent Fuel Pool Uncovery Accidents

Nuclear power reactors have a spent fuel pool where spent fuel assemblies are stored until they are ready to be transported to the reprocessing plant, in other words, when they have lost enough of their residual heat. During reactor outages, the spent fuel pool can also be used to temporarily accommodate fresh or already irradiated fuel assemblies for loading into the reactor core before it is restarted. The spent fuel pool is about 12 m deep. The fuel assemblies are stored in racks placed on the bottom of the pool, under about seven meters of water, thus providing workers with sufficient protection against ionizing radiation under normal operating conditions. The pool also serves as a major source of water for cooling fuel assemblies in the event of a cooling system failure.

A spent fuel pool can accommodate a large number of spent fuel assemblies (from 300 to 600 depending on the type of reactor) containing radioactive substances. It is located in a fuel building (BK) with a dynamic confinement system ensured by the ventilation system.

The reactor cavity in the reactor building (RB) and the spent fuel pool in the fuel building (BK) are shown in Figure 6.1.

It was considered in the safety demonstration carried out at the time of design that, in the event of pool cooling system failure, and given the low residual heat of the spent fuel assemblies, there would be enough time to implement safety measures before the fuel was uncovered. That is why, in the event of loss of fuel rod integrity, the spent fuel pool building does not have the same radioactive confinement capability as the reactor containment building.

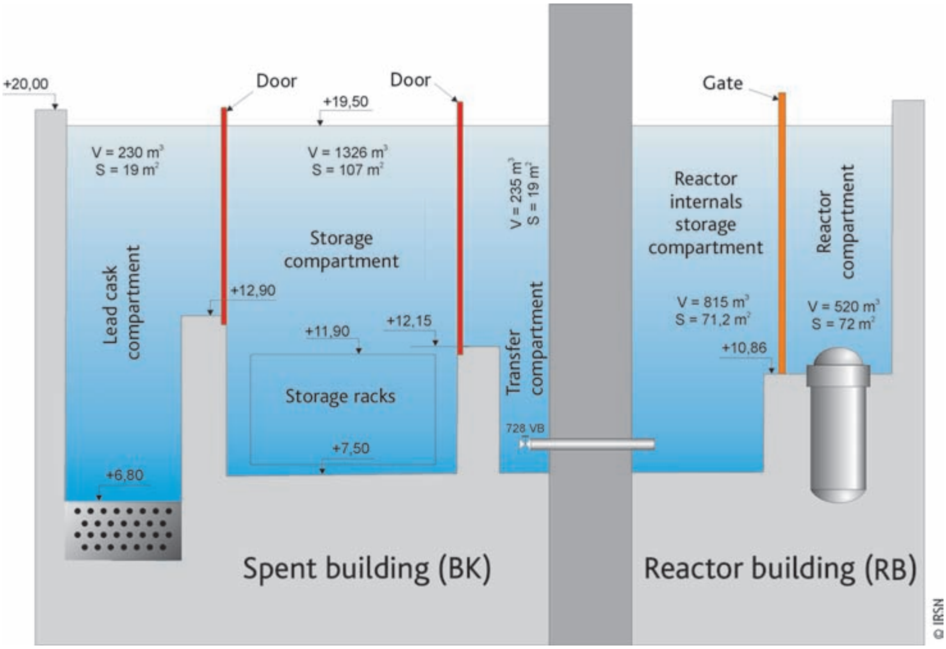

Figure 6.1 Cutaway view of the reactor cavity in the reactor building (RB) and the spent fuel pool in the fuel building (BK) in a 900 MWe CPY[48]-series plant.

Various design features, such as vacuum relief valves on the fuel pool cooling and purification systems (FPCPS), were implemented to preclude the risk of the pools emptying. These features were significantly reinforced following several incidents (loss of tightness of doors and gates, drainage induced by lineup errors, unsatisfactory performance in vacuum relief valves, foreign matter blocking these valves) or in connection with safety reviews.

The Fukushima Daiichi accident confirmed the importance of examining all the possible strategies for guaranteeing fuel assembly cooling, especially now that fuel and fuel pool operating conditions have changed considerably since design (fuels more highly irradiated, existence of MOX fuels).

If it becomes impossible to ensure sufficient fuel assembly cooling, the dreaded cliff edge effect is a runaway exothermic oxidation reaction in fuel cladding due to air and steam, leading to cladding deterioration and significant radioactive release. In this event, very little radioactive iodine-131 should be released, given the time the fuel is stored after removal from the reactor core. On the other hand, a very large quantity of ruthenium should be released. This is a particularly radiotoxic element, even if its radioactive half-life is considerably shorter than that of cesium (see Section 9.4).

48. Second serie of French power reactors of 900 MWe (EDF).

Boiling water in the storage racks could also cause the fuel assemblies in the rack to return to criticality. This could lead to rod failure, human radiation exposure and radioactive release.

Assessing the risks associated with loss-of-coolant or loss-of-cooling accidents entails modeling convection phenomena that are mixed (forced and natural); three-dimensional (owing to the shape of the storage racks and the non-uniform distribution of residual heat in the spent fuel assemblies in the pool); and involving multiple fluids (water, steam and air during the uncovery phase). Oxidation phenomena of zirconium alloys in contact with mixtures of steam, oxygen and nitrogen must also be considered [1]. These oxidation reactions generate heat and, possibly, hydrogen, with a related risk of explosion. Research work also focuses on the effectiveness of engineered safety systems according to the scenarios considered (for example, risk of pump cavitation if steam or air is entrained in the systems, effectiveness of spray systems for cooling uncovered fuel assemblies).

Although tests carried out in the 1950s already showed that the kinetics of zirconium oxidation by air at high temperatures differed considerably from that of oxidation by steam, air-induced oxidation of zirconium alloys only recently became the focus of detailed studies.

The zirconium oxidation reaction due to oxygen releases about twice as much heat (1101 kJ/mole compared with 528 kJ/mole when steam is the oxidizing agent). Zirconium also reacts with nitrogen. The resulting nitride can then react with oxygen and also release considerable heat (736 kJ/mole). If the conditions are suitable for initiating these chemical reactions (air in the storage racks and temperature around 600 °C), the heat generated could lead to runaway temperatures and damage to the fuel rods.

In France, IRSN carried out air-induced oxidation tests on Zircaloy-4 and M5™ cladding samples, a few centimeters in length, as part of the International Source Term Program (ISTP, 2005–2013, MOZART[49] test series). The studies were carried out in a temperature range between 600 °C and 1100 °C. A thermogravity system was used to continuously monitor the increase in sample weight, and thereby define oxidation kinetics laws. These were then incorporated in the ASTEC[50] (see chapter 9 on severe accidents) simulation code used to calculate the different stages of the accident.

Other analytical tests on different alloys were performed at more or less the same time in the United States and Germany. These differed in terms of the type of alloys studied, sample geometry (cladding sections with open or closed ends), the initial state of the materials (fresh or pre-oxidized and pre-hydrided to simulate the effects of in-service oxidation), and the type of oxidizing fluid studied (pure nitrogen, mixture of nitrogen and steam, or mixture of air and steam).

These tests, and those performed by the Karlsruhe Institute of Technology (KIT) in Germany, confirmed that cladding oxidation in air could lead to more serious consequences than oxidation by steam. This is because the kinetics of the oxidation reaction

49. Measurement of Zirconium Oxidation by Air at Temperature.
50. Accident Source Term Evaluation Code.

accelerates as from a certain oxidation rate. First, a dense layer of zirconium oxide is formed that limits oxygen diffusion to the metal. Its thickness increases with the square root of time (see Section 3.2). The kinetics then accelerates further when this dense oxide layer cracks. The air then reaches new areas of unoxidized metal, and the oxidation reaction proceeds in a linear manner with time. This phenomenon – known as breakaway – is not very well understood at the present time. One explanation might be the change in the zirconia crystal system (from monoclinic to tetragonal) at around 1000 °C (see Section 2.2). However, another hypothesis was put forward to explain the specific role of nitrogen and the formation of dense nitrided compounds (Zr-O-N or ZrN), the oxidation of which could cause the zirconium oxide layer to crack at temperatures below 1000 °C.

Integral tests were also carried out by Sandia National Laboratories in the United States. These were part of a program called "Spent Fuel Pool Heatup and Propagation Project" (2004–2006), set up to study BWR fuel assembly damage[51]. This was followed by the OECD/NEA "Sandia Fuel Project" (2009–2013), devoted to the study of PWR fuel assembly damage, bringing together 13 countries, including France. The purpose was to obtain thermal-hydraulic data on natural convection cooling of fuel rods by ambient air in the event of loss of cooling, in order to validate accident computer codes (such as ASTEC and a dedicated version of DRACCAR).

Experiments were carried out on a full-length, finely instrumented assembly comprising 17 x 17 Zircaloy-clad, magnesium oxide[52] heater rods. First, nondestructive tests were carried out for natural convection in air. These were followed by a destructive test that was carried out until the onset of runaway oxidation (also called zirconium fire) and its axial propagation within the assembly. In a second test campaign, the heated assembly was surrounded by four unheated assemblies to study the radial propagation of runaway oxidation to surrounding, lower power assemblies. Some of the rods in these unheated assemblies were also pressurized to simulate ballooning effects (possibility of oxidation accelerating after spalling of the oxide layers formed). It was seen that runaway oxidation did propagate to the surrounding assemblies. Following this research, core melt accident simulation codes, such as ASTEC, benefited from improved models for describing this phenomenon more accurately.

► Planned programs

In 2013, IRSN launched a research program, called DENOPI[53], to learn more about these phenomena. The following CNRS research laboratories also participate in the program: PROMES – Processes, Materials and Solar Energy, laboratory in Perpignan –, LEPMI – Laboratory of Electrochemistry and Physico-chemistry of Materials and Interfaces in Grenoble –, ARMINES-SPIN – Carnot M.I.N.E.S Institute – Center for Industrial and Natural Processes that is part of the École des Mines in Saint-Etienne; and LVEEM – Vellave Laboratory on the Development and Study of Materials in Le Puy-en-Velay.

51. Unlike PWR fuel rod assemblies, BWR assemblies are arranged inside Zircaloy channels.
52. Magnesium oxide is not radiotoxic and its thermal properties (conductivity, specific heat, etc.) are similar to those of uranium oxide.
53. Spent Fuel Pool Water Uncovering.

Scheduled to last six years, the program is jointly funded by the French National Research Agency (ANR), as part of the "Investment in the Future" program and, more specifically, the call for research projects in the field of nuclear safety and radiation protection (RSNR), issued in the wake of the Fukushima Daiichi accident in 2012.

Work scheduled as part of DENOPI includes experiments, modeling and simulation code validation, with the aim of learning more about the different phases of accidents of loss of heat removal or uncovery of fuel assemblies stored in the spent fuel pool. The program adopts an analytical approach seeking to expand knowledge in the following three areas:

- natural convection cooling at pool scale,
- thermal-hydraulic behavior at fuel assembly scale in the event of fuel uncovery, and effectiveness of spraying,
- acceleration mechanisms observed in fuel cladding oxidation in contact with an air and steam mixture.

Natural convection in the spent fuel pool will be studied (2015–2017) using a 1/5 model of a spent fuel pool, storage racks and the fuel assemblies they may contain. Results will also be interpreted using computational fluid dynamics tools designed to solve equations governing fluid movement.

In 2018–2019, the study of thermal-hydraulic behavior in a fuel assembly will use a full-length model of a fuel assembly and its storage rack, under conditions that represent the various phases of the accident, namely loss of cooling, pool water boiling, fuel uncovery, and resumption of cooling. Fluid velocities in the gaps between rods and oxygen concentrations will be measured to obtain very precise experimental data that will be fed into a database to validate the models used in accident calculations.

Cladding oxidation and the related runaway phenomena will be studied, together with the presumed role of nitrides, using high-performance laboratory technology including:

- *in situ* X-ray diffraction (XRD) in a steam-rich gas flow will be used for real-time monitoring, at reaction temperature, of changes in compounds formed at high temperatures (500 °C to 1200 °C);
- Raman micro-imaging will be used to study the microstructure and composition of the zirconium oxide layer formed.

Integral tests are also planned using simulated fuel assemblies in KIT's QUENCH facility as part of the Severe Accident Facilities for European Safety Targets (SAFEST) program, which is partly funded by the European Union.

Lastly, IRSN coordinates the AIR-SFP project (2015–2017), which is jointly funded by the European Union, following a call for research projects as part of NUGENIA+[54]. The

54. NUGENIA+ manages research for the development and safety of Generation II and III nuclear power reactors on behalf of NUGENIA (see Chapter 1). This association has more than a hundred members, including industrial companies, nuclear licensees, technical safety organizations, and research laboratories, for the most part in Europe.

purpose of this project is twofold: to assess the ability of different codes used in simulating reactor core melt accidents, such as ASTEC, to predict the different phases of an accident caused by fuel uncovery in a spent fuel pool; and to prepare a research program to fill the most significant gaps in knowledge. Fourteen countries are involved in this project, the results of which will lead to a better understanding of these accidents, and help to prevent them and mitigate their consequences.

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Chapter 7
Research on Fires

7.1. Fire risks at nuclear installations

Fire is a major risk to be taken into account for nuclear installation safety. On March 22, 1975 a cable fire broke out in Unit 1 of the Browns Ferry power plant in the US, as a result of a leaktightness test of a polyurethane foam cable sleeve through a wall, performed by candle. Although the operators involved immediately proceeded to put out the fire, unbeknown to them it spread along the cables on the other side of the wall. It caused a loss of control of some equipment important for the safety not only of Unit 1 but also of the neighboring Unit 2. The operators' reactions to this loss of control contained the incident before it could degenerate into a more severe accident. In October 1989, a turbine blade failure at the Vandellos power plant in Spain (UNGG-type reactor[55]) had multiple consequences: a hydrogen leak and explosion, turbine lubricating oil fire, loss of power and of compressed air for regulating several items of equipment involved in residual reactor heat removal, major basement flooding (including of the reactor building basement), etc. The fire, which lasted for more than four hours, was brought under control with the help of firefighting units from within a 100 km radius around the power plant. If the cooling system had stopped working, the 3000 metric tons of graphite would have caught fire. The reactor has not been restarted, mainly because of the high cost of the modifications necessary to improve safety.

55. Reactor running on graphite-moderated, gas-cooled natural uranium fuel (UNGG: Natural Uranium-Graphite-Gas).

Fires do happen from time to time in French nuclear reactors, with consequences of varying degrees of severity. Some examples are:

– an incident in July 1999 at Bugey nuclear power plant in which a single cause (an electrical fault in a board at the pumping station) was responsible for two electrical faults and two fires in Unit 3, which started almost simultaneously in different, geographically separate fire compartments;

– an electrical cable fire in 2004, caused by overheating in an opening between the turbine hall and the electrical building of Unit 2 at the Cattenom power plant;

– in 2012 an oil fire broke out in the reactor coolant pump of the reactor building for Unit 2 at the Penly nuclear power plant.

These examples demonstrate the vital importance of considering fire risks.

Fire is taken into account in the design of pressurized water reactors as both an internal and an external hazard. As a general rule, hazards of this kind should not lead to reactor accidents or endanger the operation of safety systems designed to manage accidents. As part of the defense in depth principle and the deterministic safety approach, controlling fire risk relies on preventive measures – particularly limiting the heating load in buildings –, on the detection of any fires that do start, and on measures to limit the consequences of fires, particularly by means of the compartmentation[56] of buildings and the installation of extinguishing systems. Moreover, the overall risk of core melt in which an internal fire is the originating event is assessed as part of special level 1 probabilistic safety assessments ("PSA Fire" developed by EDF and IRSN), which take account of scenarios where compartmentation elements fail. This type of assessment is all the more important given that reactor operation can reveal compartmentation anomalies or non-conformities (openings not properly sealed, etc.).

A good knowledge of all the phenomena that can come into play when fire breaks out in a room is therefore necessary: the heat level in the room, the increase in gas pressure in the room, the production of (burned or unburned) hot gases and smoke, transfer to other rooms (particularly if the compartmentation fails), etc.

Analysis of data from the OECD Fire database [1] shows that the majority of fires (around 50%) are caused by electrical sources. Electrical and electronics cabinets are one of the biggest contributors, causing more than 10% of fires, regardless of type (low,

56. Nuclear installations are designed so that fire can be contained within a defined perimeter and that inside this perimeter the consequences of the fire can be controlled. The perimeter is delimited by physical walls, or may even be separated by sufficient space. For the safety demonstration, it is assumed that all equipment within a perimeter on fire has been lost (i.e. is unavailable or has failed). The layout of the buildings, the delimitation of these perimeters and the installation of the equipment are all designed so that, if the equipment in the train of a redundant system is affected by fire, the equipment in the other trains cannot be affected by the fire.

Figure 7.1 Example of a cableway configuration in an installation. @ Georges Goué/IRSN.

medium or high voltage). Electrical cabinets and cableways (Figure 7.1) are therefore of specific interest as regards research on fire risks, for several reasons:

- they can cause fires to start;
- they can help fires to spread;
- conversely, they can also be "targets" if a fire breaks out, so their vulnerability should be reduced as far as possible if they are important for reactor safety.

Most of the research on fire aims to assess the characteristics of the fires themselves, particularly estimating heat output, and the consequences for an installation. The consequences of particular interest are the effects of gas pressure and temperature in rooms, transfers of gases (produced by combustion or unburned) and of soot between rooms, through doors or openings, or through ventilation systems, and the effects of these transfers on containment systems and electrical equipment.

Studying the development of fire inside a nuclear installation and its consequences for equipment important for safety is particularly complex because of the confinement of the rooms and the presence of mechanical ventilation. The oxygen concentration in a room on fire decreases then stabilizes, generally at a value where the flow of oxygen being consumed by the fire is balanced by the flow being drawn in by the ventilation. The degree of confinement of installations, the characteristics of ventilation systems (air renewal rate in the rooms, air flow resistance) and how they behave in a fire (shutdown, closure of fire dampers) are decisive in determining the heat output of the fire, how long it burns for and how it spreads to other fire sources, or even to other rooms.

Since the early 1980s, OECD/NEA/CSNI documents [2, 3, 4] have reported international concerns about the safety of power reactors and evolving knowledge of fire risks (including as part of the development of probabilistic safety assessments). Since 2006, IRSN has run programs, particularly the international PRISME[57] projects guided by the OECD/NEA, to improve knowledge of fires in confined, ventilated spaces representative of nuclear installations.

7.2. Organizations involved in research on fire

Research on fire is not unique to the nuclear sector. In France, there are many different organizations conducting research and development related to fire. In particular, the following four university laboratories are helping to run CNRS' fire research group:

- IUSTI[58] (UMR[59] 6595) in Marseille (compartment fires and forest fires),
- P-PRIME (UMR 9028) in Poitiers (combustion of solids, fire and smoke in open and confined spaces),
- CORIA[60] (UMR 6617) in Rouen (metrology of soot),
- LEMTA (UMR 7563) in Nancy (thermal radiation, fuel emissivity measurements).

IRSN, EDF and the French General Directorate for Armament (DGA) are also carrying out research in this field (there are similarities between fires on ships and submarines and nuclear fires), as are French technical centers such as CNPP[61], CTICM[62], INERIS[63], CERIB[64] and CSTB[65], which are conducting technological tests, sometimes on a large scale, particularly for EDF.

The research begun more than 20 years ago by IPSN, then IRSN, concerns fires of internal origin in confined, ventilated rooms similar to those found at fuel cycle laboratories and plants (glove boxes, etc.) and in nuclear reactors. However, we will limit the following discussion to research looking specifically at nuclear reactors.

The experimental resources available to IRSN enable it to run a complete study of a fire in two stages: in the first stage, its main characteristics (heat output of the fire, combustion heat, pyrolysis rate, combustion products, etc.) are determined in an open atmosphere in a device known as a cone calorimeter. Then, in the second stage, tests are run in ventilated rooms representative of those found in nuclear installations. These tests are used to assess the effect of confinement and mechanical ventilation on the development of a fire.

57. Spread of a Fire for Multi-Room Elementary Scenarios.
58. French University Institute for Industrial Thermal Systems.
59. Joint Research Unit.
60. French Aerothermochemistry Research Complex.
61. French National Center for Prevention and Protection.
62. French Industrial Technology Center for Construction in Metal.
63. French National Institute for the Industrial Environment and Risks.
64. French Concrete Industry Study and Research Center.
65. French Construction Science and Technology Center.

IRSN is conducting or has conducted research with various partners: industrial partners (AREVA, EDF, ENGIE-Tractebel Engineering, Vattenfall, etc.), universities (Aix-Marseille, Rouen, Edinburgh, Ghent, Maryland, Lund, Aalto), research bodies (CNRS, INERIS, DGA, LNE[66], etc.) and international organizations (BelV[67], GRS[68], HSE[69], NRA[70], VTT[71], CNSC[72], CSN[73], etc.).

In addition, ETIC[74], a "virtual" laboratory for studying fires in confined spaces, was established in 2010 as a joint venture between IRSN and IUSTI (University Institute for Industrial Thermal Systems, a joint research unit CNRS/Universities of Provence and of the Mediterranean).

7.3. Research facilities, simulation tools

At Cadarache IRSN has the GALAXIE experimental platform, which offers experimental facilities of various capacities. The platform was originally built to conduct research in the 1970s to 1990s on sodium fires, sodium being a coolant used for the PHENIX and SUPERPHENIX fast neutron reactors. The GALAXIE test facilities were modified in the late 1990s and new experimental equipment was added from 2000 to adapt it for research into conventional fires in fuel cycle installations and France's nuclear power reactor fleet.

The GALAXIE platform consists of:

– DANAIDES[75], an installation for performing analytical tests on the separate and combined effects of heat and soot on the operation of different electrical devices (relays, circuit-breakers, etc.);

– a 0.3 MW cone calorimeter for measuring the combustion heat of different materials in open air, and a radiant panel enabling equipment to be subjected to a fixed flow of heat, in order to study its degradation; together the devices constitute the CARINEA facility;

– a larger-scale 3 MW cone calorimeter in the SATURNE tower (2000 m^3), for studying the combustion in open air of equipment from nuclear power plants, such as electrical cabinets and electrical cableways;

66. French National Laboratory for Metrology and Testing.
67. Belgian Federal Agency for Nuclear and Radiological Inspections of nuclear installations (hospitals, universities, radiological facilities, etc.).
68. Gesellschaft für Anlagen – und Reaktorsicherheit (reactor safety organization, Germany).
69. Health and Safety Executive (UK).
70. Nuclear Regulation Authority (Japan).
71. Technical Research Center (Finland).
72. Canadian Nuclear Safety Commission.
73. Consejo de Seguridad Nuclear (Spain).
74. Laboratory for the Study of Fire in a Contained Environment.
75. Analytical Equipment for Studying Electrical Malfunctions caused by Soot during a Fire.

– a controlled-atmosphere calorimeter, CADUCEE, for studying the effect of oxygen depletion on the combustion of different types of fuel, the heat radiated and soot production;

– the 400 m^3 PLUTON box connected to a ventilation system, which can be used for making large fires (up to 5 MW) with different ventilation configurations. The HYDRA device (2.4 m high, 3.6 m long and 2.4 m wide) is currently installed in this box and can be used to study soot movements using laser velocimetry through the opening of a door for different ventilation configurations. Small-scale devices such as NYX and STYX can also be installed in the box; they are designed for studying flows of smoke through openings or doors;

– the DIVA device consists of three rooms of 120 m^3, a 150 m^3 corridor and a first-floor room of 170 m^3, connected to a ventilation system with variable configurations; the device can withstand negative pressures and overpressures within the range − 100 hPa to + 520 hPa.

Figures 7.2-a and 7.2-b show the how the test facilities fit into the different topics being studied; Figure 7.2-c shows the DIVA facility.

All these devices are fitted with significant amounts of instrumentation (up to 800 measuring channels in the case of DIVA) for measuring the main characteristics of fires (temperature, pressure, gas concentrations [combustion and pyrolysis products], soot concentrations, total and radiative flows to the walls) and to take samples to be analyzed after the test (soot composition and granulometry). Video recordings are also made during the tests.

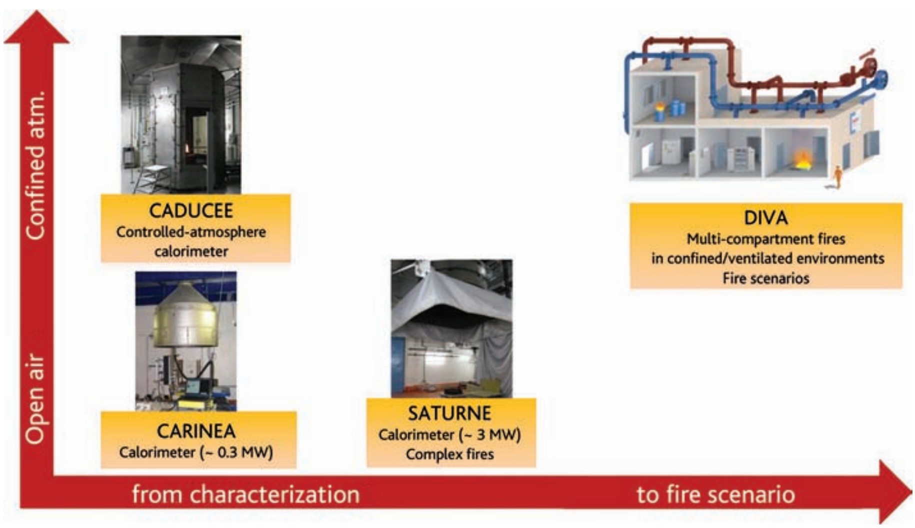

Figure 7.2-a Positioning of the different types of fire characterization test in relation to the test facilities. @ Laurence Rigollet/IRSN.

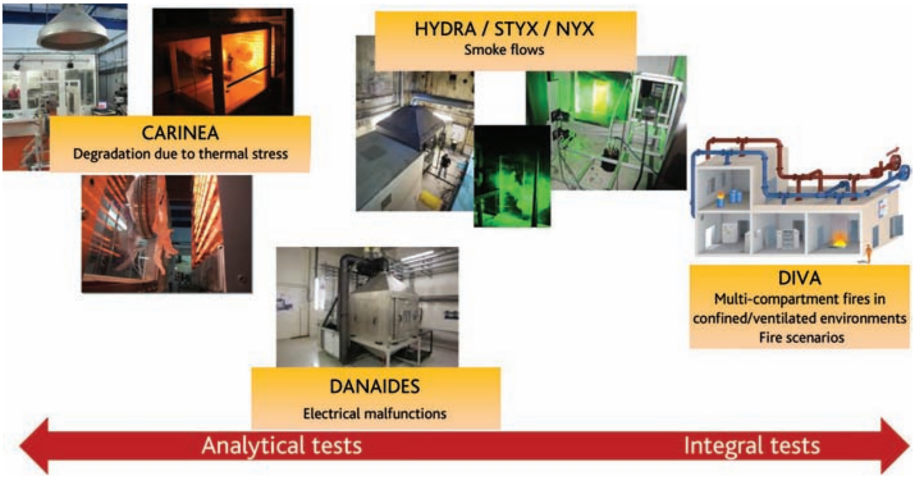

Figure 7.2-b Positioning of the different types of "target" behavior test in relation to the test facilities. @ Laurence Rigollet/IRSN.

As in other nuclear safety fields, assessing the consequences of a fire in a nuclear reactor requires the use of simulation codes incorporating models developed and validated on the basis of tests. Their capacity to simulate real fires in the configurations found in nuclear installations is verified by means of comparisons with large-scale tests performed in facilities reproducing these configurations (confinement, ventilation) as closely as possible. Obviously these tools are essential for analyzing the risks associated with fire and for studying fire scenarios to support IRSN's development of its own "PSA Fire" probabilistic safety assessments for France's nuclear power plant reactors.

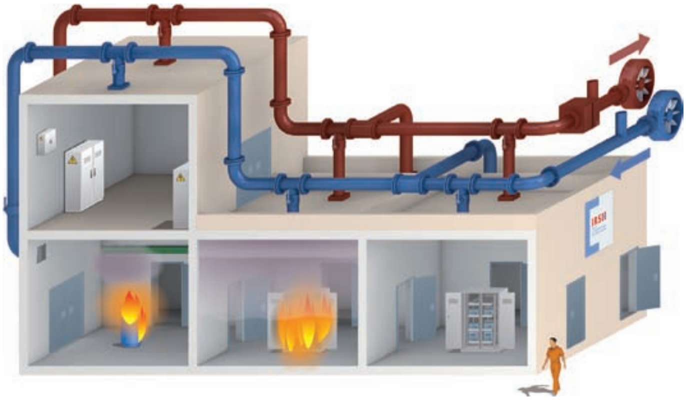

Figure 7.2-c The DIVA installation. © IRSN.

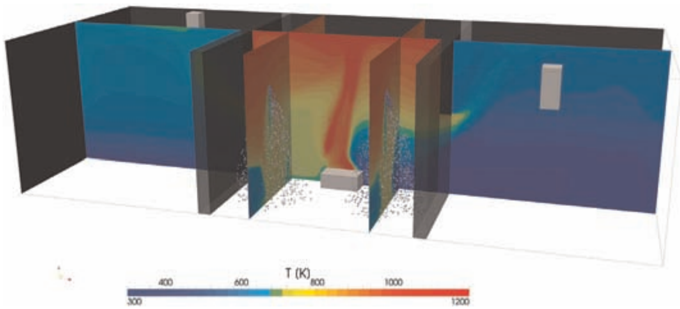

Figure 7.3 Three-dimensional temperature mapping resulting from the simulation of a test at the DIVA facility. © IRSN.

IRSN is developing and validating two types of simulation tool:

– the SYLVIA[76] code, which models the burning room as two homogeneous zones with a boundary between them (a flat horizontal surface) that moves over time. The rooms are connected to one another by doors (possibly fire doors), which are open or closed (leaks are modeled), and by the ventilation system. The ventilation system can be modelled in its entirety, with fire dampers, HEPA[77] filters, control dampers, fans, etc. Correlations of the mass and heat exchange between the zones, flames and walls complete the mass and energy balance equations for each zone. Because of its short calculation times, this code is used by IRSN for studies in support of its safety expertise and for its probabilistic safety assessments of fire;

– the ISIS (CFD-type) code, which models three-dimensional, non-stationary, marginally compressible, turbulent, reactive or chemically inert flow fields; it can be used to calculate combustion, thermal transfers and soot transport in large rooms, either ventilated naturally or confined and mechanically ventilated. Only the intake and extraction branches of the ventilation system are modeled (see Figure 7.3 for an illustration of the results achieved with ISIS).

IRSN has coupled the SYLVIA and ISIS codes together, giving it the benefit in a single tool of the precision of ISIS, which performs 3D simulations of fires in rooms, and SYLVIA's ability to describe a whole installation with a complete ventilation system connecting all the rooms. No other coupling of this kind exists anywhere else in the world.

7.4. The main research programs and their contributions

Major progress has been made since 1990 in terms of knowledge of confined, ventilated fires and how to model them, in particular due to research programs run

76. Software System for Analysing the Ventilation of a Fire and Airborne Contamination.
77. High Efficiency Particulate Air.

in collaboration with AREVA on fuel reprocessing plant safety (FLIP[78] program on solvent fires and PICSEL[79] program on electrical cabinet fires). Further progress has also been made through the international PRISME and PRISME 2 programs (2006–2011 and 2011–2016), run by IRSN under the aegis of the OECD/NEA, focusing on nuclear reactor safety.

These programs have been used to validate the ISIS and SYLVIA codes, which are able to a sufficient degree of precision of simulating scenarios involving fire in a confined environment with managed ventilation (shutdown of the air supply followed by air extraction after a certain period of time).

During the large-scale tests, significant variations in gas pressure in the rooms (overpressure when the fire starts, negative pressure when it is extinguished and, in some configurations, wide oscillations) have been observed. These gas pressure variations can subject compartmentation systems (fire doors and dampers) to stresses beyond those they were designed for, enabling fire to spread within an installation. This gas pressure variation is linked to the fire's confinement and the ventilation system's resistivity. The oscillations and instability of combustion are due to the under-oxygenation of the fire, leading to pyrolysis gas combustion. These phenomena could also occur in a nuclear power plant. The tests have also identified the effect of soot on the operation of electrical or electronic equipment.

The PICSEL program run in collaboration with AREVA between 2004 and 2011 studied fires in electrical cabinets and their consequences in experiments in the SATURNE facility (fires in open atmospheres) and the DIVA facility (fires in confined, ventilated rooms), enabling them to be modeled. These results, obtained during a program looking more specifically at the configurations of fuel cycle installations, can be transposed to nuclear reactors.

One of the findings of this program on electrical cabinet fires (Figure 7.4) concerns the heat output of these complex fires (involving multiple components and many different types of material). In particular, the tests showed that the heat output of an electrical cabinet fire with the cabinet doors open was 10 times greater than that of an electrical cabinet fire in which the doors were closed. This difference is due to the fact that soot clogs the door vents, preventing they entry of oxygen and therefore combustion, in the electrical cabinet with its doors closed. These are the first electrical cabinet fire tests run in a confined, ventilated atmosphere; a few tests had been done previously, at the Sandia National Laboratories (SNL) in 1987 and by VTT in 1994, but they had only evaluated the heat output of this type of fire in an open atmosphere.

The PRISME program consisted of 24 tests run in the DIVA facility, plus a further 13 more analytical tests in SATURNE. It produced results on the propagation of smoke and hot gases in the adjacent rooms to a burning one, on how long it would take for the cables in the burning room to malfunction, on how the fire dampers in the ventilation system would work and on how the ventilation system should be managed to prevent pressure effects that might damage the compartmentation system. The PRISME program

78. Interaction of Liquid Fires with a Wall.
79. Propagation of Solid Fuel Fires in Laboratories and Factories.

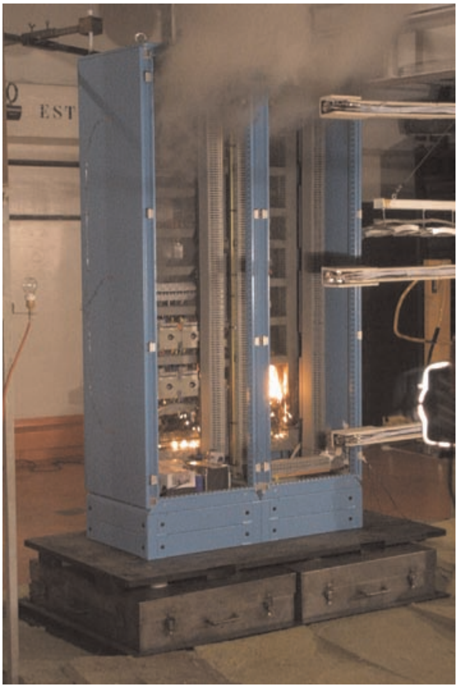

Figure 7.4 Fire in a type of electrical cabinet used in fuel cycle installations, tested at the SATURNE facility as part of the PICSEL program. © Florent-Frédéric Vigroux/IRSN.

provided a better understanding of the effect of ventilation (and therefore under-oxygenation of the fire) on the heat output from a fire breaking out in a confined, ventilated room, and particularly of how long the fire would last. Depending on the air renewal rate, the fire could rapidly go out because of a decrease in the oxygen concentration in the burning room. But the tests run under thePRISME program showed that an equilibrium could be reached between the air coming from the ventilation air supply and the heat output of the fire: in this case, any combustible material would be consumed more slowly than in the open air (lower heat output) and more completely. For example, the same fire could burn 2.5 times longer in a ventilated room with an hourly air renewal rate of 4.7 than in open air. Correlations and analytical models of pyrolysis in a confined, ventilated environment (under-oxygenated fires) have been developed and validated. These models correct the heat output obtained for a fire in open air (measured under the SATURNE hood) by a factor taking into account the under-oxygenation rate of a fire in a confined space.

The PRISME program was also used to quantify the effect on smoke propagation of "mixed" convection, which combines the forced convection created by ventilation with the natural convection induced by the steep vertical temperature gradient of the gaseous atmosphere in a room ("fresh" air on the ground and hot smoke at the ceiling). The mechanical ventilation of the burning room can significantly alter the gas flows that would naturally develop at an open door between two rooms. Depending on the way the ventilation system is adjusted, the mechanical ventilation helps to cause an imbalance in the flows entering and leaving the burning room, and to change the position of the neutral plane (the height where flow rates are zero due to flow reversal) at the door.

All the data collected during this program has been used to assess the ability of computer codes to simulate different fire scenarios. The new models, particularly for pyrolysis, have been integrated into the various partners' codes and validated by means of experimental data obtained during this program.

The PRISME 2 project (2011–2016) was launched for the purpose of studying fire-related themes to complement those studied in the first project (PRISME), such as smoke propagation between rooms on top of one another, the fire propagation between cableways and the performance of sprinkler systems. The PRISME 2 program consists of 22 tests at the DIVA facility and approximately 20 tests in the SATURNE calorimeter.

The tests, involving a simple fire (a pool of liquid) in a ventilated room which is either closed or is connected to an upper room by an opening, revealed a phenomenon of high amplitude and very low frequency oscillations, of the thermodynamic variables in the facility, which suggest combustion instability that is highly correlated with the room's ventilation and with oxygen transfer from the air supply vent to the combustion zone. Understanding these oscillations required further experimental research to identify the configuration and scenario parameters causing them. Results have been obtained for the transfer of smoke in these configurations in relatively stable combustion regimes (i.e. without wide oscillations), providing new data for validating the correlations for smoke transfer through a horizontal opening between closed, ventilated rooms and for validating detailed CFD-type codes.

Several types of cable were tested during the tests in DIVA (following their characterization in an open atmosphere under the SATURNE hood): cables with a halogenated flame retardant and cables with a non-halogenated flame retardant (components reducing the spread of a fire affecting an electrical cable). The integral tests in DIVA revealed that the propagation of fire along five horizontal cableways or from an electrical cabinet to the cableways above it depended heavily on the type of cables and the air renewal rate in the room on fire. The intensity and duration of the fire in a confined, ventilated environment could not be correlated with the degree of fire resistance of the cables. So in certain conditions, supposedly fire retardant cables burned up completely in a fire lasting a long time whereas supposedly non-fire retardant cables caused the fire to go out early on due to a lack of oxygen (in this case, the mass of burned cable was low as a result of the simultaneous consumption of oxygen by the fire in the cabinet and the fire in the electrical cables). This behavior, which may seem paradoxical, is explained simply by the fact that if the runaway of a fire in a confined environment is too sudden, this can cause a drop in the oxygen concentration, which is not compensated for by the air supplied by the ventilation. This produces conditions in which combustion at the flame front stops. These tests also revealed that the sudden reignition can occur (slow deflagration but with a significant peak in gas pressure) of unburned gases that have accumulated in the room. The results are new and original, and underline the importance and value of continuing to study the combustion of cableways in confined, ventilated facilities for the development of combustion models (simplified, such as correlations, and more detailed, such as porous environments for CFD codes), taking account of the specific characteristics of the cableways in question (cable type, size, spacing between cableways, etc.).

The PRISME 2 program also provided data on the use of extinguishing systems. During these sprinkler tests, contact between the droplets and the flame zone above the tray of burning oil and walls was avoided so that the effect of the sprinkler on controlling the fire could be studied without directly extinguishing it and avoiding the edge effects associated with the walls. This data can be used to validate simulation codes and to assess the codes' ability to simulate the phenomena identified during the tests, e.g. cooling of the gases in the room due to contact with the water droplets and their vaporization, vigorous mixing of the gases in the room producing a uniform gas distribution and therefore temperature and oxygen concentration and a significant increase in combustion of the burning liquid (measured by loss of mass by the pool of liquid over time).

Industrial protection aimed at preventing the spread of fire along a cableway was installed and tested under the SATURNE hood on a set of three cableways, one on top of the other (Figure 7.5). The fire did not cross the barrier created by this protection, but the demonstration at this stage is not fully established because there was some pyrolysis of the cables downstream of the protection and the flames could have crossed the barriers if there had been more cableways present or if the fire had been in a confined environment so that the cables downstream of the protection had been preheated.

The speed of propagation along inclined cableways was measured and compared with the speed along horizontal cableways.

Figure 7.5 An experiment involving a fire in a set of cableways, one on top of the other. © Florent-Frédéric Vigroux/IRSN.

A test of fire propagation from an electrical cabinet with open doors to adjacent cabinets and to cableways passing above the electrical cabinets was performed in the DIVA facility. The test revealed that the fire spread to one of the cabinets adjacent to the burning cabinet but that the spread of the fire was less intense than during a similar test in a previous campaign, without cabinets adjacent to the electrical burning cabinet.

The results of the tests are being analyzed by the international bodies involved in the PRISME and PRISME 2 programs (Germany, Belgium, Canada, Spain, Finland, France (IRSN, EDF, DGA, Marseille University), Japan, Sweden, the UK, South Korea, the USA, and the Netherlands, though the last three only for PRISME). Code benchmark exercises have been organized by IRSN in the context of a working group related to the PRISME and PRISME 2 programs: the various partners have compared the results of fire simulations with the experimental data. A sensitivity study carried out with six different simulation codes, involving six input parameters (heat release rate of the fire, radiative fraction of the flame, thermal properties of the walls, etc.) revealed that the fire's heat release rate is always the dominant parameter, showing that efforts to improve its modeling need to continue.

In another example of assessment of the relevance of simulation codes, the CNRS fire research group organized a benchmark exercise of multi-dimensional computer codes that use fields (ISIS code mentioned above, SATURNE code developed by EDF, and FDS[80] code developed by NIST[81] in the United States), based on a fire test in a hotel room. The exercise showed that results produced by the different codes were very widely dispersed as regards the values measured during the test. Experts attributed this dispersal in particular to the difficulty of choosing an appropriate combustion model for calculating the fire's instantaneous heat release rate.

These results demonstrate the need to continue the research in order to achieve better modeling of the characteristics of fires by developing pyrolysis and combustion models in under-oxygenated and complex fire conditions. It is also necessary to increase knowledge particularly of the instability of combustion as a result of the production and ignition of unburned gases, as observed during the study of cable fires in an under-oxygenated environment[82] in the PRISME 2 program, and of about the sudden pressure variations caused by this, so that the predictive capabilities of the simulation tools can be improved. The layout of cableways (horizontal, vertical, mixed, near a wall, etc.) and the cables they carry (loose, tight, etc.), as well as the type of cables, constitute a set of parameters with effects on flammability and the propagation of fire that are complex to model and require new experimental data. The effectiveness of measures to protect cables from catching fire (wrapping in fire-resistant tape, enclosure, etc.) is also worth studying experimentally. The first tests on fire-resistant taping were run during the PRISME 2 program. They acquired data on the fire resistance of this taping under realistic conditions.

80. Fire Dynamics Simulator.
81. National Institute of Standards and Technology.
82. Situation arising as a result of a confined, ventilated atmosphere; the phenomenon can also occur as a result of the accidental heating of cables, e.g. through the Joule effect, causing the release of pyrolysis gases.

In addition to these research programs using integral tests, academic research is also being carried out on the modeling of pyrolysis, a phenomenon strongly linked to conditions around the fire, soot production and radiation from the flames. The models, developed particularly at the ETIC joint research laboratory, have been validated in medium-scale tests, especially in the CADUCEE controlled-atmosphere calorimeter. For this validation, appropriate instrumentation was developed to cover different aspects, particularly flowmeters and non-intrusive methods using a laser (LII[83], PIV[84], etc.). Laser velocimetry techniques and PIV (visualization of the movement of particles in a moving fluid) can be used to find out the velocity fields of flows and to measure the local intensity of turbulence. These measurements are essential to validate the airflow calculations and therefore the calculations for smoke and heat transport within the rooms. The PIV technique has been used to measure the velocity fields of flows through a door in the HYDRA device and at an opening in the STYX device.

Research being conducted on fires is looking at the effects of a fire on equipment important for safety, in particular compartmentation components such as doors and fire dampers, and electronic and electrical equipment.

As far as the compartmentation components are concerned, airflow tests are performed in IRSN's STARMANIA facility at the Saclay research center. In particular this allows measurement of the effects of pressure on fire protection components (doors, dampers).

Electrical equipment malfunctions due to heat and soot were studied during the analytical test programs CATHODE (2007–2009) in an oven named SCIROCCO, then DELTA (2014–2015) using the DANAIDES device (Figure 7.6); the criteria deduced from these tests can be used to predict the operational limits of the tested equipment[85] when a fire breaks out in the room where it is located. The tests have identified three general zones (in terms of temperature and soot concentration): one zone where the equipment remains operational, one zone where malfunctions occur that are reversible, and one zone where the malfunctions are irreversible. During tests on circuit boards, the damage from soot appears to be significant from a soot concentration of 1.5 g/m^3, reducing the zone in which the equipment functions properly.

EDF is also performing analytical tests on the MILONGA experimental platform at the Chatou center. In particular these tests consist of taking measurements using a small cone calorimeter coupled with a Fourier transform infrared spectrometer and an electrical low pressure impactor, of the gases and soot released by the combustion of different materials, particularly those used in cables. Malfunctions of electrical and electronic equipment exposed to heat and smoke are also being studied in the MAFFé

83. Laser Induced Incandescence. LII is used to measure soot concentration and to validate soot production models in an under-oxygenated environment. This measurement technique is not currently used in fire tests because it is the subject of academic research.

84. Particle Image Velocimetry.

85. Tested equipment: D125 circuit-breakers, MICOM P921 and VIGIRACK A326E electronic relays, LOREME.

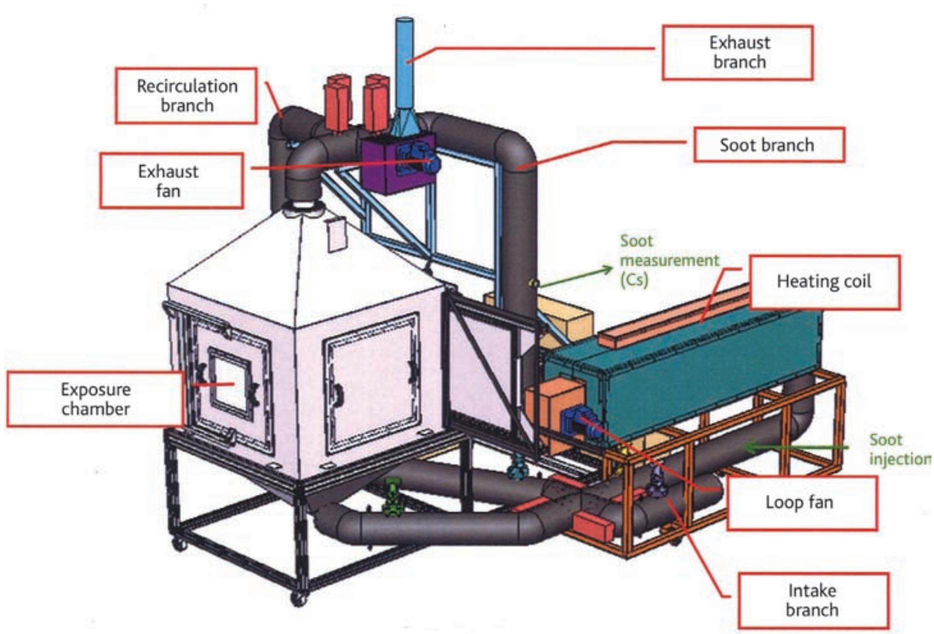

Figure 7.6 The DANAIDES facility used for the DELTA program. @ Marc Piller/IRSN.

furnace. The results of these tests are used by EDF in its safety demonstrations, subject to assessment by IRSN.

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Chapter 8
Research on Hazards associated to Natural Events

Taking natural events into account in the design of nuclear installations and during their operation is essential for their safety. The general objective is that the safety functions associated with their structures, systems and components are not harmed by such events. Scientific knowledge in the earth sciences field (such as geology, hydrogeology, seismology, etc.) forms the basis for assessing the risks associated with natural events, i.e. the characteristics to be taken into account in the design or testing of installations (seismic movements of the ground, water level, etc.). Improving installation safety particularly involves working to improve the reliability of our ability to grasp the mechanisms at work in natural phenomena and their effects. This is the general objective of the research initiatives run in the last forty years or so by IPSN (then IRSN), particularly as regards seismic risks.

Natural events, the subject of research and development by IPSN (then IRSN) either on its own or in the context of collaborations, are mainly earthquakes, floods, and climate-related hazards. This work aims to develop, improve and validate the tools (including databases) and methods used so that the hazards and their consequences can be determined more accurately. Field studies, one-off experiments and ongoing instrumental measurements (seismographic network and GPS[86]) complement the theoretical research and modeling.

As the flood at the Le Blayais nuclear power plant site in December 1999 and the accident at the Fukushima Daiichi nuclear power plant in March 2011 showed, nuclear

86. Global Positioning System.

safety can only be guaranteed at the cost of adequate protection of facilities against all types of natural events, which means that the natural events that could strike them need to be evaluated correctly, particularly those associated with seismic events and the occurrence of floods. In Japan, the Tohoku earthquake and the resulting tsunami, and also the earthquake in July 2007 at Chūetsu-oki not far from the Kashiwasaki-Kariwa nuclear power plant[87], and to a lesser extent the earthquake in Virginia (USA) in August 2011, around 18 km from the North Anna nuclear power plant[88], highlighted the importance of the knowledge and methods on which the design of nuclear installations is based and but also their limits. There is a broad international consensus on the need to extend the knowledge and assessment of natural events that can seriously affect nuclear sites – as evidenced by the conclusions of an international conference held by the International Atomic Energy Agency (IAEA) in March 2012. This is particularly true in France, where improving the way seismic risks and flood risks are taken into account is one of the priorities set by IRSN, ASN and the public authorities as a result of feedback from the Fukushima accident, reflected in the call for RSNR projects issued by the French National Research Agency (ANR) with funding from the Investment in the Future Program (PIA). In this new context, there are two issues for research and development when it comes to defining extreme hazards. One is obtaining knowledge of the phenomena (including historical indices), and the other is finding methods for taking account of the associated uncertainties and reaching a definition of the maximum risks to be taken into account when designing nuclear installations.

8.1. Earthquakes

Seismic hazard assessment practice in France for installations at special risk (nuclear and chemical installations, dams, etc.) has historically been based on a deterministic approach. This approach estimates the acceleration that would be produced by the strongest historical earthquake as close as possible to the site, with an added safety margin[89]. This margin is defined by increasing the magnitude[90] of the earthquake in question by a half-degree.

In general, seismic hazard is assessed by identifying possible aggressive earthquake "sources" and characterizing their potentiality. In low-seismicity areas where little is known about faults, the deterministic approach considers regions as possible sources of earthquakes. These regions are defined according to their geological or seismic homogeneity. In regions with slightly higher seismicity (e.g. Provence, Alsace, Pyrénées), more is usually known about faults and, most importantly, significant earthquakes can be

87. This nuclear power plant was not damaged.

88. Shallow (6 km deep) earthquake of moderate magnitude (5.8), which was unexpected given the historical seismicity of the part of Virginia concerned.

89. The strongest historical earthquake is known as the MPE (Maximum Probable Earthquake); with the margin added it is known as the SME (Seismic Margin Earthquake).

90. The magnitude of an earthquake is a measurement of the amount of energy released at the hypocenter of the earthquake. The magnitudes usually used these days are expressed as moment magnitudes (denoted M_w).

associated with these source faults (by their cartographic position). Nevertheless, seismic hazard assessment is part of a larger process requiring answers to the following questions:

- how big are the faults capable of producing earthquakes? The magnitude of an earthquake and, therefore, the danger it represents, is directly related to this parameter. The size and extent of an active fault can be determined by surface mapping, locating microseismic events, and depth imaging. This parameter can also be assessed by characterizing major "prehistoric" earthquakes;
- what is the deformation rate of the region affected by the faults considered? This is an important point for determining the periodicity of major earthquakes. In addition to dating geological markers, spatial geodesics measurements (GPS, InSar[91]) also provide useful data for assessing the deformation rate.

The seismic hazard associated with a system of active faults can be assessed by conducting research and development to find answers to the above questions.

IPSN, and later IRSN, pioneered many aspects of seismic risk research, such as studying historical seismicity with the SisFrance database, analyzing deep geophysical data, paleoseismology, and seismic mapping and monitoring of active faults, in particular with studies of the Moyenne Durance fault system, characterized by regular seismic activity (reference document [1], 2007). The results obtained by these studies have been used to characterize the microseismicity of the fault system and to produce a more precise assessment of seismic risk (particularly in terms of magnitude). This will be particularly useful for assessing or reassessing the seismic risk of the nuclear installations at CEA's center in Cadarache.

Over the years, the research programs run have enabled IPSN (then IRSN) to develop a network of scientific partnerships both in France and abroad. In the last few years, IRSN's accumulated expertise has enabled it to benefit ANR funding and has led to requests for assistance with site studies from a number of countries wishing to develop a nuclear power generation fleet (or strengthen the fleet they already have). Its activities have also gradually contributed to the definition of methods applicable to basic nuclear installations, notably those laid down in the French Basic Safety Rule 2001-01. The knowledge and data produced also feed into the expert appraisals performed by IRSN in the frame of safety reviews led by the operators. They have also been widely used in the complementary safety evaluation (ECS) carried out in the wake of the Fukushima Daiichi accident, in the presentation of the French report during the European peer review process, and in reports on the "hard core" being introduced particularly at French nuclear power plants (equipment that must withstand more severe hazards than the ones for which the installations were designed). As a result of the ENSREG[92] recommendations, a probabilistic approach to assessing risks with a low probability of occurrence has been introduced in addition to the deterministic method.

IRSN has also used the knowledge obtained on seismic risk in its role as expert supporting the French Ministry of the Environment during preparation of the seismic risk

91. Interferometric Synthetic aperture radar.
92. European Nuclear Safety Regulators Group.

mapping of metropolitan France, which came into force on May 1, 2011. This new "zoning", which covers conventional buildings and "special risk" installations (apart from nuclear installations and dams), is based not only on historic seismicity, which was essentially the case with the previous zoning in 1991, but also on a probabilistic assessment of seismic risk (estimation of the seismic movement likely to happen or be exceeded according to a fixed probability, typically 10%, over a time period of 50 years). However, to comply with the recommendations of French basic safety rules 2001-01, for nuclear installations IRSN regularly updates a "seismotectonic zoning", mainly to incorporate knowledge acquired through research in general. Also in relation to the recommendations in French basic seafety rules 2001-01, IRSN was involved in the creation of a national database[93] of indices of strong earthquakes that occurred in France a very long time ago (several thousands to several tens of thousands of years), known as paleoearthquakes. In the context of a memorandum of understanding with French General Directorate for Risk Prevention (DGPR[94]), IRSN also recently developed a database of active faults in a 50 km perimeter around any nuclear installations.

IRSN's research on seismic risk is also looking at:

- identifying and understanding how faults function (in France and abroad). The work aims to identify faults likely to cause earthquakes (active faults) and to characterize them (location, geometry, deformation rate). It is based on the analysis of records and on the study of historical seismicity and paleoseismic indices. Characterization of the activity of a fault (i.e. its ability to cause an earthquake) and the discovery of new active faults in France are necessary to form the basis, confirm or improve assessments of seismic risk. This research combines various different approaches (study of morphological anomalies using digital terrain models, Spot images, spatial imaging methods, geological studies of terrain, and trenching studies to find traces of paleoearthquakes). Since the study of the Moyenne Durance fault system mentioned above, IRSN's research has continued in France on the Vuache fault (Jura), which in particular was responsible for the earthquake that caused damage in Annecy in 1996, and on other faults in the Alpine Foreland, the Rhine Graben and the Channel and abroad (Ecuador, India, Spain);
- predicting potential seismic movements: this work is done using empirical approaches (based on observations) and numerical simulations;
- estimating the seismic response of the ground specific to a site (which comes under the term "site effects" [Figure 8.1]). The research has revealed the particular importance of "site effects", which can significantly amplify seismic movements in the frequency range of interest for concrete structures[95]. The research done in this field in particular requires the acquisition of geotechnical and geological data at the sites and the use of complex modeling that takes account in particular of

93. See the website www.neopal.net.
94. French civil service department for risk prevention.
95. A few Hertz. "Site effects" can reduce ground movements at the highest frequencies and amplify them at lower frequencies.

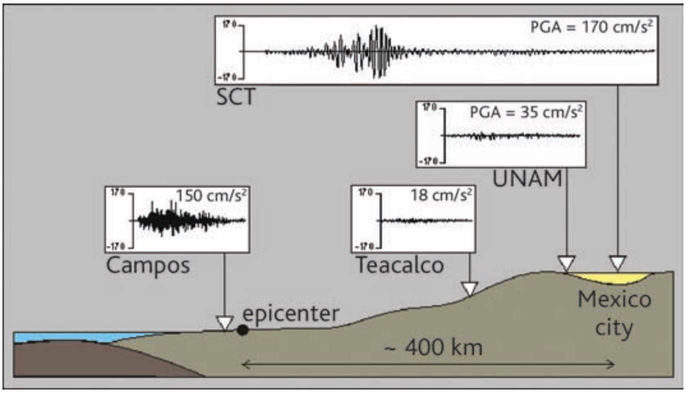

Figure 8.1 Example of site effects in the lakes area of Mexico. © Jean-François Semblat.

non-linear behavior. "Site effects" refers to effects due to the type of ground on which an installation is built (rock, sediment) and effects due to the particular configuration of the site (also known as "site-specific effects", which refer for example to sites in a sedimentary basin set into a rocky environment). The Grenoble Basin has historically been France's test site for observing "site effects" in deep valleys, following the drilling of a deep borehole in 1999 at IPSN's instigation. IRSN has also worked, in collaboration with foreign institutes, in other areas of active seismicity (Gulf of Corinth in Greece, Santiago Basin in Chile). The current sites are in France, in the sedimentary basin of Cadarache, and in Greece, near the town of Argostoli.

IRSN is actively represented within the main bodies involved in seismological research, such as the French Earthquake Engineering Association (AFPS) and the French Seismographic & Geodesic Network (RESIF).

IRSN is also a partner in the SINAPS@[96] project on seismicity and nuclear installations, with CEA (project coordinator), IFSTTAR[97], École Centrale Paris, EDF, ENS[98] Cachan, etc. SINAPS@ is a research project in which seismic risk is assessed as a whole, from the fault to concrete structures and to equipment. It aims to explore the inherent uncertainties in assessing seismic risk and the vulnerability of structures and equipment. The main ultimate objective is to identify, and indeed quantify, the seismic margins resulting from the hypotheses used either when making design choices or when devising the design strategy (conservative hypotheses, choice of materials, etc.). The main themes of the SINAPS@ project to which IRSN is contributing are:

1. Quantification of the uncertainties associated with knowledge of the input data and parameter hierarchy controlling the variability of seismic risk assessments in the deterministic and probabilistic approaches.

96. Earthquakes and Nuclear Facilities, Improving and Protecting Safety.
97. French Institute of Science and Technology for Transport, Development and Networks.
98. French higher education institution.

2. The international benchmark exercise (code and data comparisons) known as PRENOLIN[99]. The PRENOLIN project (2013-2015), reincorporated into SINAPS@ as a result of a request by ANR, aims to produce an approach that takes account of ground non-linearity phenomena in the "site effects" dimension of seismic risk, an approach validated by comparison with field observations. It should eventually be possible to apply this approach to contexts of low and moderate seismicity, which means using simulations, while making use of empirical data (recordings of small earthquakes) for the areas concerned.

3. Numerical simulation of seismic movement close to faults. The aim of this part of the project is to complement the empirical prediction of seismic movement, for which there is a paucity of data close to faults. Numerical simulation is used to explore the physical origins of the variability of seismic movement, which could reduce the uncertainties due to the current lack of knowledge. Quantifying and reducing the uncertainty associated with near-field seismic movement predictions is a major challenge in estimating site-specific seismic risk.

4. Selection of pertinent seismic signals for assessing the earthquake resistance of buildings. This theme lies at the interface between seismology and civil engineering. The multiplicity of selection procedures appropriate to the risk on the one hand, and the evolution of the methods for non-linear modeling of the behavior of buildings on the other, mean that the studies straddle both disciplines. The aim is to provide information for the existing guides to be updated.

There is another research program too: the SIGMA program (Seismic Ground Motion Assessment, 2011–2015) brought together by EDF to meet its own needs and those of other licensees and designers (AREVA, CEA, ENEL[100], etc.). The SIGMA program covers the areas mentioned above: characterization of seismic sources, prediction of ground motion, "site effects" and probabilistic methods. The findings and recommendations for designers and licensees are expected in around 2016–2017.

Lastly, one particular topic of investigation has been chosen for the seismic reviews of France's nuclear power plant reactors: the identification of active faults that could cause ground surface or near-surface movements, commonly known as capable faults[101]. EDF was asked[102] to present a survey of current knowledge concerning the identification of capable faults within a 25 km radius of sites[103] and, where necessary, a methodology for taking account of them when reviewing the seismic risk of sites. The treatment of this

99. Better Prediction of Non-Linear Effects Induced by Strong Seismic Motion.
100. Ente Nazionale per l'Energia Elettrica.
101. On capable faults, see the IAEA guide SSG-1 from 2010.
102. Request made by ASN as part of the safety review associated with the third ten-yearly outage program (VD3) of the 1300 MWe reactors, extended to VD4 900 MWe and VD2 "N4" (letter CODEP-DCN 2014-051797 of December 18, 2014).
103. Investigation radius proposed in IAEA guide SSG-9 (near regional investigations). This guide gives four spatial scales for progressively more detailed investigations: regional, near regional, site vicinity and site area. The guide recommends finding evidence of capable faults going back to very early periods, at least 10,000 years ago.

subject prompted IRSN to launch some new research and development work within international groups (IAEA and INQUA[104]).

8.2. External flooding

As far as other natural events are concerned, the most work has been done on external flooding, with studies of the suitability of statistical methods for explaining outlier events, extreme rainfall, the handling of heterogeneity in the statistical treatment of data (particularly for river floods), the historical analysis of exceptional events (tsunamis on the Atlantic coast, etc.) and assessing the risk of percolation through dikes. The knowledge acquired recently *via* research in this area is set out in the work in reference [2], the outcome of research by a group run by IRSN which brought together representatives from many organizations (ANDRA[105], AREVA, CEA, EDF, the French Technical Maritime and River Study Center (CETMEF), the French National Company of the Rhone river (CNR), the French Naval Hydrographic and Oceanographic Service (SHOM), France's national meteorological service (Météo-France), and the French Technical and Inspection Office for Large Dams (BETCGB). Specialists from the French Geological Research Mining Bureau (BRGM) and the French National Center for Agricultural Machinery, Rural Engineering, Water and Forestry (CEMAGREF) also participated in the group's activities, along with academics.

For each phenomenon (high sea levels including due to tsunamis, river floods, rainwater and runoff, rising groundwater, dam breaches), the following were examined:

- the basic data,
- the physical parameters to be quantified (intensity, volume, water level, etc.),
- the existing characterization methods (deterministic or statistical), identifying the limitations of these methods,
- the identification and incorporation of uncertainties,
- the influence of climate change,
- the dependency between the different phenomena/events.

This report on the state of the art has served notably as the basis for a guide by ASN for nuclear licensees, setting out recommendations for assessing and quantifying external flooding risks, and for defining suitable protection measures[106]. The preparation of this guide reflects the lessons learned from the partial flooding of the Le Blayais nuclear power plant during the December 1999 storm, prompting nuclear licensees to review the safety of installations in view of flood risks under stricter conditions than before and to increase protection for their installations. Aside from these improvements, the report on the state of the art [2] has allowed a detailed reflection process to take place based on

104. International Union for Quaternary Research.
105. French National Radioactive Waste Management Agency.
106. ASN Guide No 13, *Protection of Basic Nuclear Installations against external flooding* - Version of January 8, 2013.

increased knowledge, ensuring external flooding risks are taken into account more comprehensively and more robustly. ASN's guide constitutes a reference text in France not only for new nuclear installation projects but also for the ten-yearly safety reviews for those in operation.

IRSN's ongoing work, especially in the post-Fukushima context, concerns methodological aspects such as:

– statistical methods for determining extreme hydrometeorological events, paleo-sedimentary approaches;
– modeling of flows and runoff phenomena, with application to actual cases.

In addition, concerning the tsunami risk, IRSN is a partner in the TANDEM[107] project (2014–2018) funded by ANR as part of RSNR research, with partners CEA, EDF, BRGM, Ifremer[108], Inria[109] and Pau University. This research project aims to assess the effects of a tsunami on the French coast, focusing particularly on the Atlantic and Channel coasts where there are nuclear installations in operation. This project will deploy new numerical analysis methods, which will be adapted and tested on the databases concerning the March 11, 2011 tsunami off Tohoku in Japan. Once these methods have been validated, they will be applied to the French coast in order to determine the impact of a tsunami on a similar scale. In the long-term, the conclusions should make it possible to produce new guidance for assessing risks to nuclear installations.

References

[1] *Étude du potentiel sismique et de l'enracinement de la faille de la Moyenne Durance*, 2007 scientific and technical report, IRSN, 2007.
[2] *L'aléa inondation – État de l'art préalable à l'élaboration du guide inondation pour les installations nucléaires*, IRSN, Notices and reports/assessment reports/nuclear safety series, 2013.

107. Tsunami in the Atlantic and the English ChaNnel: Definition of the Effects through numerical Modeling.
108. French Research Institute for Exploitation of the Sea.
109. French National Research Institute for the Computational Sciences.

Chapter 9
Research on Core Melt Accidents

When the nuclear power plants with pressurized water reactors (PWRs) and boiling water reactors (BWRs) were built, no specific design provisions were retained as regards core melt accidents. It was not until the 1990s, with reflection and research on "new generation" reactors (Generation III, e.g. EPR, AP1000, etc.), and following the Three Mile Island (TMI) and Chernobyl accidents, that core melt was taken into account in the design of third confinement barrier components[110]. The objective is that the human and environmental consequences of a core melt accident should only lead to counter-measures "limited in space and time" (no permanent relocation of communities, no permanent bans on the consumption of foodstuffs, etc.).

Nevertheless, from the first deployment of PWRs and BWRs, research on reactor core melt and its radiological consequences was begun, particularly in the United States. The aim was to assess the risk to the public from a major accident causing radioactive releases into the environment, at a nuclear power reactor of several hundred MWe, approximately 50 kilometers from a city (*Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants*, WASH-740 in 1957 and WASH-1400 in 1975 [1]). However, it was mainly the accident at the Three Mile Island power plant in 1979, and especially the discovery in 1985 that a large portion of the core had melted, that triggered a huge raft of research and development programs globally in this field.

After more than 30 years' research, vast progress has been made with understanding and modeling the complex phenomena that could occur during a core melt accident. Numerous programs run by the OECD/NEA (CSNI), the Phebus-FP (Fission Product)

110. Or components that could pose a risk to the containment, such as the main reactor coolant system.

program, the international ISTP program, and so on (we will come back to these) made major contributions to this progress. Models using the results of these programs have been integrated into codes for simulating these accidents and their impact on the reactor, e.g. the ASTEC [2] code, developed in France[111] by IRSN, in partnership until 2016 with its German counterpart GRS, and used in its safety assessments and to build the models used in level 2 probabilistic assessments[112]. Since 2004, under the European Commission's impetus, a research excellence network on severe accidents has been set up, known as SARNET [3, 4]. This network, directed by IRSN, brings together the work of around 250 researchers from 44 research organizations and laboratories in 23 countries (EU and non-EU). The network, which is now part of NUGENIA, provides a means of coordinating the researchers' work and pooling their efforts to establish the state of the art of knowledge, determining research priorities, setting up research projects and cooperating on the development and validation of accident scenario computer codes, particularly the ASTEC code (Figure 9.1), which has been made available to the network's members.

Since the early 2010s, as part of the safety reviews for the purposes of extending the service life of power reactors to 60 years and the complementary safety evaluation (ECS) conducted in the wake of the Fukushima Daiichi accident in Japan in March 2011, research in the field of severe accidents has focused more on studying physical measures to protect the containment and limit radioactive releases. These aspects are covered at national level by the ANR RSNR projects launched following the Fukushima Daiichi accident, at European level as part of the research and development framework program (FP7) and Horizon 2020 (H2020) and at international level in the context of a number of OECD projects. In particular, physical measures designed to contain the progression of core melt in the reactor vessel, keep the molten materials in the containment if it fails and reduce radioactive releases through filtration[113] are being examined.

We will not describe here the research and development being conducted to model the often complex thermohydraulic phenomena that lead to core uncovery[114], which were discussed in part in the previous section. We strongly recommend that readers wishing to understand in more detail the phenomenology of core melt accidents and the related research should refer to IRSN's publication *Nuclear power reactor core melt accidents – State of knowledge*[115] published in 2013, or to the reports by the OECD or the U.S.NRC on core melt accidents [5, 6].

111. Other codes have been developed abroad, including MAAP (Modular Accident Analysis Program) and MELCOR.
112. Level 2 PSAs are used to assess the nature and scale of radioactive releases outside the containment as a result of a core melt accident, with the corresponding frequencies, and contribute to assessing the overall safety of the facility. They make it possible to verify that the estimated frequencies of accidents that could lead to large releases to the environment are very low. See the document "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.
113. Where filtered venting of the containment is necessary to prevent its failure due to excessively high internal pressure.
114. The research and development on core melt described in this Chapter relates to an initiating event such as a loss-of-coolant accident, aggravated so that it leads to core melt.
115. Science and Technology Series – IRSN/EDP Sciences – 2013.

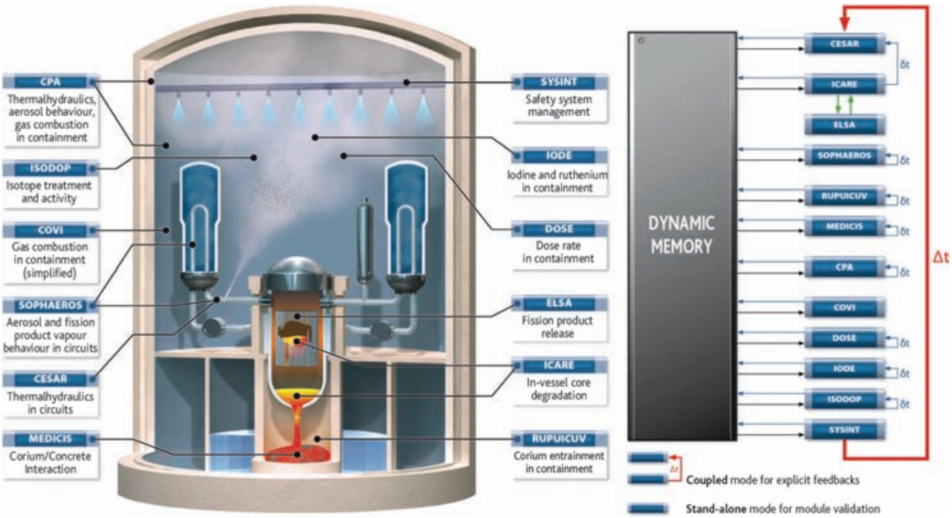

Figure 9.1 ASTEC code modules. © Didier Jacquemain/IRSN.

The different phases following core uncovery, if the accident is not brought under control, are:

- core heating and core melt inside the vessel,
- reactor vessel melt-through and erosion of the concrete basemat of the containment by the molten corium,
- loading of the containment by a sudden increase in internal pressure,
- loading of the containment by a slow increase in internal pressure,
- release of radioactive products outside the reactor (commonly referred to using the term "source term").

In the early 1980s, research focused mainly on studying the first phase.

9.1. Heating of the core and core melt inside the vessel

Many analytical tests were performed in the 1980s by different research laboratories, particularly in the United States and Germany, to study separately the various physical phenomena and find out the laws governing them.

Initially, the oxidation of the zirconium alloy cladding by steam was the subject of laboratory studies (see Section 3.2). This is an exothermic phenomenon that increases once the temperature has passed 1000 °C. The hydrogen produced spreads inside the vessel then in the reactor building if a break occurs in the reactor coolant system. The kinetics of cladding oxidation can be described by a parabolic law, expressing the fact that at a given temperature, the thickening of the oxide layer acts as a barrier to the transfer of oxygen to the remaining metal, slowing down the oxidation. It greatly increases above 1580 °C as a result of a change in the zirconia's metallurgical phase, making it less resistant to oxygen diffusion within the metal itself. Correlations allow these phenomena to be taken into account with a good level of confidence.

The core of a water reactor is made up of many materials: several dozen metric tons of uranium oxide, plus around ten metric tons of zirconium, but also stainless steel, Inconel and materials designed to control the fission reaction. These are a silver, indium and cadmium alloy (AIC) in 900 MWe reactors, boron carbide (B_4C) and AIC in 1300 MWe and 1450 MWe reactors and in the EPR reactor – boron carbide only in BWRs. All these materials can interact with one another and form liquid phases at much lower temperatures than their respective melting points. Uranium oxide, which melts at 2840 °C, can be dissolved at much lower temperatures than its own melting point in alloys where it is combined with metals with lower melting points (such as the zirconium of fuel claddings).

Experiments on small samples of materials, particularly in Germany, the United States and Japan, have enabled these interactions to be observed. Phase and correlation diagrams have been drawn up as a result, describing the reaction kinetics. Between 1200 °C and 1400 °C, the materials absorbing the neutrons (AIC, B_4C) react chemically with their stainless steel cladding, forming a liquid mixture that in turn dissolves the zirconium. Above 1760 °C, the zirconium metal melts and dissolves the zirconia and the uranium oxide. Finally, the zirconia melts at 2715°C and also dissolves the uranium oxide. The mixtures obtained can also react with steam if they are not completely oxidized, producing hydrogen.

These phenomena of oxidation and chemical interaction between all the materials in a reactor core have been modeled, usually empirically, and the models have been integrated into simulation codes. In brief, the reactor core is split into meshes – representing at fuel rod scale the average behavior of a set of fuel assemblies over a given height – and for each mesh the code calculates the velocity and composition of the fluid flowing in the core, the temperature reached, the oxidized cladding thickness and the fraction of molten material and its composition. Above a certain fraction of molten material, trapped between the fuel pellets and cladding in the case of fuel rods or accumulated in a control rod, or if the remaining envelope of solid cladding is estimated not to be thick enough to contain the molten material, the liquid materials flow by gravity and solidify in the lower, cooler regions of the core. They then form metal plugs (mainly of AIC or a mixture of boron and steel) topped by ceramic plugs (Zr, U, O mixture).

Given the complexity of the physical and physico-chemical phenomena studied, the diversity of the materials present and the geometry of a reactor core, it was quickly realized that it was necessary to perform integral tests on a larger scale with components representative of a reactor core. The aim was to check that the interactions between the different phenomena would not produce effects that had not been predicted by the models stemming from the analytical tests.

Most of these tests were performed in experimental facilities, using a nuclear reactor to heat the fuel being studied. This experimental method, which is fairly difficult and very expensive to carry out, has the advantage of keeping the fuel (and only the fuel) heated when it flows to the bottom of the experimental equipment once melted, as in a real accident.

In France, IPSN, and then IRSN, conducted two large-scale research programs in the PHEBUS experimental reactor described in Section 2.1: the Phebus-CSD[116] programs (on severely degraded fuel) from 1986 to 1989 and Phebus-FP (on fission products) from 1988 to 2012 (the last test was performed in 2004).

The **Phebus-CSD program** was used to study phenomena linked to zirconium oxidation by steam, the associated hydrogen production, physico-chemical interactions between the uranium oxide and zirconium, in a solid or molten state, and the effect on core degradation of AIC control rods. The program consisted of six tests: B9, B9R, B9+, C3, C3+ and AIC. The experimental devices each consisted of an assembly of 21 fuel rods that had not been irradiated, of the same type as those used in PWRs but shorter (a fissile length of 80 cm compared with 3.66 m in a real fuel rod), distributed in a square-pitch arrangement and held in place by two Inconel grids. The assembly was contained inside a steel tube, which in turn was inserted into the cell placed at the center of the reactor. A thick thermal screen made from porous zirconium was used to isolate the assembly from the steel tube, cooled by the circulation of pressurized cold water.

A mixture of steam, hydrogen and helium, the composition of which varied from one test to another depending on the test objectives, was injected at the foot of the assembly. The assembly was heated by gradually increasing the power of the PHEBUS reactor. A borated steel neutron screen placed between the PHEBUS reactor core and the test device was used to obtain an axial distribution of uniform power as in a power reactor core.

Each device was fitted with a very large amount of instrumentation for measuring all the physical parameters essential for interpreting the tests, particularly high temperature thermocouples (tungsten-rhenium, W/Re), which functioned at up to 2100 °C, and a mass spectrometer for measuring the amount of hydrogen being produced at every moment. An array of post-test examinations, both non-destructive (X-rays, gamma scanning[117]) and destructive (photo macrography, analyses with a scanning electron microscope), complemented the measurements taken during the experiments, so that the damage to the fuel rods could be quantified and the composition of the resolidified mixtures could be determined.

The six tests carried out produced many experimental results, which were used for developing and validating the core melt models integrated into the ASTEC simulation code. In particular, they made it possible to observe how the oxidation of the zirconium alloy cladding spreads axially with the temperature runaway above 1550 °C. The progression of the oxidation front was characterized by the occurrence of steam

116. Severely Degraded Fuel.

117. Device for measuring the intensity of the gamma radiation emitted by a source depending on its energy levels. Following calibration, analysis of the spectrum obtained can identify the radionuclide(s) responsible for the radiation and their masses.

starvation, indicating that all the steam available to the cladding had been absorbed by the cladding to oxidize the metal (the gas flowing beyond the front did not contain any more steam, only hydrogen). Very significant results were obtained for interactions between the core materials and the processes causing core melt (liquefaction of the zirconium on contact with the Inconel of the grids, dissolution of the outer layer of zirconia and the fuel inside the cladding by the molten zirconium, liquefaction of the control rods at around 1400 °C and damage to the adjacent fuel rods).

The **Phebus-FP program** [7] helped to reduce the uncertainty concerning the estimation of radioactive releases of a core melt accident at a light water reactor, and to increase IRSN's assessment and emergency response expertise in this field. The program consisted of five tests during which the main physical phenomena governing core melt, the transfer of fission products from the fuel to the reactor building, and their behavior inside it were studied. It was conducted in collaboration with a number of French and foreign partners (EDF, the European Commission and its Member States, the United States, Canada, Japan, South Korea and Switzerland). The definition of the tests, engineering and experiments, and the analysis and interpretation of the experimental results involved around 80 people over a period of approximately 15 years.

The program required major modifications to the PHEBUS facility. The reactor building was strengthened against earthquakes and enlarged to accommodate a sealed steel box containing a model of the reactor coolant system and containment of a PWR and all the necessary equipment and instrumentation (Figure 9.2). The experimental circuit reproduced, on a 1/5000 scale, the three vital components for this type of study of a 900 MWe PWR: the core, the reactor coolant system and the containment.

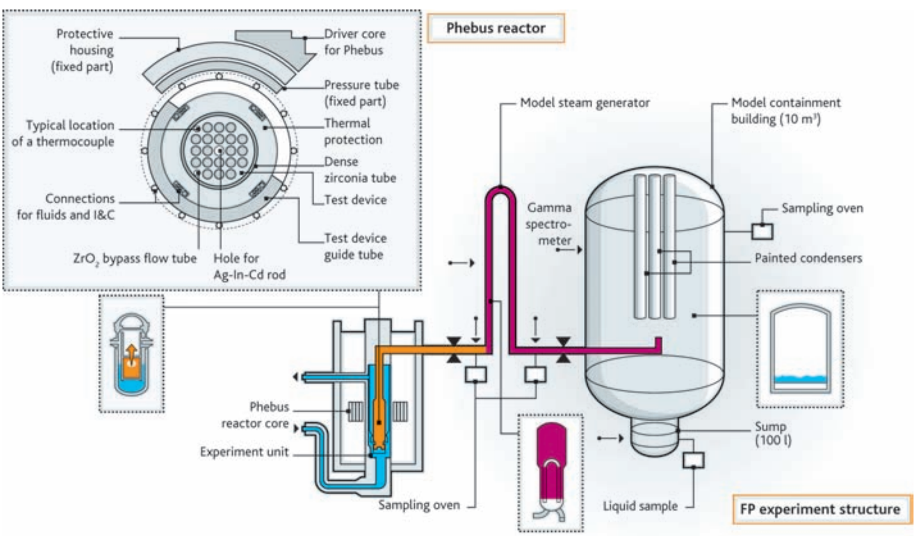

Figure 9.2 Schematic view of the experimental circuit used for the Phebus-FP program. © IRSN.

To represent the core, a similar device to the one designed for the Phebus-CSD program was used. This was an assembly consisting of 20 fuel rods with a fissile height of 1 m, and a rod in the center representative of the control rods used in power reactors, all arranged in the cell at the center of the PHEBUS reactor. The fuel used in the first test had not been irradiated before being used in the PHEBUS reactor, whereas the fuel used in subsequent tests came from the Belgian BR3[118] reactor and an EDF reactor; their burnup rates were between 23 and 38 GWd/tU. The instrumentation had been significantly improved, in particular with the use of ultrasonic thermometers that could measure temperatures of up to 2900 °C.

The reactor coolant system model consisted of a section cooled to 150 °C simulating a steam generator tube with a break at the outlet from this tube. The fluid leaking from the break was collected in a 10 m^3 tank fitted with a sump at its base, simulating a reactor containment. The external surface of the tank was thermally isolated and there were cooled cylinders at its center as a scale representation of the containment wall of a PWR on which the steam could condense. Some of the cylinders and a plate immersed in the sump were coated with identical paint to the paint used in the PWRs in France's nuclear power plant fleet.

The facility's strength was its instrumentation. As well as measuring flow rate, pressure and temperature at several points in the circuit, it could also measure hydrogen concentrations in the coolant system and the containment. It also included gamma-ray spectrometers targeted at the tube intake, several points along the wall and the outlet from the tube simulating the steam generator, the atmosphere and the walls of the tank simulating the containment, as well as the sump, to continuously identify and measure the different γ-emitting fission products in suspension in these areas or deposited on the walls. A Maypack composed of selective filters, monitored continuously during the tests by a gamma-ray spectrometer, was used to measure the concentrations of the different forms of iodine in the containment atmosphere (aerosols or gases, and among the gaseous species, molecular iodine or organic iodine). There was also an optical device for continuously measuring the aerosol concentrations in suspension in the containment.

Branch pipes at the intake and outlet of the tube simulating the steam generator, in the sump and on the containment wall, meant that samples could be taken at various points during the tests through filters, impactors (instruments for measuring the size and concentration of the aerosols), thermal-gradient tubes (instruments used for working out the chemical forms of different species in vapor form by recording the temperatures at which they condense on the walls) and in capsules. In the tanks simulating the containment, coupons collecting the aerosols deposited by settling were also exposed to the containment atmosphere for predetermined time intervals. After the tests, radiochemical analyses were performed on all the samples taken, allowing identification of the fission products and the different actinides carried or deposited, and enabling their masses to be calculated precisely.

118. Belgian Reactor 3.

Specific non-destructive examination techniques using transmission and emission tomography[119] were also developed for the program and provided valuable information about the state of degradation of the fuel rods after the tests, and about the quantities of some radioactive products emitted by those rods.

The international Phebus-FP program, which ran from 1988 to 2012, consisted of five tests. The main parameters studied in these tests are summarized in the table below.

	Fuel type	Material simulating the control rods	Steam flow rate (g/s)	pH of the sump water	Sump/atmosphere temperature difference[120] (°C)
FPT[121]0	Very lightly irradiated	AIC	0.5–3	5	–18/–37
FPT1	~20 GWd/tU	AIC	0.5–2.2	5	–18/–38
FPT2	Ditto	AIC	0.5 (+ boric acid)	9	–18/+11
FPT3	Ditto	B_4C	0.5	5	–18/–3
FPT4	Debris bed of UO_2 at 38 GWd/tU and of ZrO_2	None	0.2–0.5 H_2O 0–0.25 H_2	N/A	N/A

Apart from test FPT4, each test consisted of three successive phases:

– the first phase of re-irradiation of the fuel, during which the PHEBUS reactor was operating at full power for approximately a week. The fuel being studied was cooled by pressurized water circulation. The aim was to recreate in the fuel the short-lived fission products that had disappeared since the fuel was unloaded from its original reactor, such as iodine-131 (half-life of 8 days[122]), which was essential for studying the releases;

– a "degradation" phase lasting a few hours, during which, by gradually increasing the PHEBUS reactor power, the temperature of the uncooled test fuel increased until the materials liquefied (at between 2300 °C and 2500 °C), causing the fission products to be released and transported into the circuit and then into the tanks simulating the containment. At the end of this phase, the PHEBUS reactor was shut down;

119. Technique for reconstructing the volume of an object from a series of radiographic measurements (transmission) or gamma-ray spectrometry measurements (emission) performed on each unit from the outside.

120. On the left of "/", temperature difference during the degradation phase; on the right of "/", temperature difference during the long-term study of iodine behavior. A positive difference means that water is evaporating from the sump surface, which is likely to encourage the transfer of the volatile iodine produced in the sump water to the containment atmosphere.

121. Fission Product Test.

122. The concentration decreases by half during the half-life.

- a “containment” phase, lasting for several days during which the variables of interest for understanding the deposition of radioactive products and the iodine chemistry in the tank simulating the containment were measured.

The first group of three tests (FPT0, FPT1 and FPT2) was used to study the effect of the burnup rate and the oxygen potential (atmosphere rich in steam or conversely in hydrogen, surrounding the fuel rods) on oxidation and damage to the fuel rods, and on the release and transporting of fission products in the circuit. These three tests were performed with an AIC fuel rod. Boric acid was also injected with the steam during test FPT2 (during normal reactor operation the boric acid serves to control its reactivity; in an accident situation, the boric acid can react with some fission products such as cesium, and significantly alter the chemical balance with the different iodine species, encouraging the formation of the most volatile).

Test FPT3 was used to study the influence that the presence of boron carbide control rods has on core degradation and fission product behavior. This type of material is used in EDF's most recent PWRs (1300 MWe, 1450 MWe and EPR) in addition to AIC, as well as in BWRs.

Test FPT4 focused on the advanced phase of an accident, studying the release of low-volatile fission products and transuranium elements from a debris bed of fuel and oxidized cladding brought to melting point. It consisted of a single temperature-increase phase until a molten material pool was obtained. The fission products and actinides emitted were sampled through a bank of filters situated above the debris bed and activated in succession.

Tests FPT0, FPT1, FPT2 and FPT3 were used to reproduce fuel rod damage states that had never yet been achieved experimentally, with significant melting of the fuel (by up to 50% of its mass) and the formation of corium pools, as illustrated in Figure 9.3. The numerous results obtained contributed significantly to improving knowledge of core damage mechanisms and substantially extended the scope of simulation code validation. For example, the cladding dislocation criterion was revised to take account empirically of the mechanisms causing the zirconia outer layer of the cladding to fail, releasing the molten zirconium inside and causing oxidation and hydrogen production to stop locally. The tests also significantly improved knowledge of the effects of physico-chemical interaction between the fuel, the zirconium and the materials forming constituting the control rods, leading to liquefaction of the fuel at around 2200 °C, a much lower temperature than predicted by models. The core degradation models in simulation codes such as ASTEC have benefited extensively from these new results and their prediction capabilities have been greatly improved by them.

The originality and quality of the tests performed at the PHEBUS facility was such that the Phebus-CSD B9+ and Phebus-FP FPT1 tests were selected by the OECD/NEA for the performance of international exercises to compare core melt codes (ISP 28 and ISP 46).

Abroad, large-scale programs have also been run, in chronological order:

- the SFD (Severe Fuel Damage) program, consisting of four tests conducted between 1982 and 1985 in the PBF (Power Burst Facility) reactor of the Idaho National Laboratory in the United States;

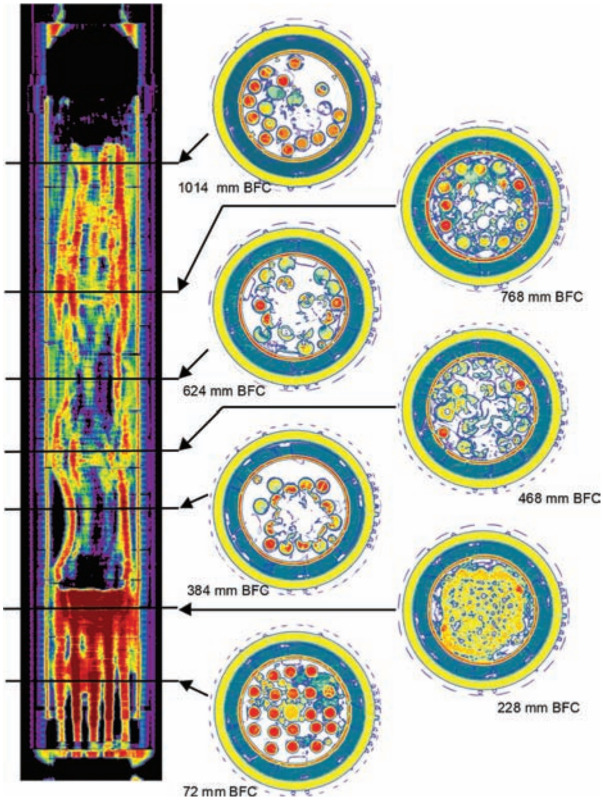

Figure 9.3 State of degradation of a fuel assembly after test FPT1; post-test non-destructive examinations of the tested fuel cluster: X-ray (left) and X-ray transmission tomography (right). (BFC: Bottom of Fissile Column). © IRSN.

– the DF (Damaged Fuel) program, consisting of four tests conducted in 1984 in the ACRR (Annular Core Research Reactor) reactor of the Sandia National Laboratory in the United States;

– the OECD/NEA LOFT-FP (Loss of Fluid Tests–Fission Product) project, consisting of two tests conducted from 1984 to 1985 in the LOFT reactor of the Idaho National Laboratory in the United States;

– the FLHT (Full Length High Temperature) program, consisting of four tests conducted between 1985 and 1987 in the NRU (National Research Universal) reactor of the Chalk River Laboratories in Canada;

– the CORA program, consisting of around 20 tests conducted between 1987 and 1997 at a non-nuclear facility of the German research center in Karlsruhe.

For experiments performed in nuclear reactors, the experimental devices consisted of assemblies of between ten (DF) and several tens (PBF) of fuel rods (or indeed 100 in the case of the LOFT-FP project) of similar manufacture to the rods used in power reactors. They were generally shorter, like those used in the Phebus-CSD and FP programs, because the cores of the research reactors hosting the experimental devices were not as high, with the exception of the FLHT program which, as the name suggests, used full length rods with a fissile height of 3.66 m in the large NRU reactor. In some tests, rods containing AIC or boron carbide were included in the assembly, as well as fuel rods that had previously been irradiated for a significant period to obtain representative burnup rates.

In the CORA program tests, the fuel rods were simulated by zirconium alloy cladding filled with alumina and heated internally using a tungsten core with an electrical current running through it. The lower cost of these tests meant that many parameters could be studied: temperature increase rate, flow rate of the injected steam, ambient pressure, cladding materials, effect of the presence of AIC or boron carbide, geometry of the different reactors (PWR, BWR, VVER), effect of quenching at the end of the test, etc. However, this heating method meant that it was not possible to study the corium pool formation phase.

All the tests contributed to a better knowledge of cladding oxidation, hydrogen production and core melt phenomena. They delivered a wealth of experimental results covering a broad spectrum of situations, which are extremely useful for developing models and validating simulation codes such as the integrated ASTEC, MAAP and MELCOR codes.

In 2016, as shown by a benchmark exercise run by the OECD/NEA for an accident scenario similar to TMI-2 [8], the integrated codes for core melt accidents are able to simulate with a good degree of precision the progression of PWR core degradation up to the reflooding phase. On the other hand, a benchmark exercise run by the OECD/NEA for the damaged BWR reactors at the Fukushima Daiichi plant [9] identified major discrepancies in the calculation of the progression of core degradation in the three damaged reactors and of hydrogen production, revealing current uncertainties as regards representing degraded configurations for BWRs. Partly these difficulties are due to the fact that core melt accident experiments have focused mainly on PWR fuel assembly configurations. There has been very little study of BWR fuel assembly configurations; major uncertainties exist with regard to molten mixture relocation phenomena in the presence of Zircaloy channels for water in the assemblies and the cruciform B_4C-steel control rod plates specific to BWRs. More detailed analyses are under way in order to understand the differences between these very different configurations.

Interestingly, none of the tests conducted abroad reached the temperature or fuel degradation levels observed in the Phebus-FP tests, making them particularly valuable. On the other hand, some of the tests performed abroad (LOFT, CORA and QUENCH programs – QUENCH succeeded CORA at the same facility from 1997) have been able to study the effects on cores molten to varying degrees, of reflooding with large flows of water or steam.

The most important finding from these tests is that reflooding a core that has already been extensively damaged may, at least initially, considerably accelerate cladding oxidation, hydrogen production and core melt. When the accident in Unit 2 at the

Three Mile Island power plant (TMI-2) happened, there is no doubt that water entering the molten core caused any cladding that had not yet melted to rupture and the debris bed observed in the upper part of the core to form. Experimental results tend to show that core reflooding can halt the progression of an accident only if it happens very early on, i.e. when the maximum temperature in the core has not yet exceeded 1800 °C, and if the flow of water is sufficient (more than 2 g/s per rod). Despite all the experimental results obtained, models are unable at present to reliably predict whether or not reflooding a damaged core can cause runaway oxidation of the cladding and aggravate the situation.

To find out more about this phenomenon, since 2010 IRSN has been conducting test programs with the PRELUDE[123] and PEARL facilities at its laboratories in Cadarache, involving reflooding experiments on debris beds consisting of steel balls heated by induction inside a cylindrical cavity (diameter 11 to 29 cm in the case of PRELUDE and diameter 50 cm in the case of PEARL). The PEARL tests performed in a large-diameter section were used to observe the two-dimensional flows. An experimental and theoretical research program on the reflooding of solid debris beds in more complex configurations, in particular with the observation of oxidation phenomena, is planned for 2018.

If core reflooding is ineffective, the accident continues and the molten materials flow to the lower head of the reactor vessel, where they accumulate, as seen in the case of the TMI-2 accident. If these materials come into contact with water at the lower head, vast quantities of steam and hydrogen are produced. Despite the many examinations carried out after the TMI-2 reactor accident, particularly during the OECD/NEA TMI-VIP program, and the tests of dropping corium into water performed on a large scale in the FARO facility at the European Ispra Joint Research Centre in the 1990s, the flow mechanisms of the corium at the lower head and its mixture with the water present are still not clearly understood. This difficulty is overcome by using pessimistic hypotheses for the quantity of materials transferred, their fragmentation on contact with water, the oxidation reactions, etc. when evaluating the progression of a core melt accident. Nevertheless, all the experts agree that a steam explosion violent enough to destroy the reactor vessel could not occur under these conditions (OECD/NEA project entitled SERENA[124], described further on).

If the corium debris at the lower head is not cooled effectively, it will melt again, forming a pool in contact with the vessel walls, threatening the vessel's integrity. Various test programs have been undertaken to study this configuration. These test programs either use simulator materials, like the BALI tests conducted by CEA in Grenoble, at full scale in 2D geometry with water, to study natural convection in the pool, or they use representative materials (UO_2, zirconium alloy, steel), like the OECD/NEA test programs RASPLAV (1994–2000), MASCA (2000–2003) and MASCA 2 (2003–2006) conducted in Russia. During the latter test programs, corium (a mixture of UO_2-ZrO_2-Zr and steel) of various compositions and in different atmospheres (inert or oxidizing) was brought to

123. Preliminary Study on the Experimental Reflooding of a Debris Bed.
124. Steam Explosion REsolution for Nuclear Applications.

melting point and kept between 2200 °C and 2650 °C. In RASPLAV, a 200 kg mass of corium, with mass composition of 76.6% UO_2, 14.3% Zr and 9.1% ZrO_2, heated by tungsten walls forming a thin semi-circular section of a reactor vessel, was used. In the smaller scale tests using corium in crucibles cooled on the outside, heating was achieved by electrical induction and the more varied corium compositions studied meant that coriums with a high metal content could be used and in some cases simulators of FPs (a stable species of the radioelements of interest in order to study their distribution between the various metallurgical phases of the corium). In parallel, smaller scale tests were conducted with coriums of identical composition, allowing the physical properties of the material mixtures to be determined. Tests using pools of molten salt, eutectic or binary mixtures, completed the studies of the effect that the material properties have on the interface temperatures and their impact in correlations for predicting the distribution of flows of heat extracted from a pool to the walls of the vessel containing it. The RASPLAV tests confirmed that using traditional correlations for the distribution of heat from a pool at its edges (established using various simulator liquids that are easier to manage) with the physical properties of the corium was entirely justified.

The experiments performed in MASCA and MASCA 2 reveled complex phenomena of interaction between materials and of stratification between the metal and oxide phases in the presence of steel. Under certain conditions (particularly where the pools initially contain a significant amount – around 50% – of unoxidized zirconium), a three-layer configuration was observed with a high density metal layer containing uranium, topped by a layer of refractory oxides (mainly uranium and zirconium oxides), itself topped by a low density metal layer (iron, chromium, nickel, zirconium, etc.). In the stratified pool configurations where there is no reflooding, a very intense flow of heat can be transmitted to the walls around the upper metal pool (heat flow concentration or focusing effect), if its thickness is below a threshold of around 50 cm. The tests also revealed the transient nature of the stratification phenomena where steam is present, since the metal layers oxidize and gradually mix with the oxide pool.

These phenomena are modeled in the ASTEC simulation code but many uncertainties remain concerning the transient phenomena, particularly oxidation of the uppermost layer of metal and the transfer of oxygen between the layers when steam is present. To reduce these uncertainties, an experimental program known as CORDEB (Corium-Debris) was launched in 2012 by the Aleksandrov Scientific Research Technological Institute (NITI) in St Petersburg. It should run for at least seven years in the MASCA facility, in partnership with EDF, AREVA, CEA, IRSN. Examining the corium pools produced during the Fukushima Daiichi accident could shed further light, of great relevance, on the subject.

9.2. Reactor vessel melt-through and basemat erosion by the molten corium

The OECD/NEA TMI-VP project to examine in detail the damaged TMI-2 reactor vessel and its contents, mentioned above, established that the internal face of the lower head, lined with stainless steel, had locally reached a temperature of 1100 °C for

30 minutes, while the pressure inside the vessel was 10 MPa. The cooling of the corium once water circulation had been restored in the reactor coolant system prevented reactor vessel melt-through, though in 2016 the reasons for this are still not clear. Once of the hypotheses advanced to explain why the reactor vessel remained intact was that direct contact between the corium and the lower head was limited by the presence of water at the lower head when the corium flowed in.

The OECD/NEA experimental project known as OLHF[125], conducted by the Sandia National Laboratories in the United States with the involvement of CEA and IRSN, obtained experimental results on a 1/5 scale model for the creep-induced failure mechanisms of a PWR lower head at a representative range of temperature/pressure parameters. However, there are uncertainties involved in applying the results to a full-scale reactor, since a number of welded tubes pass through the lower head. A test conducted with these tubes showed a significant reduction in the rupture elongation of the vessel.

Following vessel melt-through, the molten corium runs into the reactor pit and, if there is no proper cooling, causes the chemical decomposition due to heat, and therefore erosion, of the concrete reactor pit walls and basemat. This could ultimately put the integrity of the containment at risk and lead to massive radioactive gaseous and liquid releases into the environment. The interaction between the corium and the concrete also generates large amounts of gas (steam, which is reduced to hydrogen and carbon monoxide and dioxide due to oxidation of the rebars or the metal phases present in the corium), which can increase the pressure inside the containment and also threaten its integrity.

During the erosion of the concrete by the corium, which can last for several days, the composition of the pool changes. It contains some dense oxides (mainly U-Zr-Fe-O_x), low-density oxides (SiO_2, CaO) and non-oxidized metals (Fe, Cr, Ni, Zr). These are gradually oxidized by the steam or carbon dioxide whereas they can only reduce much more marginally other less stable oxides. The pool solidifies on contact with the cooler concrete, temporarily forming crusts that are mechanically unstable since the underlying concrete wall is breaking down. Stratification between the metal and oxide phases can also occur, at least temporarily, changing the distribution of the heat flows. Whether this phenomenon occurs depends on the configuration of the pool within the vessel when the vessel fails (if it is stratified or evenly mixed) and the position of the break (e.g. if it is level with a metal layer). In the longer term, it depends on how the composition of the metal phase changes, the density ratio between metal and oxide phases, the flow rate of the gases generated by decomposition of the concrete, and the residual quantity of unoxidized metal; all these parameters changing during the interaction. Incorporation of the metals from molten rebars in the concrete can contribute to the formation of a metal layer.

125. OECD Lower Head Failure.

In France, these phenomena were first studied experimentally in small-scale analytical tests using simulator materials with low melting points. The tests were conducted by CEA on behalf of IPSN (then IRSN) and EDF:

- the BALISE program studied liquid entrainment at the interface between two non-miscible liquids with different densities and viscosities (water and liquid paraffin, silicon oils, etc.), percolated through a nitrogen flow (1999–2000);
- the ABI program measured the heat transfer coefficients at the interface between two non-miscible liquids with very different densities (Wood metal and gallium to simulate the corium, water and oils of different viscosities to simulate the concrete), percolated through a nitrogen and argon flow (2007–2008);
- the ARTEMIS program simulated the erosion phenomena in one-dimensional then two-dimensional geometry using a salt ($BaCl_2$) to simulate the corium and a binary eutectic mixture ($LiCl$-$BaCl_2$) to simulate the concrete (2003–2008);
- the CLARA program measured heat transfers in a pool of water in the presence of a flow of gas injected through the walls using water and additives to vary the viscosity (2008–2012).

Given the importance of the combination of convection and thermal and physico-chemical phenomena, tests with real materials were necessary. At Cadarache CEA built the VULCANO[126] facility, commissioned in 1997[127], which uses a rotary plasma arc furnace to melt prototypic corium mixtures (a UO_2 and ZrO_2 mixture, also incorporating oxides from the decomposition of the concrete to study the later phases of the interaction) with a mass of around 40 kg, which are then poured into an experimental section. For the programs conducted from 2003 with the participation of EDF and IRSN, to study interaction between the corium and concrete (CCI[128]), corium pools were poured into cylindrical concrete crucibles of varying compositions. High frequency induction heating was used to reproduce the residual heat released in the fuel throughout the interaction. The instrumentation in place was used to determine the axial and lateral progression of the concrete ablation front.

One specific characteristic of the facility is that it can also be used to study the erosion caused by metal/oxide pools and the impact of stratification. A mass of steel of approximately 25 kg representing the molten structural materials was melted in three separate induction furnaces then poured into the experimental section containing the molten corium. In the test section, a system of magnetic screens meant that it was possible to heat only at the level of the oxide phase of a stratified mixture.

126. Versatile UO_2 Laboratory for COrium ANalysis and Observation.
127. The facility was initially used to study corium spreading in the core catcher installed under the EPR reactor vessel. The tests performed there were also used to validate the CROCO corium spreading simulation code developed by IRSN.
128. Corium-Concrete Interaction.

The following tests were devoted to studying the interaction between corium and concrete:

- nine tests with all-oxide corium (VB-U series),
- four tests with corium and steel (VBS-U series),
- four tests that were more analytical (VBES-U series).

The results revealed the importance of the type of concrete (siliceous or limestone) to the concrete's ablation profile, confirming the results of tests carried out by the Americans (see further on). They also contributed to the development of models for evaluating a conservative approach taking account of the uncertainties that still exist concerning the time to penetration of the concrete containment basemats of French reactors if a corium/concrete interaction were to occur. It has therefore been possible to demonstrate that basemat penetration should not occur within 24 hours of the start of an accident[129], allowing time for measures to be taken to protect the public.

In Germany, tests were performed in the 1990s at the BETA facility in the Karlsruhe research center. The corium was simulated by a molten mixture of 400 kg of alumina and iron obtained through an exothermic oxidation/reduction reaction, to which several dozen kilograms of zirconium were added. This mixture was poured into a concrete crucible.

In the United States, three research programs were conducted at facilities of the Argonne National Laboratory: the MACE (Melt Attack and Coolability Experiments, 1989–2000) programs on behalf of EPRI (Electric Power Research Institute) and the OECD/NEA MCCI[130] programs (2006–2010) in which France (IRSN, EDF and CEA) was a partner. The objective of these two programs was to conduct large-scale and smaller-scale integral tests using representative materials, to study:

- concrete erosion phenomena in a two-dimensional geometry in the absence of any cooling;
- the various cooling mechanisms identified during the MACE experiments when water was poured on the surface of the corium;
- new technological concepts for cooling the corium and halting the concrete erosion.

Two experimental devices were used, SSWICS (Small Scale Water Ingression and Crust Strength) and CCI (Corium Concrete Interaction). In the first, a mass of around 75 kg of UO_2 and ZrO_2, mixed with siliceous or limestone concrete, brought to a temperature of more than 2000 °C by an exothermic oxidation/reduction reaction, was melted in a cylinder of diameter 30 cm with inert walls made from magnesium oxide. Water was poured on the corium to study the cooling mechanisms and their effectiveness, in particular the mechanism of water penetration into a top crust that has cracked

129. In the case of Fessenheim, after basemat thickening.
130. Molten Core-Concrete Interaction.

due to thermomechanical stresses. Gas could also be injected through the bottom to simulate the gases released during the corium-concrete interaction.

In the second device, a mixture of 400 to 590 kg of UO_2 and ZrO_2 was melted using the same method in a larger cavity with a square cross-section of 50 cm by 50 cm (one test was performed in a 50 cm by 70 cm cavity with a single ablatable side wall to study any effects of scale). Tungsten electrodes were arranged on two opposite sides of the cavity so that an electrical current could be applied to the corium pool to generate a release of heat representative of the residual heat. The other two sides and the bottom were made from concrete; the type of concrete (siliceous or limestone) was one of the parameters of the study. On this test device, the "Joule effect" heating technique made it impossible to study stratified pools with two separate phases, oxide and metal, because the stratification of the metal would cause an electrical short circuit between the electrodes. Following a dry corium-concrete interaction phase to study the concrete ablation mechanisms, water was added at the top to study the cooling mechanisms.

In all, 21 tests were carried out during the two OECD/NEA MCCI programs. They provided important results that significantly improved understanding and modeling of concrete ablation and corium pool cooling mechanisms in a corium-concrete interaction. As far as concrete erosion kinetics in the absence of cooling are concerned, it was observed that the ratio between the final erosion thicknesses in the axial and lateral directions depends on the type of concrete. Whereas the erosion is almost uniform in the case of limestone concrete, it can be three times greater laterally in the case of siliceous concrete. As regards cooling mechanisms, it was shown that cracking of the upper crust and water penetration is much more effective when the cooling above the pool starts early, i.e. while the concrete content inside the pool is still low and the crust is more fragile. Eruption and fragmentation of the corium ejected into the water through the crust covering the pool were also observed. They help to cool the pool by turning a compact mass into a bed of debris several millimeters in diameter. This is even more effective when the concrete is a limestone type (more gas is released ejecting more corium through the cracks in the crust). At the end of phase 2 of the MCCI program, a test with siliceous concrete but with cooling at the start of the corium-concrete interaction meant that it was possible to observe ejections with this type of concrete for the first time. This encouraging result prompted EDF to continue funding tests (CCI), as part of its work program on extending the operating life of France's nuclear power plants, to study cooling mechanisms where water is injected early on in the corium-concrete interaction at the surface of the pool, with support from IRSN, the U.S.NRC, NRA and CEA.

Experiments have also been conducted in Germany to test a different corium cooling concept (COMET) by injecting water through a special device consisting of several porous concrete layers. This device could prove more effective than pouring water on the corium because then the liquid and steam are not circulating against the flow. These experiments were conducted at the Karlsruhe center, using up to 1300 kg of alumina and iron melted using an exothermic oxidation-reduction reaction and kept at temperature by induction heating. Following the erosion of a sacrificial layer of concrete, the pool is cooled by water injection *via* a layer of highly porous concrete through porous concrete blocks, the section of which was optimized to maximize cooling efficiency while avoiding a too fast production of steam, which would increase pressure inside the containment. The layer of highly porous concrete through which the water was injected on to the blocks was linked to a tank on a higher level, so water was flowing by gravity only. During the OECD/NEA MCCI-2 test program and specifically a VULCANO VWU[131]-1 test, cooling with a device following the same principle was also studied using prototypic materials. The results led to the conclusion that this type of cooling system was effective, meaning that it could be implemented in new reactor concepts.

Knowledge of concrete erosion and corium cooling phenomena has advanced significantly thanks to all the experimental results produced in the last 20 years. More robust models based on a large database of experimental data have been integrated into simulation codes, particularly the ASTEC code. However, it is still not possible to model in any detail the phenomena occurring at the interfaces between the corium and concrete and therefore to understand from a mechanistic point of view the differences between erosion kinetics due to the concrete type. IRSN, in partnership with universities and the CNRS, has tried to use direct simulation techniques to advance this area of knowledge, e.g. by studying the material flows at the corium-concrete interface or the consequences of bubbles passing through an interface between two non-miscible liquids, in terms of the mass of liquid entrained and the effects on heat transfer through convection. Without any local information for validation purposes, these efforts have not delivered any very significant advances for the erosion models and so the current models are still very empirical.

Moreover, without sufficient experimental data, uncertainties remain about the possible effects of a stratified pool. Two mechanisms could limit the life of such a pool: oxidation of the metal phase and mixture of the materials due to the gases released by decomposition of the concrete. Further tests are therefore necessary to study these phenomena and reduce the uncertainty. The impact of the rebars on the erosion mechanisms is also unresolved because results from the MOCKA tests conducted at Karlsruhe in the last few years, with simulator materials at high temperatures, are fairly contradictory and to date no equivalent tests have been performed with prototypic materials.

Nationally, research is also continuing to study the possibility of cooling and solidifying the corium by covering it with water or injecting water into a COMET-type device, with a view to designing systems that would avoid basemat penetration in existing reactors. They are being carried out in France as part of the MIT3BAR[132] project in response to the call for RSNR projects launched in 2012 by ANR in the wake of the Fukushima accident, and are coordinated by CEA in partnership with EDF and AREVA.

Examination of Unit 1 of the Fukushima Daiichi power plant could also help to improve knowledge of the interaction between corium and concrete and the effectiveness of cooling mechanisms, by providing information about a situation potentially involving large corium masses at reactor scale. Indeed, accident calculations performed

131. VULCANO Water-Uranium.

132. Assessment and Mitigation of the Risk of Third Containment Barrier Breakthrough at Nuclear Power Plants.

using core melt accident simulation codes like ASTEC, MAAP and MELCOR suggest that the core of Unit 1 was seriously degraded, that the vessel failed and that large quantities of molten corium flowed on to the concrete under the vessel, eroding a significant thickness of it. To understand more about the corium-concrete interaction phase in this reactor, it is necessary to work backwards and analyze the in-vessel degradation phase which, depending on the code used, produces some very different results in terms of composition, temperatures and the physico-chemical properties of the debris and corium [9].

► Possibility of in-vessel corium retention

One countermeasure envisaged by some licensees to try to avoid vessel melt-through in a reactor where the core has melted is to flood the reactor pit in order to cool the pit by organizing a two-phase flow of water around it. The effectiveness of this action depends on many different factors, including the reactor's size and power, the moment when the materials relocated to the lower head, the fraction and composition of the corium in the lower head, which determines the distribution of the flow of heat to the lower head internal wall, and the geometry of the vessel pit and characteristics of the thermal insulation around the vessel. The presence (or not) of specific geometric features (discontinuous thickness between the vessel shell and the lower head, penetrations through the lower head for instrumentation), which determine the flow of heat that can be evacuated by the outer face in the water, should not be ignored.

In a deterministic approach, a robust demonstration of the possibility of keeping the corium in the vessel, known as in-vessel retention (IVR), requires checking that the flow of heat through the vessel can be evacuated at any point on the vessel's external surface. This heat evacuation must be performed without drying out the wall, particularly in areas where the flow of heat is greatest, and this applies to all feasible configurations (stabilized or transient) of the debris bed[133], which are generally inhomogeneous. This depends on the distribution of heat on the vessel wall's internal surface and on the distribution of the critical flow at each point on the vessel's external surface, which depends on a number of parameters as explained above. The distribution of heat on the vessel wall's internal surface is the result of natural convection mechanisms within the different layers or piles of corium at the vessel's lower head, whereas the absolute values are dependent on the length of time before relocation to the lower head and the change in mass of the relocated corium, i.e. ultimately the total residual heat to be evacuated through the lower head wall. The physico-chemical balance of the materials that compose the corium will change and the situation can gradually evolve so that a stratified pool develops in which a non-miscible metal layer floats at the surface of a pool of oxides, which may itself be composed of two phases with different densities[134]. This

133. "Debris bed" is the term used to refer to the corium from the damaged core that has relocated to the lower head. It may contain solidified clumps, and pieces of the structures at the bottom of fuel assemblies (grids, etc.) that have fallen into the corium.

134. The behavior of the corium at the lower head is explained in brief here; it is described in more detail in Section 5.1.2 of the publication "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.

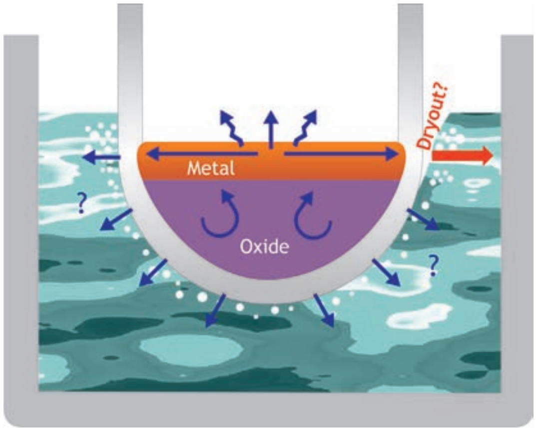

Figure 9.4 Heat flow distribution and risk of dryout © Georges Goué/IRSN.

configuration may be temporary when there is steam in the vessel because the gradual oxidation of the metal phases can produce a more favorable situation in terms of heat distribution, of a homogeneous pool.

If water has not been injected into the vessel – a likely situation in the first few hours of a core melt accident, which may in turn be due to a loss of coolant – the transfer through radiation from the metal layer floating on the oxide pool to the very overheated upper structures of the core is reflected in the fact that the surface temperature of the metal layer is higher than the melting point of steel (the interface temperature between the metal layer and the internal face of the vessel). The metal layer therefore transfers more heat to the vessel due to the imbalance between the interface temperatures. This results in a heat flow concentration, known as a "focusing effect", which causes a heat flow peak in the vessel at the location of the metal layer (Figure 9.4). The thinner the floating metal layer and the larger the mass of oxides under it (from which the residual heat is being released), the greater this peak is. If water is present, the formation of a metal crust on the pool's surface will reduce the interface temperature imbalance and limit this concentration.

Furthermore, given the big thickness of the vessel wall and the intensity of the heat flows to be evacuated through it, the vessel will partially melt starting from its internal face, and its thickness will be reduced to a few centimeters at drying flows of around 1.5 MW/m^2. The corium crust may then become unstable on contact with the liquid steel film, causing the transfer of the underlying molten steel to the corium pool and causing transient heat flows that are locally very intense when the liquid corium once again comes into contact with the wall[135]. From a mechanical point of view, a few centimeters of residual steel will be enough to keep the vessel intact once external cooling is provided and the vessel has been depressurized.

135. This phenomenon has been observed experimentally in tests conducted in the SCARABEE reactor on the total instantaneous blockage (TIB) of fuel assemblies in fast neutron reactors.

Although in-vessel retention appears possible for small, low-power reactors, the highest power level at which it would still be possible without being in danger of causing even more serious consequences is still the subject of uncertainty. If vessel failure should occur when the reactor pit is completely flooded, the ejection of the molten corium in the lower head could cause a steam explosion on contact with the water, which could damage the reactor containment (see Section 9.3.1 further on) and the components contributing to its leaktightness.

This is the reason why France is playing an active part in the European In-Vessel Melt Retention (IVMR – H2020) project, coordinated by IRSN with the participation of AREVA, CEA, EDF and 19 other European study and research bodies, with the aim of assessing whether a strategy of this kind could be successfully applied to Europe's existing 1000 MWe reactors (pressurized water reactors, boiling water reactors, VVER reactors). A common methodology will be defined for the project, which is expected to finish in 2019, for analyzing in-vessel retention capabilities and to complete experimental data in order to reduce uncertainty related to phenomena about which still little is known.

In the context of IVR, the effectiveness of core reflooding at a particular moment determines the mass of corium that will relocate to the lower head and therefore the absolute value of the residual heat to be evacuated from the lower head. IRSN has begun some research on this theme, specifically concerning:

- modeling of the geometric characteristics of the different possible debris bed configurations (porosity, exchange surfaces, permeability, etc.);
- modeling of the thermohydraulic phenomena likely to occur during reflooding for these different configurations;
- modeling of the oxidation phenomena of the metal-oxide mixtures in corium and of hydrogen production;
- impact studies of how all these phenomena interact.

The development and validation of the models rely in particular on the PRELUDE tests and the tests currently being performed by IRSN on a larger scale at the PEARL facility as part of IRSN's analytical experimental program on debris bed reflooding (PROGRES[136]) (see Section 9.1).

Concerning focusing effect risks, IRSN is working to identify the conditions in which the layer of metallic materials can float for a long period of time on the oxide layer, and for this it is studying the kinetics of transition between these layers, taking particular account of crust formation at the interfaces and the oxidation of the metal phases. The experiments on this subject are being run as part of the CORDEB program mentioned in Section 9.1 and will continue at the facility used for this program once it is finished (2016–2019), as part of the European IVMR program coordinated by IRSN.

136. Analytical Axperimental Program on Debris Bed Reflooding.

Finally, concerning interaction between the corium and water, research activity has been combined within the ANR RSNR ICE[137] project, which will be discussed in Section 9.3.1 further on.

An in-vessel retention strategy was adopted in Finland in 1995 when the power of the VVER-440/213 reactors at the Loviisa nuclear power plant was increased to 510 MWe, by installing a system of mobile insulation to create a cooling space around the vessel and by modifying the reactor pit ventilation system to create a cooling system. Experimental programs (COPO[138] in Finland and ULPU[139] in the United States) and numerical simulations were used to check that the cooling system performance was sufficient, taking into account bounding heat flow distributions at the lower head.

As far as new reactors are concerned, the in-vessel retention strategy has been adopted for the AP600. Given the power level of this reactor (600 MWe), the margins were considered adequate, even in the absence of water injection into the vessel, to validate the demonstration, allowing the U.S.NRC to certify the reactor in 1999. The same approach was used for the AP1000, which was certified by the U.S.NRC in 2005. It should be noted that there are no penetrations through the lower head for instrumentation in the AP600 and AP1000 reactors.

Three types of information were taken into account by the U.S.NRC:

– deterministic data;
– assessments giving sufficient confidence that if the vessel were to fail – particularly due to the focusing effect – there would be no cliff edge effect, or more specifically the reactor pit and the basemat could then provide their confinement function, withstanding any overpressure that might be caused by interaction between the molten materials and water[140];
– probabilistic data, integrating the risk of vessel melt-through and early loss of containment[141].

Interestingly, IVR is also the strategy chosen by the designer of the VVER 1000, which still provided an external core catcher (TianWan reactor in China and Kudankulam reactor in India).

137. Corium-Water Interaction.
138. Corium Pool Facility.
139. An IVR-related full-scale boiling heat transfer facility at University of California, Santa Barbara.
140. For the AP1000, this was the subject of an analysis presented in the document *Analysis of In-Vessel Retention and Ex-Vessel Fuel Coolant Interaction for AP1000* – NUREG/CR-6849, ERI/NRC-04-201 – August 2004. For the AP600, refer to the report *AP600 Final Safety Evaluation Report Related to the Certification of the AP600 Design* – NUREG-1512, 1998.
141. According to US regulations, probabilistic assessments must show that the frequency of massive early releases (less than 24 hours after an accident) is less than 10^{-6}/year.reactor. Given the uncertainties surrounding the models for corium progression to the lower head, for the AP1000 the designer assumed a conditional probability of vessel failure of between 4% and 30%, and postulated a systematic failure of the containment with a massive release into the environment. Since the calculated frequency of containment failure is 1.9×10^{-8}/year.reactor, with 38% of failures leading to early releases, the designer concluded that its design was well below the limit.

9.3. Dynamic loading of the containment by a sudden increase in internal pressure

This kind of phenomenon is likely to cause a relatively early loss of integrity of the containment, followed by a massive radioactive release into the environment. In line with the classification proposed by Norman C. Rasmussen in the early 1970s (Figure 9.5), we will distinguish between steam explosions (failure mode "α") and hydrogen explosions (failure mode "γ"); direct heating of the containment gases (Direct Containment Heating [DCH]) should also be taken into account.

9.3.1. Steam explosions

Steam explosions can occur when two fluids come into contact, where one (in this case the liquid corium from the core melting) is at a higher temperature than the boiling point of the other (the coolant). It is a thermodynamic reaction. When it enters the water, the corium jet fragments and causes intense vaporization of the water. This vaporization can create a shock wave that fragments the corium even more finely, further increasing the contact area between the two fluids and, as a result, the heat transfer. The process can propagate in the corium-water mixture, triggering an explosion. Simple contact

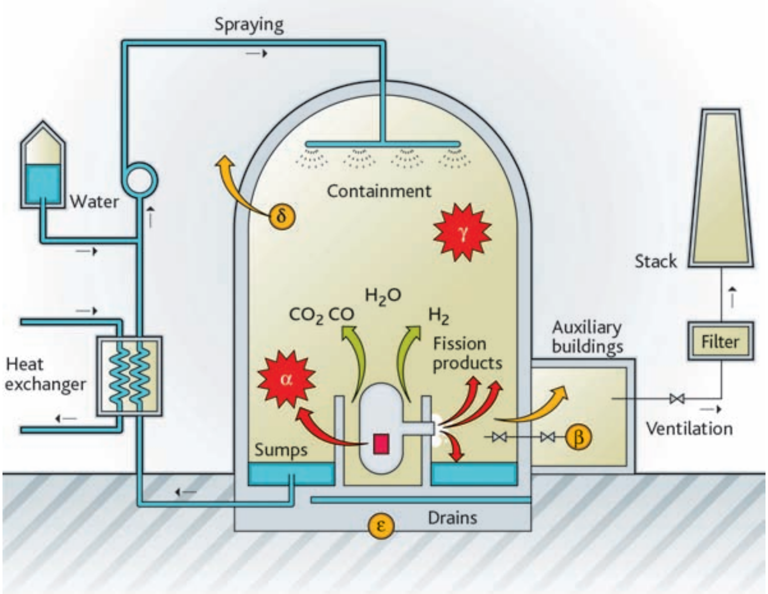

Figure 9.5 Schematic diagram of the possible containment failure modes according to the Rasmussen report [1].

between the two fluids is not enough to cause such an explosion. If the hot fluid fragments into quite large pieces (measuring millimeters or centimeters), the heat transfer will not be fast enough or intense enough, and will merely cause steam production characterized by a slow pressure increase in the system. If the hot fluid stays liquid on contact with the cold fluid due to the formation of a steam film and significant confinement restricting steam production in the first few moments, this is likely to trigger a very forceful explosion.

Fine fragmentation and explosion phenomena were initially studied in the 1970s as part of analyses of core melt accidents likely to occur in fast neutron reactors with sodium as coolant. From the late 1980s, this research was extended to PWRs. In France, analytical test programs were conducted by CEA on behalf of IPSN (then IRSN) and led to the development of heat transfer models.

The BILLEAU tests (which ended in 1997) were used to characterize heat transfer on the surface of solid metallic spheres heated to 2200 °C and plunged into water. The TREPAM program (which ended in 2003) consisted of measuring the heat flow on the surface of a tungsten filament plunged into water, for different filament diameter values (10 to 250 µm), filament temperatures (1080 °C to 2630 °C), water pressure values (0.1 to 21 MPa), water undercooling values (0 to 80 °C below the water saturation temperature) and plunge speeds into the water (0.2 to 46 m/s).

Abroad, tests have been conducted with more representative materials, sometimes on a larger scale. These include the following programs:

- CCM[142], run by the Argonne National Laboratory (United States) in the early 1990s, which studied the fragmentation of molten mixtures containing 2 to 12 kg of UO_2, ZrO_2 and steel at 2800 °C;
- SUW[143] and WUMT, run at Winfrith (United Kingdom) at the same time, with a uranium-based thermite[144];
- FITS[145], run at the Sandia National Laboratories (United States) with an aluminum-based thermite;
- ALPHA, conducted by the Japan Atomic Energy Research Institute (JAERI), Japan, with the same material;
- FARO, run by the European Commission's Joint Research Centre at Ispra (Italy), which ended in 1999 and studied, in the TERMOS vessel, the fragmentation of molten mixtures of 18 to 176 kg of UO_2 and ZrO_2 and the steam explosion resulting from dropping this into water, under pressures of 0.2 to 5 MPa;

142. Cold Crucible Melting.
143. Scale-Urania-Water.
144. The corium is represented by a mixture consisting of an oxide and a molten metal obtained by means of an exothermic oxidation-reduction reaction like the reaction between iron oxide, Fe_2O_3, and aluminum (aluminothermic).
145. Fully Instrumented Test Series.

- KROTOS, originally run by the same research laboratory, then by CEA after 2004 in a shock tube (one-dimensional geometry), to study fragmentation on contact with water of a liter of various molten materials (tin at 1000 °C, alumina from 2300 °C to 2800 °C and a UO_2-ZrO_2 mixture at 2800 °C), and the steam explosions that occur as a result;
- PREMIX, run by the research center at Karlsruhe (Germany), which studied the fragmentation of 16 to 60 kg of aluminum-based thermite producing molten alumina brought to 2300 °C;
- ECO[146], conducted at the same laboratory, which studied the explosion resulting from contact between 6 to 18 kg of molten alumina brought to a temperature of 2300 °C and one to two meters of water.

The results of these various research programs provided a large amount of experimental data. However, the results were fairly diverse: some tests produced explosions but others did not, for no apparent reason except the physical properties of the materials used (the molten alumina had a greater tendency to cause explosive interactions than the UO_2–ZrO_2 mixture, the thermal conductivity of which is lower).

On the basis of the analytical tests, IRSN and CEA together developed the MC3D[147] steam explosion simulation code. The code uses an eulerian description with a mixed finite volume (mass and energy balances) and finite difference (momentum balance) method with a three-dimensional (cartesian or cylindrical) structured mesh. It simulates both the fine fragmentation and the explosion phases. The complexity of the phenomena to be described during fragmentation requires very detailed modeling with precise numerical schemes. The fuel behavior is described using a three-field model:

- a continuous liquid field,
- a field for droplets from fragmentation of the continuous field,
- a fine solid fragments field.

The Rayleigh-Taylor instabilities at the interface between the corium and the water are also modeled.

To advance the modeling of the phenomena, the OECD/NEA launched the international SERENA project (2001–2005) [10], to which France (IRSN and CEA) contributed alongside Germany, Japan, Korea, the United States and Russia. Its objective was to determine the state of the art of knowledge about steam explosions, to compare simulation codes with one another, to assess their ability to predict explosions and to identify the areas where uncertainty needs to be reduced.

The OECD/NEA's international SERENA-2 program [11], which succeeded it (2005–2012) included 12 tests conducted by CEA at the KROTOS facility, which was moved from Ispra to Cadarache at the end of the FARO program, and by the Korea Atomic Energy

146. Experiments on energy COnversion during a steam explosion.
147. Multiphasic and thermohydraulic 3D simulation code.

Research Institute (KAERI) at the TROI[148] facility in Daejon, South Korea. The objective was to gain a better understanding of the mechanisms that trigger steam explosions and the effect of the type of materials used to represent corium, and to check the models' capacity to reproduce effects related to the geometry of the interaction zone.

The tests run at the KROTOS facility consisted of injecting 5 kg of molten corium (UO_2–ZrO_2 mixture but also more realistic compositions including the metallic phases or iron oxide) in a tube (one-dimensional geometry). An X-ray cinematography system was used to film the fragmentation of the jet in real time and the vaporization of the water around the corium fragments. In the TROI facility, larger corium masses of 20 kg (identical composition to those at KROTOS in the frame of the SERENA program) were injected into a container of similar height and width (two-dimensional geometry representative of a PWR reactor pit). The tests were defined after preliminary calculations had been made with the various simulation codes. The analysis of the results was discussed by international experts. The program also included code benchmarking exercises and reactor-scale applications.

The SERENA and SERENA-2 programs advanced understanding of the fine fragmentation of a corium jet in water and the explosions that can be caused by this. They revealed the influence of corium fragment solidification kinetics on the magnitude of the explosion (rapid solidification restricts the fine fragmentation process, and therefore heat transfers and water vaporization). In general, the simulation codes developed globally give a good overall prediction of the mechanical energy evaluated in tests using molten alumina and overestimate the mechanical energy evaluated during tests with molten corium. An analysis of the images taken during tests at the KROTOS facility shows that instabilities of the steam film surrounding the corium jet are correctly reproduced by the MC3D code's models. The code also correctly predicts the attenuation of the shock wave when the explosion occurs in a two-dimensional geometry like those studied in the TROI facility.

However, the effects related to corium oxidation during the interaction are still not properly understood. The research, particularly on fragmentation of the jet and oxidation of the corium, is continuing with theoretical and experimental analytical studies as part of the ICE project (2013–2017), launched in response to ANR's call for RSNR projects mentioned above and coordinated by IRSN in partnership with CEA, EDF and Nancy University.

9.3.2. *Hydrogen-related risks*

Hydrogen, which is produced mainly by oxidation of the zirconium in the cladding during the core degradation phase, but also by the oxidation of the other metals in the corium pool or in the basemat during the interaction phase between the corium and the concrete, can accumulate in the containment. The amount of movement of the atmosphere in the containment has an impact on the hydrogen distribution, which may therefore not be uniform. Locally, the hydrogen can reach high concentrations that

148. Test for Real cOrium Interaction with water.

exceed the threshold for the gaseous mixture to be flammable; hydrogen detonation in a containment is an event to be avoided.

The hydrogen distribution in the containment is the result of thermohydraulic phenomena which are complex to study due to the number of fluids in the containment atmosphere to be taken into account and the containment's compartmentalized geometry: convection caused by temperature differences between the atmosphere and walls and the possible use of spraying, condensation of steam on the containment walls with the water droplets produced by the spraying, diffusion, turbulence, the effect of the hydrogen recombiners, etc. [12, 13].

To assess the risk of hydrogen combustion in the containment, IRSN uses its own validated simulation codes (ASTEC, TONUS and the FLUENT code pending the implementation of P^2REMICS) and criteria from its own research programs, enabling it to determine which configurations are potentially dangerous. These tools enable it to predict hydrogen distribution and combustion, taking account of the effect of mitigating means of core melt such as recombiners, spraying and the venting system.

To predict hydrogen distribution inside the containment, IRSN uses two approaches: modeling based on several compartments in which fluid concentrations and temperatures are uniform, and more detailed multi-dimensional modeling with a large number of meshes. The first approach, which is economical in terms of calculation time, is used in the ASTEC code. The second is used in the TONUS code developed by IRSN in collaboration with CEA[149], and in FLUENT. The two approaches are complementary: the first approach is used to identify scenarios requiring detailed analysis using the second approach.

The validation of these simulation codes is reliant upon a number of experimental programs being conducted in France or abroad, at well-instrumented facilities simulating the geometry of a reactor containment on different scales:

- the PANDA facility (Figure 9.6a), at the Paul Scherrer Institute (PSI) in Villigen, Switzerland, consisting of four compartments with a total volume of 460 m^3, allowing the shape of fluid flows to be studied among other things;
- the THAI[150] facility (Figure 9.6b), run by Becker Technologies, in Eschborn, Germany, consisting of several compartments with a total volume of 60 m^3, with thermally insulated walls, designed for studying the distribution and combustion of hydrogen, and the operation of catalytic hydrogen recombiners;
- the MISTRA facility (Figure 9.6c) run by CEA in Saclay, with a cylindrical internal volume of 100 m^3, suited to the study of steam condensation on the walls and spray droplets;
- the TOSQAN facility (Figure 9.6d) run by IRSN in Saclay, with a cylindrical internal volume of 7 m^3, dedicated to the study of condensation and spraying, and of thermal and mass exchanges with the sump water.

149. EDF is developing the SATURNE code.
150. Thermal-hydraulics, Hydrogen, Aerosols and Iodine.

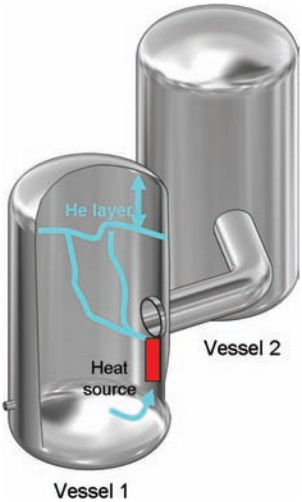

Figure 9.6a Diagram of the PANDA facility. © PSI.

Except in the THAI facility, for safety reasons hydrogen is simulated by helium, a gas with very similar physical properties but with the advantage of not being combustible. Tests carried out at the THAI facility have confirmed the similarity of hydrogen and helium distributions in the test volume.

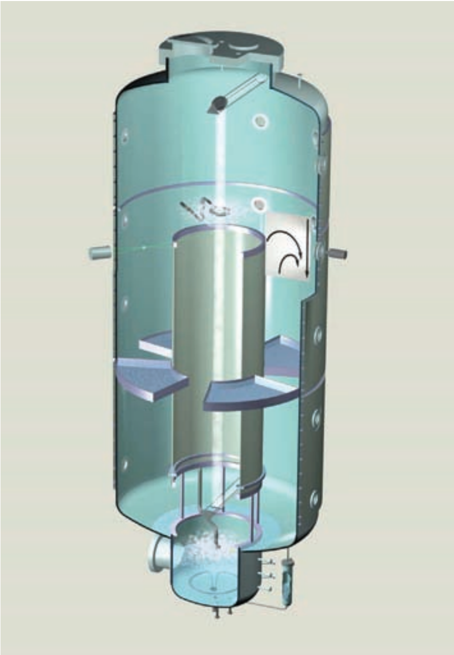

Figure 9.6b Diagram of the THAI facility [14].

Figure 9.6c View of the MISTRA facility. © A. Gonin/CEA.

Many national and international projects and programs have collected a large amount of experimental data, enabling the scope of validation of the simulation codes to be extended. In particular these include the international OECD/NEA programs SETH (SESAR THermalhydraulics, 2000–2004) at the PANDA facility, SETH-2 (2007–2010) using both the PANDA and MISTRA facilities, THAI (2007–2009) and THAI-2 (2011–2014) at the THAI facility. The SETH-2 program studied the phenomena that cause the destabilization of stratified configurations, due to the effect of heat released by the hydrogen recombiners and spraying; because of its low density, the hydrogen is initially concentrated in the upper part of the containment.

Effects of scale have also been studied as part of the ERCOSAM[151]-SAMARA program (2010–2014), supported by the European Commission and by ROSATOM[152] (Russia), in

151. Containment thermal-hydraulics of current and future LWRs for Severe Accident Management.
152. Federal Agency on Atomic Energy.

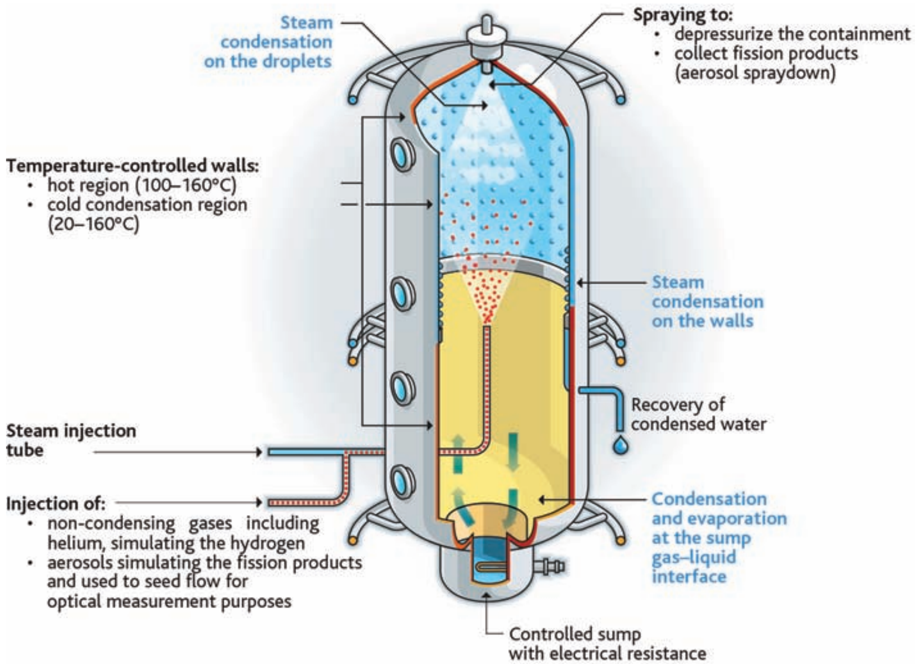

Figure 9.6d Diagram of the TOSQAN facility. © Stéphane Jungers/IRSN-source IRSN.

particular including tests at the PANDA, MISTRA and TOSQAN facilities already mentioned, as well as at the SPOT facility (60 m^3) in Russia. The extrapolation of the results to large scales was performed on the basis of preliminary calculations. It could be confirmed in the future by means of tests at the 2000 m^3 Russian KMS facility (currently at the design stage).

Concerning combustion, it should be noted that a ternary mixture of air, water and hydrogen (Figure 9.7), considered to be flammable, can lead to different combustion regimes, which depend mainly on the maximum hydrogen concentration reached and the distribution of this gas between the compartments:

– laminar deflagration,
– rapid deflagration,
– the deflagration-detonation transition (DDT).
– detonation – a phenomenon that can have disastrous consequences and that should therefore be avoided.

In a mixture known to be flammable, combustion may be triggered by an energy source of a few millijoules. Consequently, in the presence of electrical power sources or hot points, it seems probable that ignition would occur rapidly once the gas mixture enters the flammability domain. In contrast, a much more powerful energy source (at least 100 kJ) is required to trigger a stable detonation. This explains why direct detonation of the gaseous mixture in the

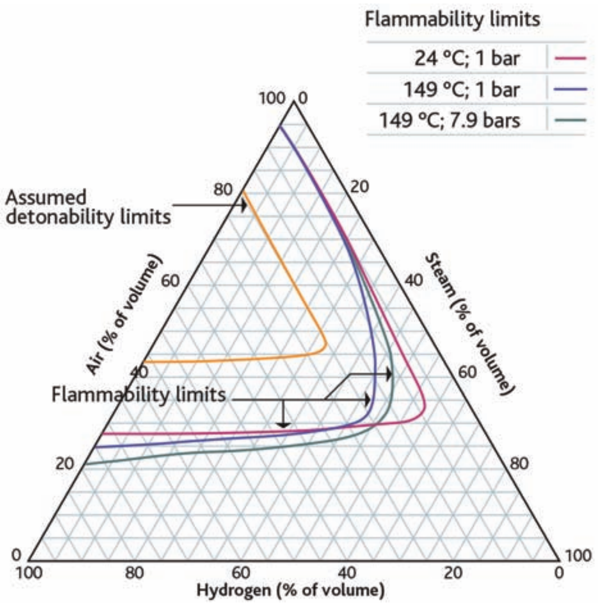

Figure 9.7 Shapiro diagram for hydrogen, air and steam mixtures (please note that the detonability limits shown have been called into question by later studies). @ DR.

containment can be ruled out; the only mechanism considered likely to cause detonation is flame acceleration and the deflagration to detonation transition.

Two types of criteria have been defined and adopted:

– the criterion "σ" related to flame acceleration. The quantity σ 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^* above which flame acceleration is possible depends on the initial temperature of the gases and the stability of the flame; it has been determined using the results of many experiments at different scales and in different geometries;

– the criterion "λ": similarly, prerequisite conditions have been defined for assessing the possibility of a deflagration–detonation transition (DDT). They are based on comparison of a characteristic dimension of the space being studied with the detonation cell size (denoted λ), which characterizes the mixture's reactivity.

Flame propagation after ignition, in an air, steam and hydrogen mixture, has been a study theme of many experimental programs [15]. The purpose of these tests is twofold: to study the transition between combustion modes and to produce a database of experimental data for validating simulation codes.

On this theme, IRSN is engaged in a long-term collaboration with the CNRS Institute for Combustion, Aerothermal Engineering, Reactivity and Environment (ICARE) in

Orléans. Besides determining the flammability limits of the mixtures, work done as part of this collaboration has enabled the flame acceleration criterion "σ" to be refined.

An analytical test program has therefore been run since 2004 at ICARE's ENACCEF[153] facility (Figure 9.8). This facility was designed to represent, on a 1/24 scale, a steam generator bunker and the containment dome of a 900 MWe PWR. ENACCEF consists of a dome with a volume adjustable between 781 and 958 liters (representing the free volume of the containment above the steam generator), connected to a cylindrical tube that is 3.2 m high, has an internal diameter of 0.17 m (representing the steam generator bunker) and can contain obstacles, in which flame acceleration can be studied. The instrumentation it contains includes photomultiplier and pressure sensors to measure the progression of the flame front and the pressure generated along the tube. In addition, the composition of the gaseous mixture is analyzed at several places in the test section; laser Doppler velocimetry (LDV) and particle image velocimetry (PIV) measurement techniques are used to measure the velocity field of the flow before the flame arrives.

The tests have been used to refine the expression of the critical values of expansion factor "σ" for homogeneous mixtures and extend it partially to heterogeneous mixtures.

Tests on a larger scale had also been conducted previously in the 62 m-long RUT facility at the Kurchatov Institute in Moscow, Russia. This facility consists of a straight canyon connected to a cavity, which is itself connected to a second, shorter canyon curved at the end. All three zones are rectangular in cross-section and may be obstructed

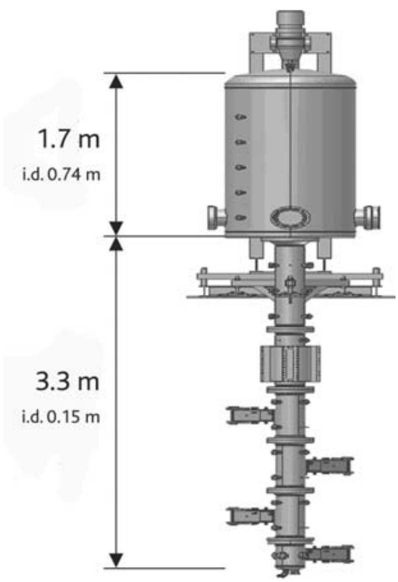

Figure 9.8 Diagram of the ENACCEF facility [16].

153. Flame Acceleration Enclosure.

by obstacles. This geometry can be used to study both monodirectional flame acceleration in the canyons and more complex 3D effects or interactions in the cavity. The mixtures used contained hydrogen, air and sometimes a diluent (steam). A test program run on behalf of IPSN in partnership with German researchers at the Karlsruhe center (FzK) studied the different combustion regimes and obtained values for the criteria "σ" (flame acceleration) and "λ" (detonation cell size), which are used for predicting different combustion regimes in the ASTEC and TONUS simulation codes[154].

This program was completed by the HYCOM (integral large scale experiments on HYdrogen COMbustion for severe accident code validation) program, run from 2000 to 2003 with support from the European Commission. Its aim was to study flame acceleration in mixtures of hydrogen and air (without water), particularly validation of the "σ" criterion. The effect of the expansion of the burned gases (piston effect) and the effect of compartmentation were studied in the 12 tests conducted in the RUT facility. The effects of pressure vents, concentration gradients and changes in cross-section were studied in 46 more analytical tests run at the smaller-scale DRIVER and TORPEDO facilities at the Karlsruhe center in Germany. These facilities consisted of two cylindrical tubes with diameter 174 mm and 520 mm respectively and length 12 m and 6 m respectively. The program, in which IPSN (then IRSN) played an active part with the support of EDF, also included a reactor-scale simulation code benchmark exercise, studying a containment geometry similar to that of EPR.

Research on hydrogen distribution in a reactor containment (effects of spraying and recombiners – see below), on combustion (deflagration to detonation transition on a large scale) and explosion is continuing as part of the MITHYGENE[155] program (2013–2018). The program was launched in response to ANR's call for RSNR projects mentioned earlier, and is coordinated by IRSN in partnership with CEA, ICARE, AREVA Expansion, the German Jülich research center (KfJ), EDF and Air Liquide.

To reduce the hydrogen content of a containment in a core melt accident, passive catalytic hydrogen recombiners (Figure 9.9) have been installed in all the containments of French reactors.

The behavior of recombiners in situations representative of core melt accidents has been the subject of a number of test programs, conducted primarily by manufacturers (Siemens, AECL, etc.). IPSN ran the H2PAR[156] test program (1998 to 2000) in partnership with EDF, to study the possible poisoning of catalytic sites in recombiners by fission products, particularly the halogens. The facility consisted of an oven that could bring approximately 10 kg of a UO_2–ZrO_2 mixture up to a temperature of 2900 °C. Inert compounds representative of the main fission products were included in this mixture. The aerosols and the gases released by the molten mixture spread inside a 7.6 m^3 containment made from Terphane, in which a mock-up representing a hydrogen recombiner used in French reactors was installed. The tests also provided an opportunity to study the risk of the hydrogen in the recombiner igniting. The atmosphere in the

154. The equivalent German code, COM3D, uses equivalent criteria.
155. Hydrogen Mitigation.
156. Hydrogen Passive Autocatalytic Recombiners.

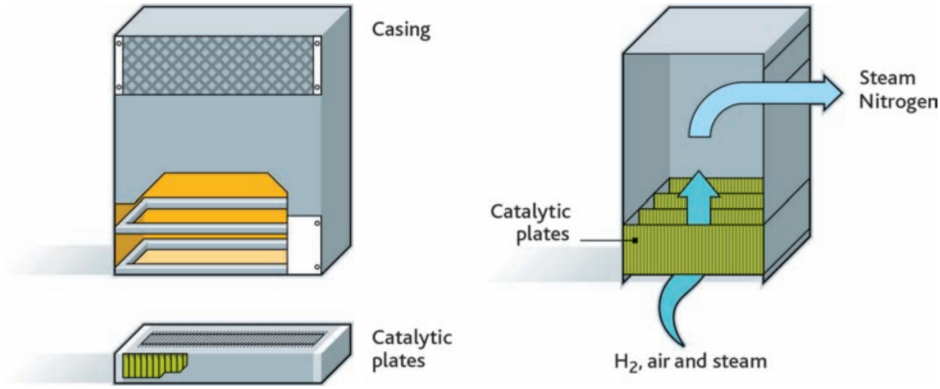

Figure 9.9 Diagram of a passive catalytic hydrogen recombiner. © IRSN.

containment contained a mixture of air, hydrogen and steam; the proportions were one of the parameters studied. The impact on recombination performance of various geometric parameters (number of catalytic plates, height of stack) and chemical parameters (several catalytic plates replaced by neutral chemical plates) was also studied.

The KALI H2 test program, run by CEA at the end of the 1990s in partnership with EDF, also provided an opportunity to study the effects of humidity, exposure to smoke from cable fires and carbon monoxide on recombiner performance. The results of this program revealed that humidity has little impact on recombiner performance. They also showed that recombination was deactivated by carbon monoxide in low-oxygen atmospheres and that there was a loss of performance of approximately 5% for recombiners exposed to cable fire smoke. The fire on May 10, 2011 in the containment of reactor 2 at the Ringhals nuclear power plant in Sweden also revealed the possibility that catalytic recombiner plates could be perforated by chlorinated products from the combustion of plastics. Investigations are therefore being carried out by IRSN and the Jülich research center to study the effect of cable fires on recombiners. The first results of this study have revealed that the start of recombination is delayed in the case of catalytic plates exposed to smoke from cable fires [17].

During the Phebus FPT3 test, specimens from recombiner plates made by various manufacturers were exposed for several dozen minutes to the radioactive aerosols and steam in the tank simulating the containment. The low oxygen content of the tank (for safety reasons) slightly complicated the analysis of the results, which nevertheless not revealed unexpected phenomenon.

To investigate the effect of a lack of oxygen, of carbon monoxide or of the presence of dust on the startup and effectiveness of recombiners, IRSN is working with the Jülich research center in Germany, which has the REKO experimental platform with a number of facilities on different scales (Figure 9.10). On the basis of the experimental results, IRSN is developing the SPARK simulation code, which can be used to take account of all the physico-chemical phenomena governing recombination. This collaboration has

Figure 9.10 View of the REKO platform. © Forschungszentrum Jülich.

already enabled limits to be defined for hydrogen ignition by recombiners. These results were confirmed by the test program conducted as part of the OECD THAI and THAI-2 projects. The findings of these various tests allow the improvement of the models integrated into the simulation codes used at IRSN (including ASTEC) and as a result to refine the assessments of hydrogen explosion risk in the annulus space between dual containments, which IRSN is required to produce for 1300 MWe and 1450 MWe reactors, and in fuel storage buildings (and the ITER facility).

9.3.3. Direct containment heating

Direct containment heating (DCH) was originally studied experimentally and analytically by the Americans in the 1990s for the PWRs at the Zion, Surry and Calvert Cliffs sites. If a reactor vessel fails under pressure, corium could be ejected and fragmented in the reactor pit and the fragments could be carried into the containment. The metallic components of corium (such as zirconium and iron) would oxidize on contact with the steam and air in the containment, releasing heat and producing a quantity of hydrogen that would be added to the amount already produced by oxidation of the cladding at the start of the accident but not recombined. Direct containment heating due to radiation and convection, and hydrogen combustion, would cause a sudden increase in pressure inside the containment which, depending on the mass of corium ejected and hydrogen produced, would threaten its integrity.

The American studies concluded that the corium dispersal and the amplitude of the resulting overpressure were very dependent on the geometry of the reactor pit. Analytical tests in which the corium was simulated by water and the geometry studied (reactor pit and communication channels with the free volume of the containment) was that of a 900 MWe reactor, were conducted at KAERI in South Korea on behalf of IPSN in the late 1990s. The results enabled an empirical model to be created that was integrated into the ASTEC simulation code, giving the fraction of corium ejected from the reactor pit as a function of the geometric parameters and ejection conditions.

In the early 2000s, IRSN launched a new research program in partnership with German researchers in the DISCO[157] facility at the Karlsruhe research center, consisting of two experimental devices. The first was a plexiglass mock-up enabling the flows to be seen, with the corium simulated by water, gallium or Wood metal[158], an alloy with a low melting point. The second was used for more representative experiments, where the corium was represented by a mixture of 10 kg of molten alumina and iron obtained using an exothermic oxidation-reduction reaction, brought to approximately 2700 °C. Several geometries were studied in these devices: those of an EPR, a P'4-type reactor (1300 MWe), a VVER-1000 and a German KONVOI PWR. The program, which was funded by IRSN[159], consisted of 60 analytical tests at low temperature and 12 integral tests at high temperature on a 1/16 scale model of the containment of a P'4 reactor representing the geometry of the reactor pit, the compartments in communication with the reactor pit and the dome.

The experimental results were analyzed using the MC3D simulation code (Section 9.3.1). The analysis established a correlation giving the corium fraction dispersed axially in the reactor pit as a function of the pressure in the vessel at the moment when it fails. Another finding is that the pressure increase in the containment is mainly due to combustion of the hydrogen produced during dispersal of the corium and, to a lesser extent, of the hydrogen already in the containment when the vessel failed. The rate of hydrogen production after vessel failure is correlated with the fraction of corium dispersed axially in the dome.

These results have significantly improved the direct containment heating model in the ASTEC simulation code, enabling it to be applied with greater confidence to the geometry of French PWRs.

157. DISpersion of simulated COrium.

158. Wood metal is an eutectic alloy of bismuth, lead, tin and cadmium. Its melting point is approximately 70 °C.

159. Except in the case of the KONVOI reactors.

9.4. Radioactive releases

The amplitude and nature of radioactive releases depend on three main processes:

– the emission of radioactive products by the fuel when it is damaged,

– the transporting of these radioactive products into the reactor's circuits,

– their behavior in the containment.

The volatility of the radioactive products depends on their physical state, which in turn is determined by their chemical nature. Chemical reactions therefore play a key role, bearing in mind that they occur at a very wide range of temperatures (from 50 °C to 3000 °C) and involve a large number of elements: fission products, the fuel and the core's structural materials (zirconium, steel, materials that absorb neutrons such as AIC and/or boron carbide, etc.). These materials release a considerable mass of steam and aerosols, which can be much higher than the mass of all the fission products put together. The nature of the materials used in the walls of the circuits and containment also plays a part.

The research (analytical experiments) conducted specifically on the above three processes is discussed in more detail in the sections that follow. The findings concerning each of these processes from the more general (integral) tests in the Phebus-FP program will then be explained.

The reference information [18] provides a summary of the knowledge acquired about radioactive releases in the event of a core melt accident.

9.4.1. Emission of radioactive products by fuel

The emission of fission products by the fuel while it is being heated has been studied in France as part of the HEVA[160] program (eight tests in the 1980s) and VERCORS[161] program (17 tests from 1989 to 2002) run by CEA at its Grenoble center on behalf of IPSN and EDF. At around the same time, similar programs were being run abroad:

– the HI[162] tests (six tests) and VI tests (seven tests) at the Oak Ridge National Laboratory (ORNL), United States,

– the VEGA[163] tests (10 tests) by the Japan Atomic Energy Research Institute (JAERI), Japan,

– the AECL-CRI tests (more than 300 tests) by Atomic Energy of Canada Limited (AECL), at the Chalk River Laboratories, Canada.

All these tests consisted of heating, in an oven, specimens of fuel that had been in a reactor, to identify the fission products released and measure the quantities, using in-line gamma-ray spectrometry and other radiochemical analysis techniques on samples. The main parameters studied were the type of fuel, its burnup rate, whether or not there was

160. Helium, Steam.
161. Realistic Verification of the Containment of Reactors.
162. Horizontal Induction.
163. Verification Experiments of radionuclides Gas/Aerosol release.

cladding present, the maximum temperature reached during the test and the composition of the gas flowing around the fuel: a mixture of steam and hydrogen in variable proportions, inert gases, air.

In France, the first six tests in the VERCORS program (VERCORS-1 to 6) were conducted between 1989 and 1994 on samples of UO_2 fuel heated to 2300 °C. Then, two test series, VERCORS-HT (three tests) and RT[164] (eight tests), at higher temperatures up to the melting of the fuel, followed between 1996 and 2002. These tests used UO_2 and MOX fuels in the form of fuel rod segments or a debris bed. The device used for the HT tests included a cooled, instrumented channel downstream of the fuel, allowing the behavior of the fission products emitted during their transfer in a reactor coolant system to be studied, and in particular their deposition on the walls.

The fuel specimens generally consisted of a fuel rod segment containing three fuel pellets and their cladding, sampled from fuel rods that had spent several years in an EDF reactor. In most of the tests, the segment was re-irradiated for several days in a research reactor (the SILOE reactor in Grenoble, then the OSIRIS reactor at the Saclay center once the SILOE reactor had been shut down) in order to restore the inventory of short-lived fission products such as iodine-131 that would have disappeared after the rod was unloaded from the core of the origin reactor.

The tests enabled a database of very precise experimental data to be obtained on the emission kinetics of fission products and actinides in different configurations. This database was used to develop and validate empirical models integrated into the ASTEC simulation code. A more sophisticated simulation tool, MFPR[165], based on a more detailed description of the physical and physico-chemical phenomena, was also developed by researchers at the Nuclear Safety Institute of the Russian Academy of Science (IBRAE) in Moscow in collaboration with IRSN; it is used to analyze the results of the tests.

It emerged from all this research that it is possible to distinguish broadly speaking between the following:

- the volatile fission products such as the isotopes of krypton (Kr), xenon (Xe), iodine (I), cesium (Cs), rubidium (Rb), tellurium (Te), and antimony (Sb), which are almost entirely released by the fuel when it reaches 2300 °C;
- the semi-volatile fission products, such as the isotopes of molybdenum (Mo), barium (Ba), yttrium (Y), rhodium (Rh), palladium (Pd), and technetium (Tc), the emission rate of which lies between 10 and 100% and depends heavily on the oxygen potential of the fluid flowing along the fuel being studied[166];
- the low-volatile fission products such as the isotopes of strontium (Sr), niobium (Nb), ruthenium (Ru), lanthanum (La), cerium (Ce), europium (Eu), neptunium (Np)

164. Release of Transuranics.
165. Module for Fission Product Release.
166. The molybdenum oxides are much more volatile than the metal molybdenum, which explains why greater quantities are emitted in steam; the opposite is the case with barium, which is emitted more readily in its metal form in hydrogen.

and some actinides such as uranium (U), of which 10% maximum is generally emitted, with the notable exception of ruthenium, which is more volatile when the fuel is exposed to air rather than steam;

– the non-volatile fission products and the actinides such as the isotopes of zirconium (Zr), neodymium (Nd) and plutonium (Pu), of which no emissions were really detected during the different tests.

The VERCORS facility was dismantled and a new facility that performed the same functions, VERDON, was built by CEA on the Cadarache site. The facility is used to study not only the release, but also the transporting and deposition of the fission products in thermal-gradient tubes. Tests with MOX and UO_2 fuel with high burnup rates, in a mixture of hydrogen and steam, and a test in the presence of air, were performed in this new facility as part of the International Source Term Program (ISTP, 2005–2013).

The emission of fission products during core reflooding was studied experimentally in the LOFT-FP tests already mentioned in Section 9.1. They revealed significant releases of volatile species (iodine and cesium isotopes) during runaway oxidation of the cladding. Tests were also carried out in the early 2000s as part of the ISTC 1648 research program run by the Scientific Research Institute of Atomic Reactors (NIIAR) in Russia, on sections of irradiated fuel rods from a VVER reactor heated to 1700 °C.

The emission of steam and of aerosols released by control rods containing a silver, indium and cadmium alloy (AIC) has been modeled by means of measurements made in the 1990s during specific tests (EMIS[167] program) conducted by CEA in Grenoble on behalf of IPSN on material specimens brought to melting point. AIC specimens were also vaporized during certain VERCORS tests (EMAIC[168] tests) to study their influence on the chemistry of the fission products released.

Once emitted by the core, the radioactive products and elements from the degradation of the core structures undergo rapid cooling. The vapors condense, forming aerosols (homogeneous nucleation), or condense on the aerosols already formed (heterogeneous nucleation) and on the reactor coolant system walls. The vapor of certain fission products such as cesium and iodine can also react chemically with the materials in the RCS walls (Ni, Cr, Fe) and be fixed, to a greater or lesser extent reversibly. The heating of the wall due to the heat released by the fission products or the partial pressure reduction of these elements in the fluid in the RCS can cause some of the deposited fission products to be re-emitted.

9.4.2. Transporting of radioactive products into the reactor's circuits

The mechanisms governing the transporting – and deposition – of aerosols in a circuit has been the subject of many experimental and theoretical studies, often conducted outside the nuclear field. This work has allowed basic models to be established that have

167. Fission Product Emission.
168. Silver, Indium and Cadmium Emission.

been integrated into severe accident simulation codes such as ASTEC. It involves the agglomeration of the aerosols contributing to an increase in their size, thermophoresis and diffusiophoresis[169], phenomena responsible for deposition in the presence, respectively, of a temperature gradient between the carrier fluid and the wall or of condensation of the carrier fluid on the wall, gravitational settling or impaction on a wall if the carrier fluid changes direction, Brownian or turbulent diffusion, etc.

Analytical test programs have been used to check these models, particularly the TUBA and TRANSAT programs run by IPSN in the 1990s to study thermophoresis and diffusiophoresis, and the DEVAP[170] program run at the same time by CEA on IPSN's behalf at the Grenoble research center, consisting of measuring deposits by condensation of cesium iodide and hydroxide injected into a tube with a controlled thermal gradient. The temperature profile of the wall of this kind of tube decreases between the inlet and the outlet, which is representative of the thermal conditions in the reactor coolant system (RCS) of a PWR in accidental conditions. The FALCON program in the United Kingdom in the early 1990s also studied experimentally the deposition of simulated fission products (I, Cs) in controlled thermal-gradient tubes and the interaction between vapors and aerosols.

The aerosols deposited can go back into suspension if there is a sudden increase in flow rate of the fluid in the RCS, e.g. following an attempt at core reflooding. The phenomena involved were studied experimentally during the 1990s in the STORM facility at the Ispra Joint Research Centre in Italy.

Tests were also performed in the ARTIST[171] facility at PSI, Switzerland, as part of an international program, to study the retention of aerosols in the secondary system of a steam generator. The results obtained made it possible to evaluate more precisely the releases in the event of steam generator tubes rupture during reactor operation.

Given the uncertainties concerning the behavior of iodine in the reactor coolant system, confirmed during the Phebus-FP tests (see further on), IRSN built an experimental device named CHIP at its laboratories in Cadarache, to study the behavior of different chemical species in a reactor coolant system. The device consists of:

– several generators of chemical species in vapor or gas form: mixture of hydrogen and steam (H_2/H_20), simulators of highly volatile, reactive fission products such as diiodine (I_2), cesium (Cs), molybdenum (Mo), reagents present in large quantities during core melt accidents, such as boron (B), silver (Ag), indium (In), cadmium (Cd) from the reactor's control rods melting;

– a heating system to bring the various reagents to a temperature between 1500 °C and 1600 °C;

169. Movement of the aerosols caused by diffusion in a gaseous environment of water molecules to a cold wall on which they condense.

170. Volatile Fission Product Deposits in the Vapour Phase on Reactor System Surfaces.

171. Aerosol Trapping in a Steam-Generator.

– a controlled thermal gradient circuit made from the same metal as the reactor coolant system, into which the reagents are injected; this circuit includes high-temperature (500 °C to 900 °C) sampling devices, and devices for sampling at lower temperatures (approximately 150 °C) for separating the aerosol and gaseous species of iodine at both temperature levels;

– a filter for recovering at the circuit outlet the aerosols formed during the reagent cooling process so that they can be analyzed.

Tests were conducted by IRSN as part of the International Source Term Program (ISTP, 2005–2015) in order to quantify the fractions of volatile iodine produced in the reactor coolant system for reaction systems representative of the conditions of a core melt accident in a water reactor. An analysis of the chemical species that make up the aerosols collected in the filter also provides more information about the chemical species of iodine that are transported in a condensed form. Analysis of these results provides a better understanding of the chemical reactions involved in the production of volatile iodine and their kinetics, and also improves the iodine transport models in the ASTEC simulation code, so that potential radioactive releases in the event of a core melt accident can also be characterized.

9.4.3. Radioactive product behavior in the containment

The aerosols emitted within the containment are also subject to the phenomena of agglomeration, deposition and possibly re-suspension. The basic physical phenomena are the same as those at work in the reactor coolant system. Some components of the aerosols may be hygroscopic and absorb the steam in the containment, which increases their mass and accelerates their deposition by settling. The main deposition phenomena of aerosols in the containment are settling and diffusiophoresis. Spraying, which could be used to cool the atmosphere in the containment and reduce the pressure there, can also accelerate the deposition of aerosols.

The phenomenon of aerosol spraydown in particular has been studied at IRSN, during an analytical test program at its CARAIDAS facility in Saclay. This program was used to study and model the condensation of steam on water droplets and its evaporation, aerosol collection by the droplets when they fall and the absorption of gaseous iodine by the droplets. Experiments on the effects of spraying have also been conducted on a larger scale at the TOSQAN facility already mentioned. The results were used to develop and validate an aerosol deposition model integrated into the ASTEC simulation code.

A large amount of the radioactive products released by the core ultimately ends up in the sump water following their spraydown, leaching of the containment walls by the steam condensates or their deposition by settling. Consequently, the sumps are intensely radioactive. Moreover, with the notable exception of silver iodide (AgI), the majority of the metal iodides (CsI, RbI, CdI_2, InI) are water soluble and form I^- ions. Due to the action of the radiolysis products of the water and a large number of radiolytic and thermally activated chemical reactions, the I^- iodide ions can be oxidized to become volatile I_2 iodine, which is then released into the containment atmosphere. The formation of volatile iodine depends on many parameters, the most important being water pH. At a basic pH, the formation rate is very low, making this a relatively simple countermeasure to implement in the event of an accident (addition of soda). The water in the sumps also contains organic materials originating mainly from the immersed paintwork. The iodine reactions with the organic radicals can produce volatile organic iodides such as methyl iodide CH_3I, which are also released into the containment atmosphere.

These different chemical reactions have been studied in France at IRSN's laboratories during analytical tests on specimens using a γ irradiator (IODE program in the late 1980s–early 1990s).

Because the iodine emanations from the sump also mean that mass is transferred from the water to the containment atmosphere, more global tests were then performed in the 1990s at CEA's CAIMAN facility on behalf of IPSN in partnership with EDF. This consisted of a 300 liters containment with a 25-liters sump in the field of a cobalt-60 source delivering a γ dose rate of 10 kGy/h. Plates coated with the same type of paint as used in French PWRs could be placed in the sump water and in the containment atmosphere. The water in the sump could be brought to 90 °C and the atmosphere to 130 °C. The iodine placed in the water in soluble form was mixed with a tracer (iodine-131). A system of selective filters, known as Maypack[172], was used to identify the different iodine species produced (I_2 and CH_3I) and their physical forms (aerosols or gases) and to measure the quantities using a gamma-ray spectrometer. The SISYPHE program, conducted at IPSN in support of the Phebus-FP program in a similar 10 m^3 containment, also contributed significantly to the development of models for iodine compound mass transfers in sump evaporation conditions.

The different programs provided a better understanding of the phenomena governing the production of volatile iodine species in sumps and enabled the modeling of those phenomena to be improved. They also identified the role of paints, the iodine absorption rate of which increases significantly with temperature. However, the reaction is not irreversible and some of the absorbed iodine revolatilizes in an organic form. With this research, data were gathered on the phenomenon and enabled initial correlations to be established for use in the simulation codes used to evaluate potential radioactive releases in the event of a core melt accident.

In Canada, a practically identical facility to the CAIMAN facility, the Radioiodine Test Facility (RTF) was used by the AECL at the Chalk River laboratories to run many experimental programs on volatile iodine production in sumps. In particular we will mention the Phebus-RTF tests conducted as part of the international Phebus-FP program. They were used to quantify the effects associated with the presence of insoluble silver in

172. Measuring device invented by F.G. May in the 1960s, in which a sampled gas circulates in succession through a membrane trapping aerosols then two filters: the first consisting of a knit mesh, which is a silver-coated braid that traps molecular iodine, the second consisting of active charcoal (impregnated with potassium), which absorbs the other forms of gaseous iodine, including the organic iodine. Measurement of the concentrations deposited at the different stages gives the desired result. Improvements were subsequently made by various IRSN researchers, particularly by replacing the active charcoal with zeolite and measuring the deposits in real time by γ scanning when the iodine is radioactive.

sump water, observed during the first Phebus-FP tests. When silver is significantly in excess that reduces the concentration of I^- ions and thus volatile iodine production in the sumps, even when the pH is acid.

Analytical tests were performed in the context of two international programs run by CNL[173] (formerly AECL) under the guidance of the OECD/NEA, in which France took part: the Behavior of Iodine Project (BIP, 2007–2011) and the Behavior of Iodine Project – Phase 2 (BIP, 2011–2014). Their objective was to study organic iodine production in more detail and establish more accurate models than those obtained to date, so that results obtained in the laboratory could be extrapolated with greater confidence to the scale of a reactor.

Given the uncertainties over iodine behavior, IRSN built a new facility, EPICUR, at its laboratories in Cadarache to study in more detail the physico-chemical processes of volatile iodine production; the tests began in 2005. The facility consists of a 5-liters tank that can be brought to a temperature of 120 °C and placed in the field of six cobalt-60 sources emitting an average γ radiation dose rate of 2.5 kGy/h. The iodine is traced by the iodine-131 and a Maypack scanned in real time by a gamma-ray spectrometer allows accurately measuring the production kinetics of the volatile iodine and whether it is molecular or organic iodine. A series of 30 tests was conducted as part of the ISTP international program (2005–2013), allowing the formation of organic iodine from painted surfaces, whether immersed or not, to be studied; the tests revealed the importance of organic iodine production from painted surfaces when they are in the tank atmosphere.

The tests were also used to study the formation and stability of the iodine oxides (IO_x) formed under radiation in the containment atmosphere. Indeed, the iodine in the containment atmosphere in its molecular and organic forms can react with the radiolysis products of the air, ozone and nitrogen oxides, to form iodine oxides and nitroxides, which are not volatile. These aerosols are likely to be deposited in the containment or in filters if the containment is vented. Preliminary tests at a laboratory in Germany (PARIS program), to help with understanding the results from the Phebus-FP program, had already showed the importance of these reactions, even for low molecular iodine concentrations. The IOx formation and decomposition mechanisms are still being researched as part of the OECD/NEA STEM[174] and STEM-2 programs being run by IRSN in order to develop the associated models.

The EPICUR facility is also being used, particularly as part of the OECD/NEA STEM program (2011–2015), to study the stability of iodine aerosols subject to radiation at different humidity levels, the effect of aging of paints on their ability to adsorb iodine and release volatile organic iodides, and lastly the mechanisms for long-term revolatilization of iodine deposited on the walls or in filtration media. The analysis of the results should reduce the uncertainties about predicting potential iodine releases in the event of a reactor core melt accident.

173. Canadian Nuclear Laboratories.
174. Source Term Evaluation and Mitigation.

In the early 2000s, tests conducted by IPSN at the laboratory in Saclay (RECI[175] tests) showed that heating the air loaded with aerosols passing through hydrogen recombiners can lead to decomposition of the metal iodides (typically CsI, AgI, InI and CdI_2) deposited on these aerosols by forming volatile iodine compounds such as I_2, HOI and HI. Tests on a larger scale conducted under more realistic conditions at the THAI facility mentioned earlier (THAI program) confirmed this phenomenon and enabled decomposition rates of around a few percent to be measured.

In parallel with all these analytical research programs, programs were run using large-scale research facilities. As we have seen, chemistry plays a key role in the behavior of the fission products emitted during core melt and evaluating the resulting releases. For this reason it was considered essential to check, by means of "integral" experiments, i.e. experiments that are as close as possible to reality, that the phenomena and their potential interactions are properly accounted for in the models used in computer codes for calculating releases. In the 1980s and 1990s, the Americans and Canadians had already conducted experiments in a reactor using irradiated fuel (PBF tests, LOFT-FP tests and FLHT tests mentioned earlier), in which appropriate instrumentation was used to identify and quantify the fission products emitted by the fuel. However, none of the experiments modeled the reactor containment. Moreover, the information obtained about the chemical forms in which the main fission products (Cs, I) were emitted where very imprecise.

9.4.4. Contribution of the Phebus-FP program on the various processes involved in radioactive releases

To fill these gaps, IPSN (then IRSN) ran the Phebus-FP program (1988–2012) at the PHEBUS experimental facility, already described in detail in Section 9.1.

The program's main findings concerning fission products and/or the validation of the associated models are as follows [7]:

- the models for the emission of fission products and actinides when fuel degrades, deduced from the VERCORS tests, are confirmed except for barium which is released in small quantities, probably as a result of the formation with zirconium of a less volatile chemical compound;
- the models for the transporting and deposition of aerosols allows their observed behavior to be reproduced overall, both in the reactor coolant system and in the containment;
- cesium is transported in the form of cesium molybdate ($CsMoO_3$), a less volatile species than the cesium hydroxide (CsOH) commonly assumed by experts to be the form and used until then in all the tests of separate effects to study the behavior of cesium;

175. Recombiner and Iodine.

- a fraction of the iodine released into the containment at a break in the reactor coolant system is gaseous, whereas the equilibrium thermodynamics calculations predict that all the iodine emitted by the fuel should at 150 °C be in a condensed form, cesium iodide (CsI); the reactor coolant system therefore constitutes a further source of gaseous iodine in the containment, which needs to be taken into account when calculating releases;
- this fraction of a few percent in tests performed with a silver, indium and cadmium (AIC) alloy as the control rod material, reaches nearly 100% in tests where the AIC is replaced by boron carbide;
- even when the sump water is acidic, there is very little iodine gas emanation from the sump in tests run with AIC since silver, as studied subsequently in more analytical tests, fixes the I^- ions.
- after rapidly decreasing, the volatile iodine concentrations in the containment stabilize after a few days at a value that is approximately identical regardless of the test. This reflects an equilibrium between the various chemical reactions that produce and destroy volatile forms of iodine in the containment: reactions in the containment atmosphere with the products of radiolysis of the air (ozone, nitrogen oxides), transfers between the atmosphere and paints on the walls, transfers between the atmosphere and the sump water, etc.

Iodine's unexpected behavior observed during the Phebus-FP tests (reactor coolant system as the main source of gaseous iodine in the short-term, concentration not changing in the containment in the long-term) led to a reorientation of the research programs studying the elementary phenomena (ISTP, STEM, BIP, BIP2, THAI, THAI-2), which until then had focused on studying radiochemical and chemical reactions in the aqueous phase (IODE, CAIMAN, RTF, etc.), because at that time the sump water was considered to be the only source of volatile iodine production. Very analytical and theoretical research was also begun as part of the process of setting up a laboratory "without walls", C3R[176], involving IRSN, the Laboratory for the Physical Chemistry of Combustion Processes and the Atmosphere (PC2A) in Lille and CNRS.

9.4.5. Aspects to be explored

A) Iodine-related and ruthenium-related aspects

In some accident configurations (after the vessel has failed, during an accident in a reactor that is shut down with the vessel open for reloading fuel, or during a spent fuel pool uncovery accident), fuel could degrade in the presence of a steam and air mixture. The Canadian AECL-CRL tests already mentioned revealed that substantial amounts of ruthenium were released under these conditions whereas, with a steam and hydrogen mixture, very little ruthenium was released as we saw earlier. Ruthenium is highly radiotoxic, and particularly its two isotopes ruthenium-103 and ruthenium-106, which have half lives of 39 days and 369 days respectively. The VERDON-2 test conducted in

176. Laboratory of Chemical Kinetics, Combustion and Reactivity.

2012 at CEA's VERDON facility as part of the ISTP program, on a MOX fuel irradiated at approximately 60 GWd/t, confirmed this phenomenon and enabled it to be better quantified. When it is transported in the circuits, ruthenium changes into less volatile species. However, as shown by analytical experiments performed in Hungary and Finland in the early 2000s, ruthenium tetroxide (RuO_4) in gaseous form, despite being unstable at room temperature, could continue to exist in small quantities and could accumulate in the containment. Tests conducted at the EPICUR facility as part of the ISTP program studied the behavior under radiation of this oxide in the containment and enabled it to be modeled.

Studies of the complex behavior of iodine and ruthenium in the circuits was pursued until 2015 as part of the OECD/NEA STEM program and continue beyond that date with the MIRE[177] and STEM-2 programs. If necessary, other tests could be performed beyond that at the CHIP facility to study and model in more detail the effect of the control rod materials on the chemical forms in which iodine is transported in the reactor coolant system.

B) Filtration of atmospheric releases

The course of the Fukushima Daiichi accident attracted attention to the value of fitting containment depressurization systems with very efficient filters, in particular that will capture all the volatile forms of iodine (diiodine [I_2], organic iodides such as methyl iodide [CH_3I]) even under core melt accident conditions[178] and for periods of several days. Research carried out recently at the PSI in Switzerland tended to show that the effectiveness of trapping all the volatile forms of iodine by aqueous solutions could be considerably increased by using an alkaline thiosulfate and quaternary amine combination. Moreover, in response to the call for RSNR projects launched by ANR in 2012 in the wake of the Fukushima Daiichi accident, IRSN is coordinating the MIRE research program (2013–2017) in partnership with EDF, AREVA and various universities (Lille, Marseille, Nancy, Nantes) on improving the retention of all forms of iodine in filtration media[179]. Lastly, the Passive and Active Systems on Severe Accident source term Mitigation (PASSAM, 2013–2016) research and development project coordinated by IRSN and supported by the European Commission brings together eight partners including EDF to explore possible improvements to the filtration systems in service and to study innovative devices that would be much more efficient.

C) A "water-borne release" counter-measure to be explored

IRSN is conducting studies of a geotechnical barrier as a "waterborne-release counter-measure" in a core melt accident situation. If the event of a core melt accident in a pressurized water reactor that does not have a core catcher, the molten materials can enter into direct contact with the reactor building's concrete basemat. As seen earlier, interaction

177. Mitigation of Releases into the Environment in the Event of a Nuclear Accident.

178. For example, where the electrical systems for heating the filters for removing the steam saturating the iodine trapping sites on certain materials such as active charcoal, are unavailable.

179. The containment depressurization systems of French nuclear power plant reactors are fitted with sand filters.

between corium and concrete is a subject of research aimed at establishing the precise conditions of erosion and the time until basemat melt-through. There is a risk – which has not been completely eliminated though it has been greatly reduced since EDF thickened the basemats of the two Fessenheim reactors seen as critical – that the molten materials and sump water will go into the soil, the ground water, then the nearest river or the sea.

In 2009, IRSN began a study of the feasibility and effectiveness of a vertical geotechnical barrier in the ground under the foundations of nuclear power plants, extending downwards until it reaches a relatively impermeable geological formation, to act as a retention system to prevent ground water from flowing to the site boundaries. A geotechnical barrier like this would offer a grace period of several months following a core melt accident, and also a practical way of limiting pollution outside the site, through the pumping out, storage and treatment of the contaminated water inside this enclosure.

Creating a geotechnical barrier of this kind would not be easy but would generally be possible with existing reactors. However, it would mean that there would have to be a layer of sufficiently watertight materials (clay, compact chalk, rock with few fractures, etc.) at an accessible depth in the ground in order to create a closed volume.

The investigations carried out by IRSN suggest that the feasibility of geotechnical barriers – using existing geotechnical barriers formed for the purpose of installation construction – has been demonstrated for all French nuclear power plant sites except Fessenheim, for which further studies are required, and hydrogeological studies have led to the conclusion that these systems are likely to be effective at most sites, though the Belleville and Cruas sites would still require specific study.

This matter is the subject of discussions with EDF in the context of ASN's formal requests. It could ultimately lead to research and development. For its part, IRSN plans to look into the design (from a safety and radiation protection point of view) of a system for pumping out the water collected in a geotechnical barrier during a core melt accident.

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Chapter 10
Research on the Behavior of Components Important to the Safety of NPPs and More Particularly the Aging of Such Components

This chapter provides an overview of the most significant R&D efforts on metallic components, civil engineering structures, and other components that play a role in ensuring the safe operation of pressurized water reactors in France's NPPs and which, in particular, play a confinement role ("barrier").

The R&D efforts in which IRSN is particularly involved are related to:

– the behavior of components during accident conditions (earthquake, core melt, etc.);

– aging in a broad sense, i.e. the effects of various mechanisms of damage – or pathologies – that may adversely affect components (metallic structures, civil engineering structures, etc.) over time and subsequent to their use (under normal operating conditions).

A gradual damaging of components can be generated by operating loads and their surrounding environment, such as pressure, temperature, thermal transients, vibrations, irradiation, and the chemical composition of the surrounding medium. Operating experience, and the results of periodic tests and inspections in particular, shows that crippling degradations can occur even though mechanisms of damage—at least those that are known—are taken into account in the design, sizing, manufacturing, and operation of components. Cracks on vessel head adapters and tube bundles of steam generators are just some examples of such degradations. In other cases, the kinetics of degradations were faster than expected (thermal fatigue of pipes).

In the case of components studied in terms of loading during accident conditions, the compliance of "end-of-life" components with safety requirements must of course be verified by factoring in the effects of aging during normal operation—or their possible replacement by new ones.

Research on the behavior of metallic components subjected to loading during accident conditions focuses on understanding the behavior of complex structures such as overhead cranes. Regarding civil engineering structures, this research focuses on developing laws and models of thermomechanical behavior for each component element of these structures (concrete, rebar, etc.). These laws and models are then integrated into simulation codes such as Cast3M, developed by the CEA and which is used by IRSN as parts of its research in support of assessments or before key deadlines—such as the ten-yearly safety reviews of France's reactors and the possibility of operating these reactors beyond their 40-year design lives ("operating life" project [DDF]).

Aging management is based on two key principles—**proactiveness** and **monitoring**. It relates particularly to the components involved in the second and third confinement "barriers"—the reactor-coolant pressure boundary and the containment itself. Indeed, it is vital that these components retain their intended design characteristics (compliance) throughout their service lives and up to and including their disassembly during facility dismantling operations. It is worth reminding that a fundamental objective of metallic components that play a containment barrier role is to maintain a sufficient degree of ductility throughout their service lives. This requirement particularly relates to the PWR vessels, which gradually weaken under the effects of irradiation (increase in the ductile-to-brittle transition temperature).

The purpose of R&D on aging is to improve knowledge on the mechanisms of damage, or pathologies, that can adversely affect components (Figure 10.1). In France, this research took on a new dimension after EDF announced its intention to continue operating its NPPs well beyond their 40-year design lives although some of their components are either irreplaceable or difficult to replace (vessel, containment).

The R&D efforts discussed below—specifically those led by IRSN—thus relate to:

- the metallic components of the reactor coolant system,
- the concrete containments,
- the polymers used as insulation around electrical cables, the inner linings of some containments, and seals.

Generally speaking, IRSN's research is intended to highlight phenomena not taken into account by licensees and improve its understanding of phenomena that are not well documented but nevertheless are important to safety. Therefore, and along with R&D efforts led by other bodies (either alone or jointly), it contributes to the development of standard practices in terms of the design, fabrication, and in-service monitoring of

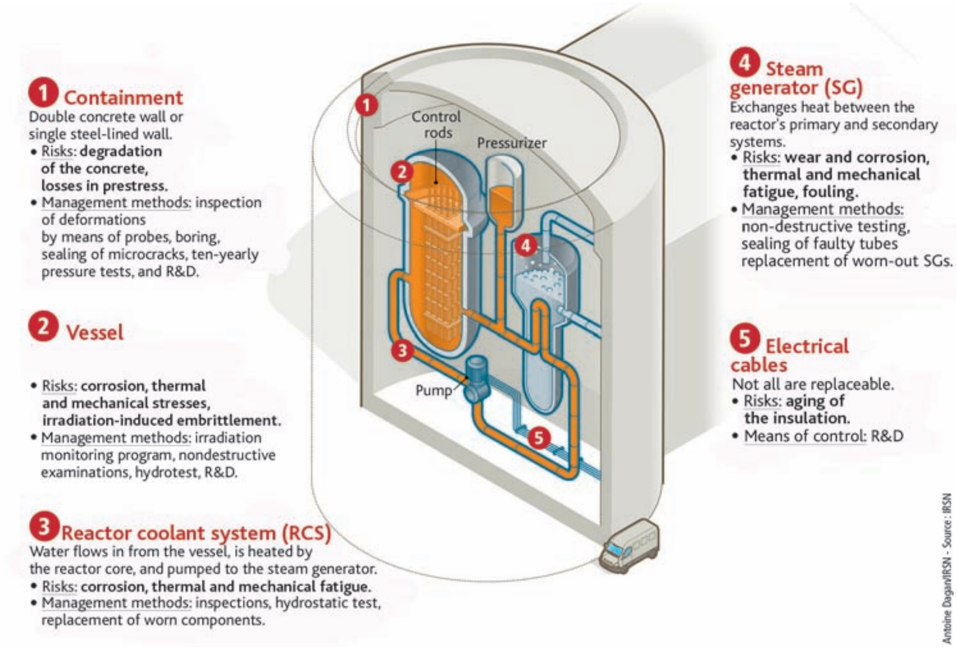

Figure 10.1 Areas of aging in PWRs. Areas 1 and 2 are highly critical, for they contain non-replaceable equipment.

metallic or civil engineering structures, such as those of the RCC-M[180] which applies to the metallic structures in France's NPPs.

10.1. R&D on metallic components

R&D efforts on metallic components involving the key players in France's nuclear power industry primarily relate to the following phenomena:

– irradiation-induced embrittlement of vessel steel. As stated above, this embrittlement alters the mechanical properties of the steel (and increases the ductile-to-brittle transition temperature in particular) and adversely affects its behavior in the case of thermal shocks. This embrittlement can be assessed using empirical prediction formulas adjusted to data derived from analysis of test specimens from the irradiation monitoring program[181] and supplemented by irradiation programs

180. Rules on the Design and Construction of Metallic Equipment (design and construction rules derived and adapted, in the 1970s, from those issued by the ASME [American Society of Mechanical Engineers]).

181. The irradiation monitoring program (PSI) involves testing, for each reactor vessels in France's fleet, samples representative of the vessel steel placed inside capsules along the circumference of the reactor core. These capsules also contain dosimeters to measure the neutron flux density received by the samples. Due to their locations, the capsules are exposed to a higher neutron flux than that received by the vessel walls. This makes it possible to foresee the behavior of the materials after 10, 20, 30, and 40 or more years of operation.

conducted in research reactors. The European PERFORM[182]-60 (2009–2013) project led by EDF and associating AREVA and CEA made it possible to develop a first generation of tools for simulating the microstructural effects of steel irradiation. Research in this field is continuing;

– thermal embrittlement of cast austenitic-ferritic steels used to fabricate a number of components (such as the elbows of the RCS in some French reactors currently in operation). In the early 1980s, it was found that these cast items were susceptible to thermal aging when the RCS was kept at its service temperature for extended periods. The resulting degradation of their mechanical properties (hardening and gradual embrittlement, loss of toughness) was attributed to unmixing of the Fe-Cr-based solid solution into Fe-rich α and Cr-rich α' domains *via* a spinodal mechanisms and precipitation in the ferrite of a nickel and silicon-rich intermetallic phase. The potential risk considered was that of a change in the failure mode of the components, i.e. from a ductile mode to a brittle one requiring little energy, the suddenness of which was particularly worrisome for pressure vessels. With CEA, IPSN studied and tested model materials to (i) understand the structural changes that take place when they are maintained at high temperatures for extended periods and assess the consequences of such changes, (ii) study the scale effects (tests on large and small specimens), and (iii) identify the influence of the ferrite content and the aging duration. These studies and tests made it possible to identify that although the failure of aged cast items is macroscopically brittle, it is actually caused by a ductile failure mechanism that is less dangerous because it requires an input of energy. Numerical simulation tools have been developed to predict the aging of cast elbows on RCSs throughout their entire service life and assess their behavior in the presence of fabrication defects (casting defects, such as shrinkage cavities) or taken into consideration in safety studies (cracks). EDF is also conducting R&D on this issue to justify whether cast elbows deemed sensitive should be left in place or replaced;

– stress corrosion of stainless steels. Conditions that aggravate this damage mechanism (water chemistry, choice of materials and manufacturing conditions) have been identified. Experimental studies have been conducted on this phenomenon as part of international projects, such as the OECD/NEA Halden reactor project (1995–2008) and the EPRI CIR[183] project (1995–2009). Both of these projects yielded significant findings about the initiation and propagation of cracks;

– irradiation-assisted stress corrosion of stainless steels (affecting the vessel internals), which significantly decreases their ductility and leads to the initiation of cracks. This phenomenon has been studied as part of international projects, such as the European PERFECT project (2004–2008), which made use of a multi-scale approach to predicting damage;

182. Prediction of the Effects of Radiation FOR Pressure Vessel and in-core Materials using multi-scale Modelling – 60 years foreseen plant lifetime.

183. Cooperative Irradiation-assisted stress corrosion cracking Research.

- irradiation-induced swelling of stainless steels, which can adversely affect their operability. This phenomenon is being studied by the GONDOLE program (2006–2016) led by CEA at the OSIRIS reactor in Saclay, France, as part of an international partnership (EDF, AREVA, SUEZ-GDF[184], EPRI);
- stress corrosion of nickel-based alloys used to fabricate components such as steam generator tubes and bimetal bonds such as in the vessel head adapters and vessel bottom head penetrations. The effects associated with the chemical conditions of the water in the primary and secondary systems studied as part of the CIRCE and CIRCE-2 programs implemented by EDF and EPRI. IRSN has undertaken research in this field (thesis) as well;
- thermal fatigue, particularly following the 1998 incident at Civaux. This subject will be discussed hereafter;
- wear of steam generator tubes due to their supports. This phenomenon is the main mechanism of damage affecting tube bundles on the latest generation of steam generators in France's NPPs. IRSN, working with CEA, has set up an experimental program to study this wear and determine changes in the coefficient of friction of these tubes with their surrounding conditions (chemistry, steam, etc.).

The implementation of probabilistic approaches to better take the uncertainties and variabilities of the parameters involved into account, particularly in terms of assessing vessel behavior, is also being considered in relation to the possibility of operating France's NPPs beyond their 40-year design lives. EDF is conducting development efforts in collaboration with CEA on this subject. However, IRSN has reservations about the possibility and relevance of lending such approaches to safety demonstrations.

IRSN, working with CEA, is also leading research on the seismic behavior of metallic components such as overhead cranes and precast floors.

Research is also being conducted in the field of non-destructive testing of metallic components.

Both of these subjects will also be discussed further hereafter.

In 2008, EDF created the Materials Ageing Institute (MAI). Led by EDF, the institute is co-financed by NPP licensees including EPRI (Electric Power Research Institute, which represents all nuclear reactor licensees in the United Sates) and KEPCO (Kansai Electric Power Company, in Japan). The MAI brings together the skills of these organizations to predict aging in NPPs and increase the durability of the materials, components, and structures used in them. Knowledge about this phenomenon will be shared with IRSN.

Some of the aforementioned research subjects are discussed more in detail hereafter.

184. French gas utility.

10.1.1. Research on thermal fatigue

Thermal fatigue is a mechanism of damage that EDF, among others, was forced to deal with when, in May 1998, a 30 m^3/h water leak occurred on an RHRS[185] pipe in reactor 1 at the Civaux NPP (N4 series) while the reactor had been shut down for maintenance. This leak was caused by a crazing-like pattern of cracks—known as elephant skin fracture—emerging at the extrados of a pipe elbow (Figure 10.2).

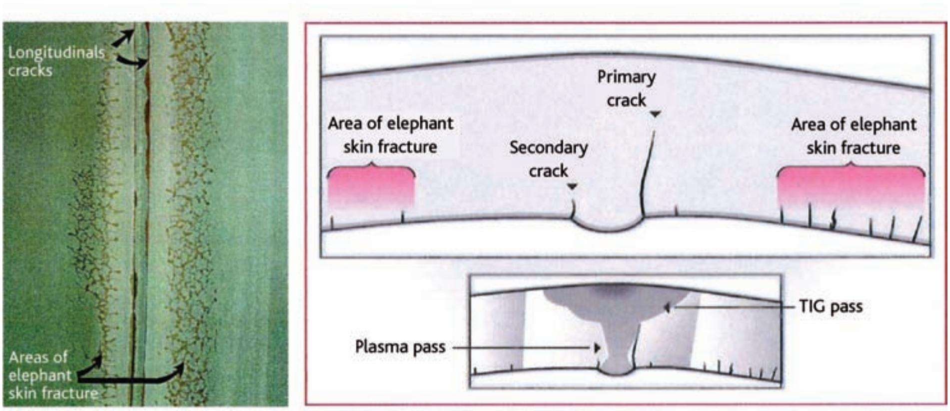

Figure 10.2 The inside of the pipe elbow that caused the leak that occurred at the Civaux NPP in 1998. Crazing-like fractures are visible on either side of the weld bead. © EDF (left) and IRSN–Source: EDF (right).

Thermal fatigue can be summed up as follows: materials expand or contract when subjected to temperature variations. But stresses occur if they cannot expand or contract so freely. Damage from thermal fatigue can result if these variations are too frequent[186].

The crazing-like fracture area of the RHRS pipe (Figure 10.3) of Civaux reactor 1, which was made of austenitic stainless steel, was subjected to significant temperature fluctuations downstream of a mixture of streams (one at 180 °C and the other at 20 °C) at a pressure of 27 bar. The leak occurred after only 1500 hours of operation.

The phenomenon of thermal fatigue in the mixing areas had not been foreseen during the design of the PWRs. The incident that occurred in 1998 could neither be foreseen nor explained by conventional methods and criteria of mechanical fatigue analysis, such as those encoded in the RCC-M and based on the assessment of fatigue "usage factor[187]." It is worth noting that water leaks induced by thermal fatigue had occurred on the RCS before the incident at Civaux. However, they were caused by an altogether different

185. Residual Heat Removal System.
186. Known as high-cycle fatigue, as opposed to oligocyclic fatigue.
187. This usage factor corresponds to the ratio between the number of loads applied to a given component and the maximum number of loads indicated by the mechanical fatigue curve of the component's material.

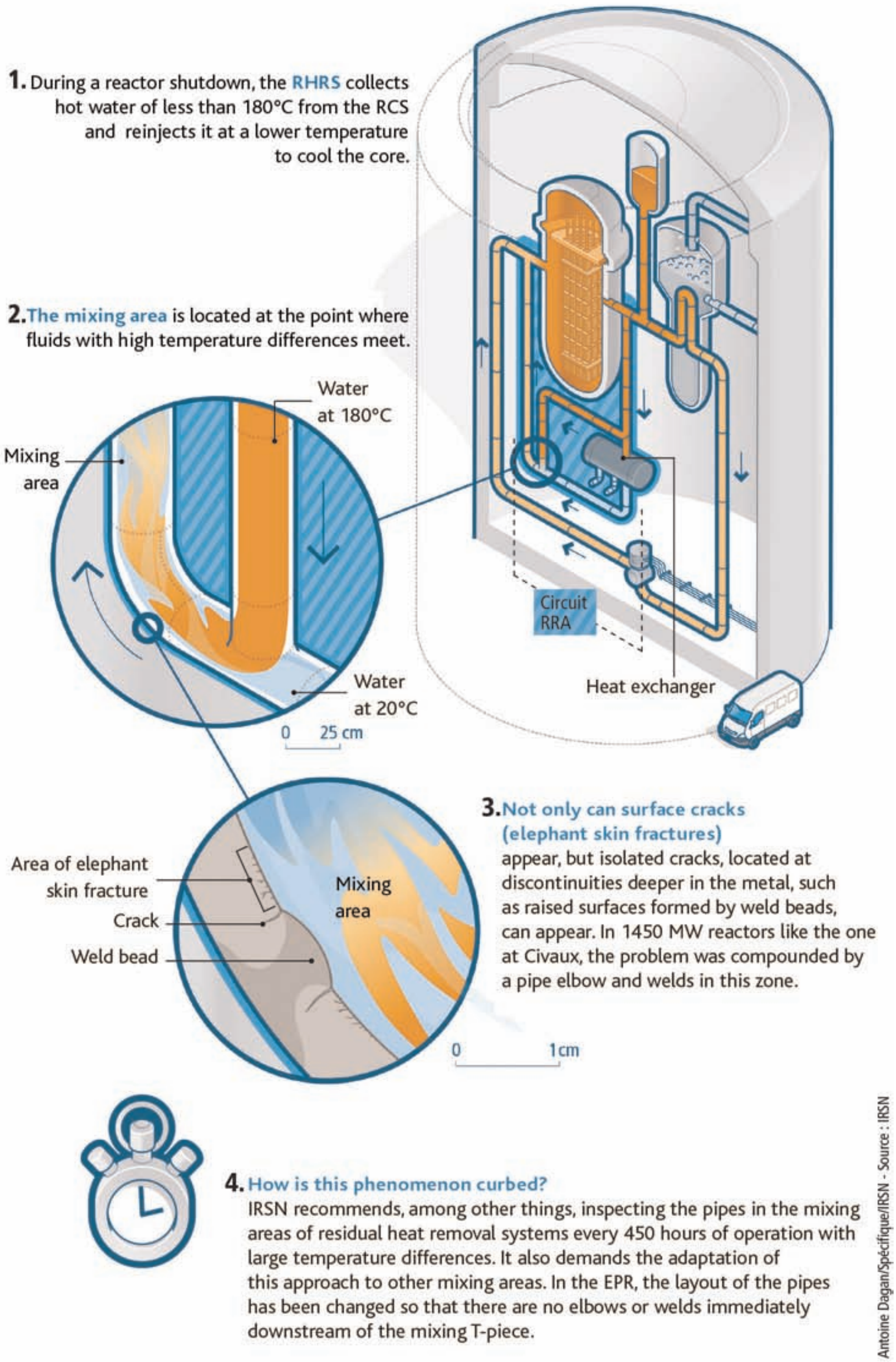

Figure 10.3 Crazing-like fracture on a pipe elbow of the RHRS at Civaux in 1998. © Antoine Dagan/ Spécifique/IRSN–Source: IRSN (above) and IRSN–Source: EDF (below).

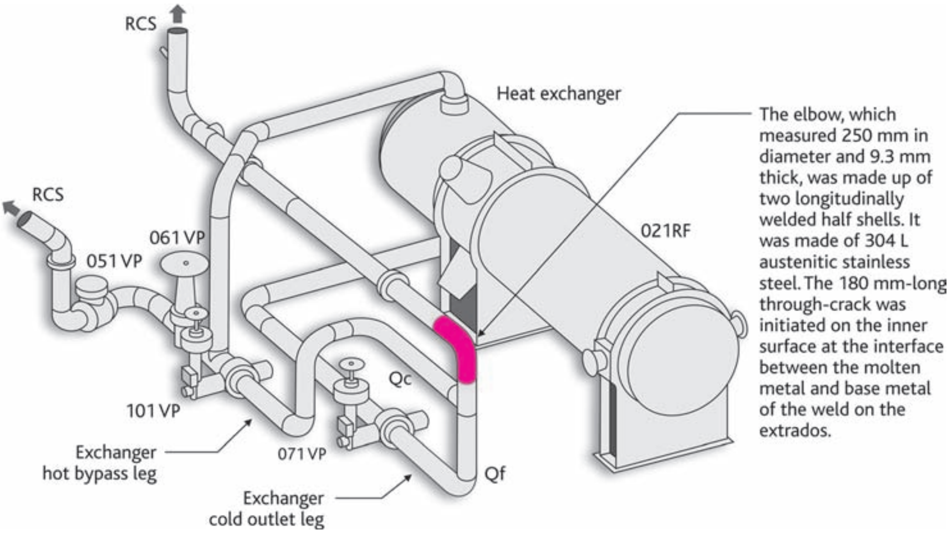

Figure 10.3 (*afterpart*).

phenomenon: dead leg (Farley 2 in 1987, Tihange 1 in 1988, Dampierre 2 in 1992, and Dampierre 1 in 1996).

Starting in 1999, ultrasonic examinations were conducted at every NPP in France. These inspections revealed that the problem was generic (not specific to Civaux) and that all of the examined pipes showed signs of cracking. This prompted EDF to replace the mixing areas of the RHRS circuits in the entire fleet and make improvements to reduce the susceptibility to thermal fatigue (particularly by leveling out the welds).

This finding marked the starting point of more than 10 years of studies and research to understand the cause of the phenomenon and find adequate solutions. These efforts were conducted by EDF and AREVA, and also by IRSN with CEA in particular. Academic laboratories were associated.

EDF's work focused on identifying and assessing the risks of fatigue. Systematic analysis of dismantled pipework, supplemented by mock-up tests (BVS, DUPLEX, FATHER, etc.) and modeling, revealed key factors in the onset of cracks. The primary cause is temperature differences greater than 50 °C between hot fluids and cold fluids and extended periods of repeated pipe loading with this high temperature difference.

IRSN, working with CEA and academic laboratories, implemented a two-pronged research and study program:

– thermal-hydraulic simulations of the flow and mixing of water streams at different temperatures in pipes to better understand the mechanisms at work and key parameters that affect mechanical loads applied to pipes. These simulations were carried out using the Cast3M code in particular;

– mock-up tests (FABIME, SPLASH and FAT3D) to understand the initiation and propagation conditions of thermal fatigue cracks. The FAT3D testing system is illustrated in Figure 10.4 below. Cold water is periodically injected along the inner surface of the tube (test specimen). This water describes a parabola on the inner surface of the tube such as to obtain a temperature gradient in three directions.

A few key teachings were derived from these studies and tests:

– temperature fluctuations at low frequencies (approx. 1 Hz) are responsible for the rapid propagation of cracks through the thickness of components;
– they are caused by turbulence and the specific flow geometry, which are influenced by the configuration of the circuit upstream of the stream mixing area (elbows, straight sections, etc.);
– although thermal fatigue damage is accelerated by welded joints, it can also occur along continuous weld beads (FAT3D test 6). This important teaching invalidated the position previously defended by EDF, prompting it to expand inspections of its facilities to areas not comprising welds;

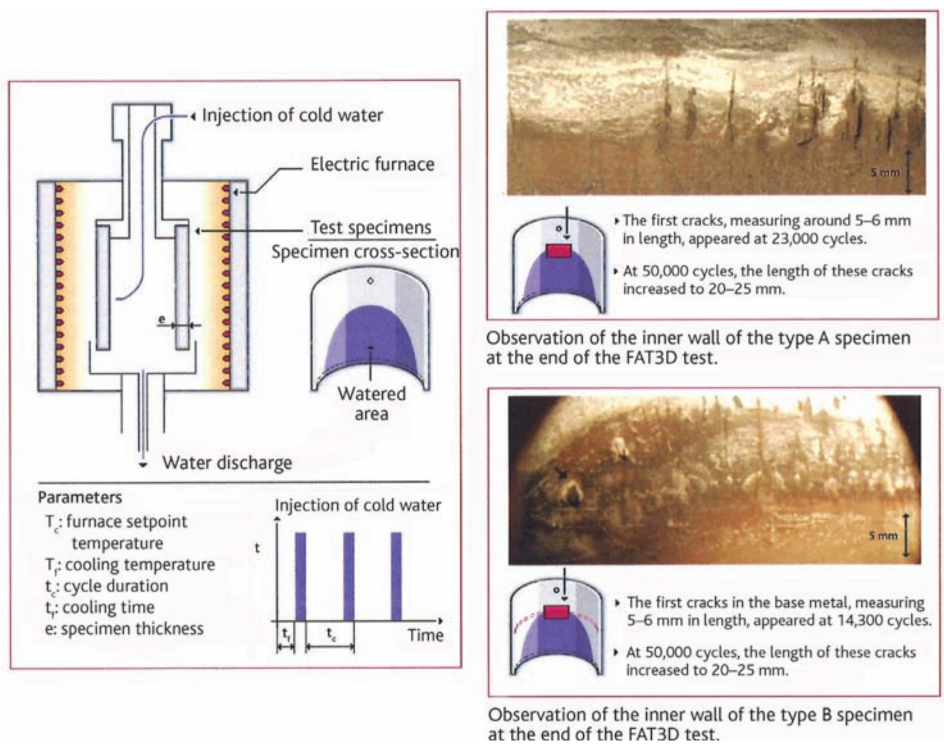

Figure 10.4 The FAT3D test system (left) and the results obtained from tests on specimens (A) without a weld bead (top right) and with (B) a weld bead (bottom right). © IRSN.

- the biaxial nature of thermo-mechanical loads and "environmental" effects may account for the deficiency of conventional methods and criteria used to design and size materials subject to fatigue. These aspects are still being studied (see further on);
- studies conducted on the nozzle area of the chemical and volume control system (CVCS) of the RCS — an area subjected to temperature differences much greater than those occurring in the elbows of the RHRS system (and reaching as high as 280 °C) — led researchers to consider that *a priori* the role of the flow type is more important than that of the temperature difference. Indeed, investigations have shown that the CVCS nozzles in the fleet suffered very little damage from thermal fatigue. This finding has not been invalidated by in-service inspections conducted since.

Based on the results of thermal fatigue studies and research, EDF defined a policy for the operation, in-service monitoring and replacement of mixing areas within all of its reactors. Starting in 2000, the areas of the RHRS circuits subjected to high temperature differences were ultrasonically examined after every 450 hours of operation (following, in this regard, a recommendation of IRSN), and maximum periods of operation at high temperature differences were defined for all susceptible areas.

Nonetheless, research on fatigue is continuing, for much work remains ahead to understand the mechanisms involved and the conditions that cause damage to appear. In 2013, IRSN embarked on the EVA[188] project with INSA in Lyon. The aim of the project is to specifically explore the aforementioned environment effects under the conditions typically encountered in PWRs (pressure, temperature, water chemistry) on the fatigue life of austenitic (and even austeno-ferritic) steels. These effects may explain the lack of conservatism of the in-air fatigue curves of the ASME and transposed in the RCC-M. The testbed will consist of a fatigue loading machine and an autoclave. IRSN has also been a partner of the European INCEFA[189] project, which explores the same subject, since 2014.

10.1.2. R&D on non-destructive testing

Inspections performed during the fabrication of components intended to be used in the construction and operation (in-service inspections) of nuclear facilities are a vital part of defense in depth. For the French fleet of NPPs, these practices are codified in a rule book on in-service inspection for mechanical components (RSE-M[190]). Nonetheless, a number of questions and difficulties arise, such as the adequacy of the type and performance levels of inspection equipment used by manufacturers and licensees (or their contractors) to areas to be inspected that have complex geometries (see Figures 10.5 and 10.6) or materials with specific metallurgical structures. This prompted IPSN, in the early 1990s, to begin R&D efforts to develop sensor prototypes able to adjust to these complex shapes. This research in the field of non-destructive testing (or non-destructive examination [NDE]) deals with ultrasonic waves and eddy currents as well as

188. Study of the Ageing of Steel.
189. INcreasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment.
190. Rules for the Monitoring of Mechanical Equipment in Operation.

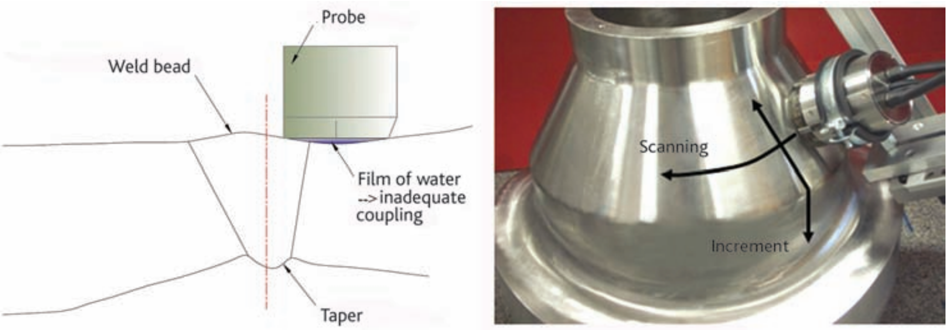

Figure 10.5 Examples of complex geometries: manual ultrasonic examination of a weld bead and ultrasonic examination of an intricately shaped nozzle. © IRSN.

simulations of radiographic examinations. Its aim is to motivate licensees to look for and use the best possible techniques for inspecting the components in their facilities.

Research in this field, conducted primarily by IRSN and CEA, pertains to all nuclear facilities that may be affected by aging. As mentioned above, it deals with the development of innovative prototypes of transducers or probes and examination simulation models. The transducers or probes developed through this research are

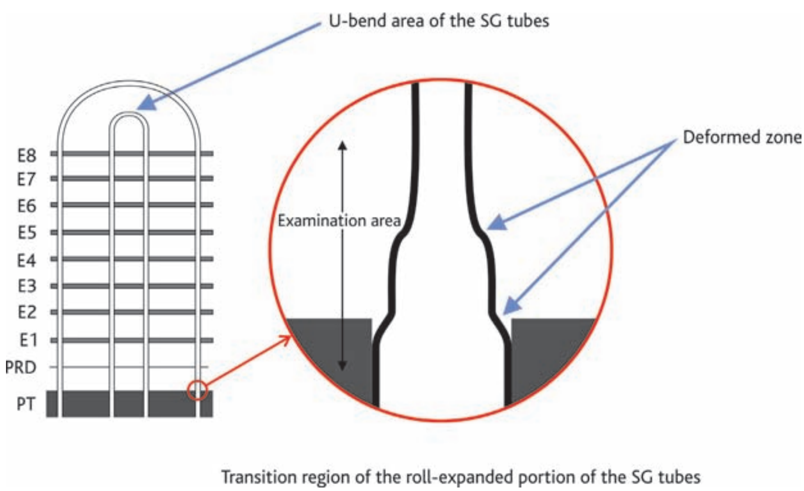

Figure 10.6 Another example of complex geometry: the transition region of the roll-expanded portion of the steam generator tubes and the[191] U-bend areas of the tubes. © IRSN.

191. The purpose of roll expansion of a tube into a plate is to create a radially expansion by mechanical mean or by internal hydraulic overpressure and cause the tube to plastically deform. The residual stresses cause contact pressure between the tube and the plate, resulting in a strong mechanical bond between the tube and the plate. The tools used must not create sharp sloped marks on the metal, tear off any metal, or initiate cracks (Figure 10.6).

intended to be used to check for damage, such as cracks, that may adversely affect safety. These innovative devices, which are usually patented, relate especially to components important to the safe operation of PWRs, some of which are considered failsafe[192] in the safety analysis but for which degradation mechanisms are nevertheless postulated. Significant examination difficulties still remain for some components, such a reactor-coolant pipes made of coarse-grained heterogeneous materials, in which the ultrasonic waves propagate with difficulty, or components with complex surfaces, or the U-bend portions of steam-generator tubes (Figure 10.6). The difficulties encountered with these components should be overcome by the higher performance obtained with these adaptive sensors, which make it easier for ultrasonic waves and signals to penetrate into materials. Signal processing techniques are also being considered for coarse-grained materials.

Simulation models have been developed for the most commonly used inspection techniques, i.e. as ultrasonic examination, radiographic examination, and eddy current examination. These models have been integrated into the CIVA simulation platform developed by CEA and are now accessible to all the relevant users. They are used routinely by IRSN and have become essential to assessing the performance of examinations conducted by manufacturers, licensees or their contractors, in order to substantiate its technical opinions to safety authorities. Most of these developments have been engineered in collaboration with CEA. More specific studies on the examination of coarse-grained materials are being conducted with the U.S.NRC in association with PNNL, and studies on the examination of steam generator tubes are being conducted with ANL.

A) Development of transducers and probes

► Development of a "conformable" ultrasonic transducer for parts with complex geometries

A conformable ultrasonic transducer—i.e. one that conforms to the shapes of parts—was developed by IPSN in collaboration with CEA and patented in 2003.

The main objective of this transducer (Figure 10.7) is to enable or improve detection and geometric characterization of defects in components with complex geometries (small elbows, small nozzles, non-uniform surfaces, etc.) on which acoustic coupling of conventional rigid sensors can no longer be suitably accomplished. It was necessary to improve these examinations, particularly for components most important to safety and for which cracking damage is a possibility. The aforementioned multi-element flexible transducer adjusts to the shape of parts.

192. Components for which design, fabrication, and in-service inspection provisions make it possible to consider that their failure is highly unlikely. Their failure is not addressed by specific consequence-mitigation provisions.

Figure 10.7 Prototype of a conformable transducer – IRSN/CEA patent. © IRSN/CEA.

As seen in Figure 10.8, which shows a study conducted by Laborelec and CEA for the examination of welds on large nozzles, this type of transducer is now industrially applicable.

Figure 10.8 Examination of a nozzle with a conformable transducer (2014 Cofrend Conference – Strategy for robotized examination of nuclear components: from examination design to experimental results). © CEA.

► Development of eddy current sensors for examining steam generator tubes

Examination of steam generator tubes is an important safety concern due to the risk of tube failure and release into the environment of radioactive substances (reactor containment bypass). A steam generator contains anywhere between 3000 and 5000 tubes.

These tubes are examined using eddy current sensors that are generally rigid and well suited to detecting defects in uniform surfaces, such as the straight sections of the tubes. However, conventional rigid sensors can quickly reach their limitations with complex shapes encountered in the transition regions of the roll-expanded portions of steam generators or in small tube bends that may contain deformations (ovality, crushing, etc.). IRSN and CEA jointly developed a flexible sensor to overcome such difficulties and adequately conduct examinations in these areas potentially subject to damage.

Flexible sensor technology (Figures 10.9 and 10.10) is based on the use of winding etched on a Kapton substrate[193] and giant magnetoresistance (GMR) receivers that are highly sensitive to magnetic fields. This technology improves the detection of circumferential defects in areas that are hard to examine due to the presence of longitudinal cracks. CEA and IRSN filed a patent for this sensor in 2009.

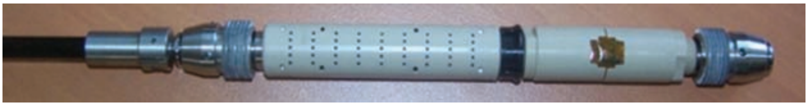

Figure 10.9 Flexible eddy current sensor for examining SG tubes. © DR.

Figure 10.10 Prototype of a flexible eddy current sensor for examining SG tube bends. © DR.

► Development of sensors for castings and heterogeneous structures

Prototype ultrasonic transducers and a prototype of a low-frequency eddy current sensor have also been developed to detect and size planar defects in castings, which have a very coarse grain structure.

Surface-breaking defects along the inner walls of components (pipework, etc.) are generally adequately detected by ultrasonic waves, for the echoes result from an ultrasonic concentration of energy obtained by a "corner effect[194]." However, the size of the planar defects is obtained by measuring the diffraction echoes obtained at the tops of the defects. These echoes actually have very low amplitudes and are

193. Kapton is in insulating polyamide film with mechanical properties (flexibility, thinness, temperature resistance, film etching, etc.) that allow for many uses in electrical and electronic applications.

194. The corner effect is a phenomenon of overintensity of the ultrasonic echo due to multiple reflections on the corner formed at the intersection of the inner surface of the material with planar crack, whether surface-breaking or near-surface, directed generally perpendicular to the surface.

accompanied by a more or less loud noise due to the coarse-grained metallurgical structure that leads to a low signal-to-noise ratio. The progress is significant since extension defects at depths of 10 to 15 mm are sized (characterized) in structures known to be difficult. That said, more progress remains to be made to further improve and confirm performance through tests on a wider range of materials and by improving signal processing and transducer technology.

IRSN and CEA are jointly developing low-frequency eddy current sensors to detect defects located below the surface of thick components. Encouraging results have been obtained with the low-frequency sensor, which also includes a GMR receiver. The prototype detects planar defects that vertically extend between 5 and 15 mm below the surface and produce signals having an amplitude related to the vertical extension of the defect. This eddy current technology should be able to supplement and, in some cases, even replace ultrasonic examinations where measuring the vertical depth of planar defects is not possible, particularly in components made of coarse-grained stainless steels and other materials that strongly disrupt the propagation of ultrasonic waves. Eddy current technology nevertheless has the drawback of requiring access to the inner surface of the piping or component to be examined.

Research to improve the detectability of defects for both ultrasonic examination and eddy current examination will be conducted through a partnership between IRSN and the U.S.NRC.

B) Development of simulation models

► Simulation of ultrasonic examinations of materials with a homogeneous structure

In the late 1990s, IPSN and CEA began developing ultrasonic examination simulation models to provide support tools for the expertises. These models were integrated in the CIVA platform. The studies made it possible to address the case of homogeneous materials in which ultrasonic waves propagate without any particular difficulty for parts of simple or complex shape. These models can be applied to the ultrasonic contact sensors (i.e. in direct contact with components) and contactless immersion sensors[195] on the in-service inspection equipment used to examine welds on the reactor vessels used in France's NPPs. This research was subsequently continued to take into account defects of more complex shape that may be disoriented, in order to be able to adequately simulate defects most similar to those actually encountered in the components. An example of a study is shown in Figure 10.11.

Engineered for homogeneous materials, these models are operational and frequently used in expertises conducted by IRSN for thenuclear safety authorities.

195. Other models for simulating examinations using the time of flight diffraction (TOFD) technique have been developed by CEA and EDF and are available in CIVA.

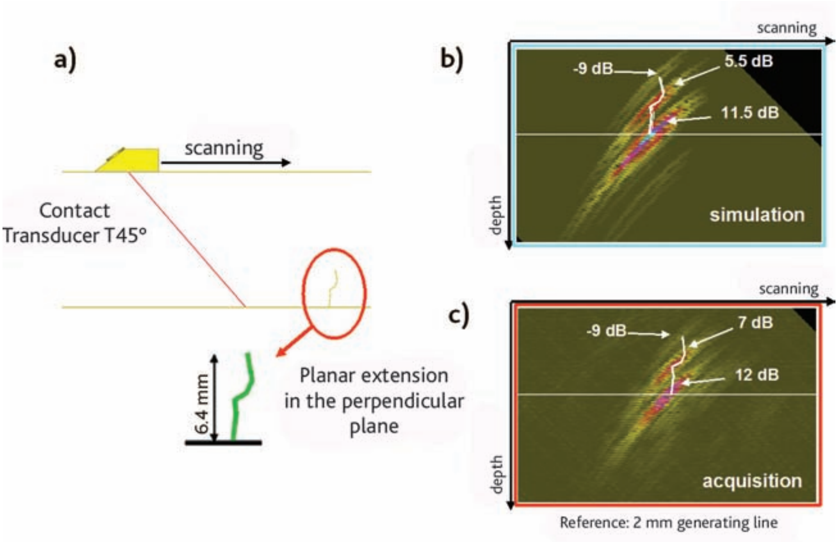

Figure 10.11 Ultrasonic response of a multifaceted defect during contact examination using 45° transverse waves. a) examination configuration; b) "simulated" defect image; c) "experimental" defect image. © DR.

► Simulation of eddy current examinations

CIVA's functionalities for eddy current examinations make it possible to simulate examinations of steam generator tubes with an axial probe (Figure 10.12), rotating probes (STL[196], STT[197], +Point), and multi-element probes. The first phase of development of these models was intentded for a simple configuration with a tube and an isolated defect. Now models make it possible to take into consideration a more realistic geometry of tubes and their environment (geometric transition region between the roll-expanded portion of tubed in tubes plates and their straight portions, areas located under the spacer plates, U-bend areas). It is also possible to model complex defects or a network of defects. Validation of CIVA is accomplished by comparing it with experimental data and by conducting test cases (benchmarks). The resulting confidence in the use of simulation tools makes it possible to use them to assess examination techniques.

Furthermore, CEA has developed CIVA functionalities that simulate the X-Probe, an eddy current multi-element probe most commonly used outside France, particularly the United States. The functionalities will enable IRSN to better understand the performance and limits of this probe, which now numbers among the examination methods usable by EDF.

196. Long Rotating Probe.
197. Transverse Rotating Probe.

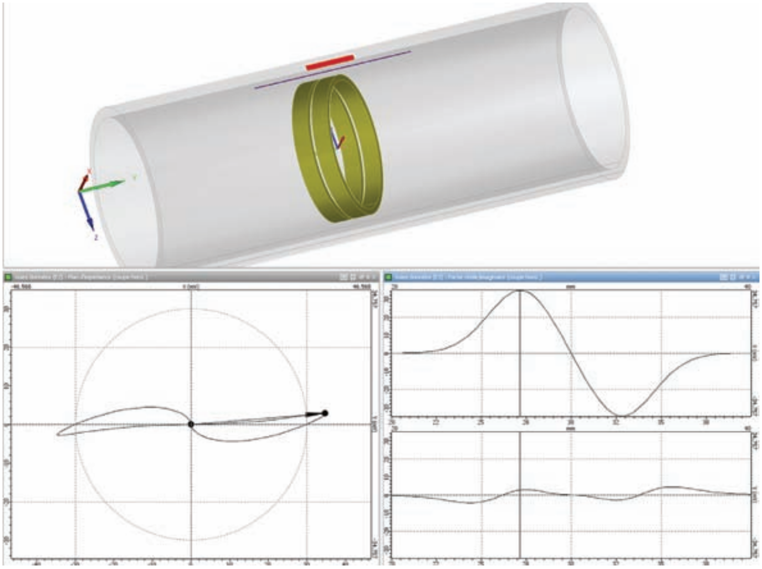

Figure 10.12 Simulation of the response of the axial eddy current probe for an outer notch. © IRSN.

► Simulation of radiographic examinations

The development by IRSN, in collaboration with CEA, of tools for simulating radiographic examinations began much later than research on ultrasonic and eddy current examinations. As a first step, the configurations of the radiographs most commonly used for the components of nuclear facilities—which can have various dimensions, thicknesses, shapes, access conditions—were identified. This made it possible to develop models that reproduced the most common operating conditions and took into account iridium and cobalt radiation sources and silver-image films in use in the facilities. Attention was then turned to the case of welds of dissimilar materials (e.g. stainless steel coatings deposited on ferritic steel, bimetallic welds used to bond stainless steel pipes to ferritic steel components). This was accomplished, for example, by using computer-aided design (CAD) software to describe the part to be radiographed and the various areas of material on the part and to which the properties of each material in question is assigned. A phased approach was also applied to defects for which detection is to be simulated; these defects shifted from very simple shapes at the start of the studies to much more complex shapes that can be described using 3D CAD software.

Starting in 2010, emphasis was especially placed on experimental validation of simulation models using mock-ups representative of components and with calibrated defects. Such validation was conducted on a few of the radiograph configurations studied previously. They made it possible to compare the results obtained through simulation with those obtained during the experimental radiographs. These model validations are being continued to the works to address the most common operating conditions[198] of radiographic examination and are the subject of publications or benchmarks.

Since 2013, research is being continued to take into account such aspects as other radiation sources like selenium (less exposure to operators, but penetrate less through structures), X-ray for large thicknesses, and digital films.

Here, too, the models developed and integrated into CIVA are used regularly at IRSN for expertises and parameter studies.

► Simulation of ultrasonic examinations of coarse-grained materials with a heterogeneous structure and improvement in these examinations

After developing simulation models suited to ultrasonic examinations of components fabricated from homogeneous materials or homogeneous welds—which generally do not prevent the propagation of ultrasonic waves and make it possible to assess with a high degree of confidence the performance of many ultrasonic examinations—IRSN considered that it was necessary to continue its studies in order to develop simulation models more specifically suited to ultrasonic examination of heterogeneous materials.

The heterogeneous materials in the RCS of PWRs—found, for example, in cast portions or welds used to bond stainless steel pipes to large RCS components made of ferritic steel (bimetal bonds)—can significantly hinder the propagation of ultrasonic waves. This is due to the special metallurgical structure of the materials, which contain coarse grains (both castings and bimetal bonds) that vary in direction and size according to the depth. This causes the speed of the waves to change and leads to various dispersion or attenuation phenomena at the interfaces with the grains. These alterations impair the detection or sizing of defects. In either case, simulated and real-life examination of these specific areas of materials remains a major focus of R&D and there are many difficulties to overcome. Simulating these examinations requires being able to describe the complex structure of the material (base metal and weld) in the 2D or 3D model and to calculate the acoustic field transmitted through the materials while not forgetting that complex structures will produce more noise than homogeneous structures. Lastly, it requires being able to calculate the various interactions in the disrupted field during penetration into the various structures with the encountered defects.

Despite the limited number (Figure 10.13) of mock-ups representative of this type of materials, it was possible to develop models that predict the propagation of ultrasonic waves. This was accomplished by macrographic examinations, conducted directly on these mock-ups, and which made it possible describe simplified structures that yield the contours and directions of grains. The results of these models were very promising, but, due to the limited number of cases studied, tests on other mock-ups had to be conducted to confirm and improve the predictions.

198. Panoramic radiograph with the source centered in the piping and the films located outside, radiographs with the source located outside in contact with the piping, etc.

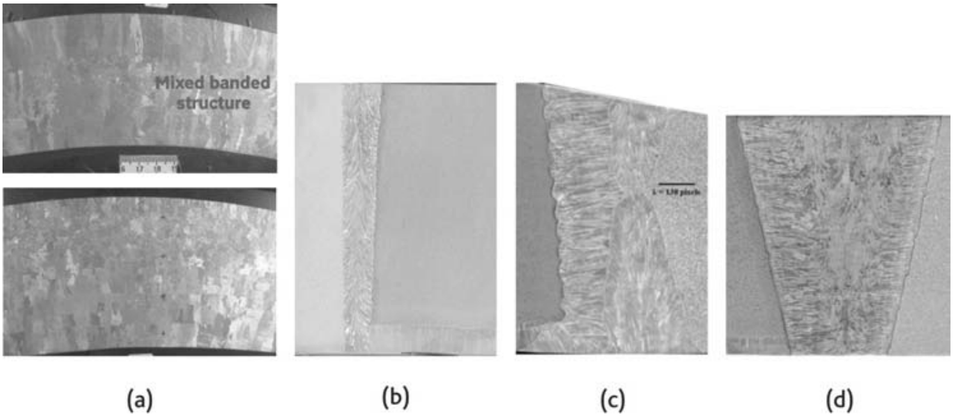

Figure 10.13 Coarse-grained metallurgical structures observed on IRSN mock-ups representative of: (a) a cast component fabricated by Manoir; (b) a narrow groove weld; (c) stainless steel bimetal bond fabricated by Cockerill; (d) "N4" vessel bimetal bond. © IRSN.

Another major difficulty in the simulation of ultrasonic examination of these complex structures is that the description of the structure of the materials must be known in order to make a simplified description of the structure and its grains. This can be achieved via shape recognition (Figure 10.14) from macrographic examinations such as those shown in Figure 10.13 for mock-ups used to develop the models.

Unfortunately, there are no representative samples of such components for most of the heterogeneous structures and/or welds encountered in France's NPPs or in those of other countries. Other methods of description than those using macrographic examination must therefore be investigated.

Studies are thus continuing. Firstly, to explore non-destructive methods that will make it possible to obtain sufficient descriptions of the structures of materials and their

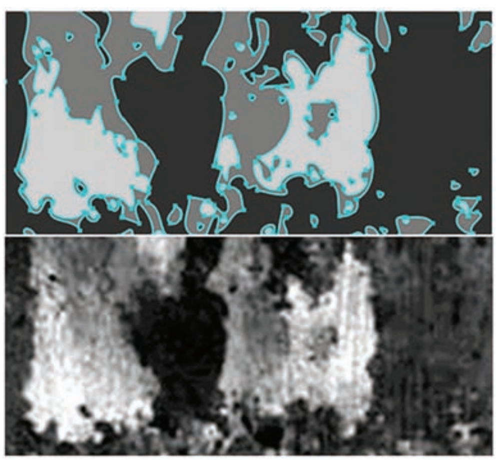

Figure 10.14 Example of grain-shape recognition for cast steel fabricated by Manoir. © IRSN.

grains and correctly predict the performance of examinations, and secondly to assess and even develop codes that simulate solidification from fabrication parameters for cast and welded components for bimetal bonds. It would then be necessary, for example, to be able to simulate examinations on the components in the fleet, to have access to fabrication or welding data that could be described by category. Such access could be made possible by contacting the manufacturers or builders.

Models developed for bimetal bonds and cast coarse-grained materials require further improvement, for the differences between simulated and experimental results remain too significant and depend on the reliability of the methods used for determining the characteristics of the materials. IRSN thus considers that simulations for these components are not sufficiently operational. Research being conducted by IRSN and the U.S.NRC, in association with CEA and PNNL, will make it possible for these bodies to share their data- and mock-ups representative of components. This collaboration is expected to continue until 2017 and will also include research on developing prototypes of transducers and eddy current probes better suited to detecting and sizing defects in these difficult-to-inspect materials.

10.1.3. Studies and research on the seismic behavior of overhead cranes

IRSN, in collaboration with CEA, began conducting studies and research on the dynamic behavior, under seismic loads, of overhead cranes made up of welded elements[199]. The behavior under seismic loading of such structures is highly complex, for phenomena such as multiple impacts (in case of blocking of a wheel) or trolley slippage may occur.

IRSN investigated this issue during assessments of documents relating to seismic reviews of various nuclear facilities, for the demonstrations by the licensees of the earthquake resistance of overhead cranes and absence of load drops involved methods having a robustness that called for confirmation (modal analysis method associated with the use of "reduced spectra" or "reduced weights").

It first appeared necessary to improve the understanding of the seismic behavior of overhead cranes by deriving the maximum possible benefit from the experimental (shake tables) and simulation (finite-element calculation codes) means available.

This research (2005–2013 [1]) consisted of a theoretical part of numerical simulations, and an experimental part with tests performed on the AZALEE shake table at CEA's Saclay research center[200] (Figure 10.15). The tests were conducted on a reduced-scale

199. I.e. assemblies with welded connections, as opposed to assemblies with bolted connections, which have gaps.

200. The Seismic Mechanic Studies Laboratory (EMSI) operated by CEA’s Saclay research center has been conducting earthquake-resistance engineering research for more than 40 years. EMSI’s aim is to use its numerical and experimental tools to further understand the behavior of structures, equipment, and components under seismic excitation. The laboratory’s experimental equipment is housed within the TAMARIS (Tables and Means of Analysing Seismic Risks) facility. AZALEE is currently the largest three-axis shake table in Europe.

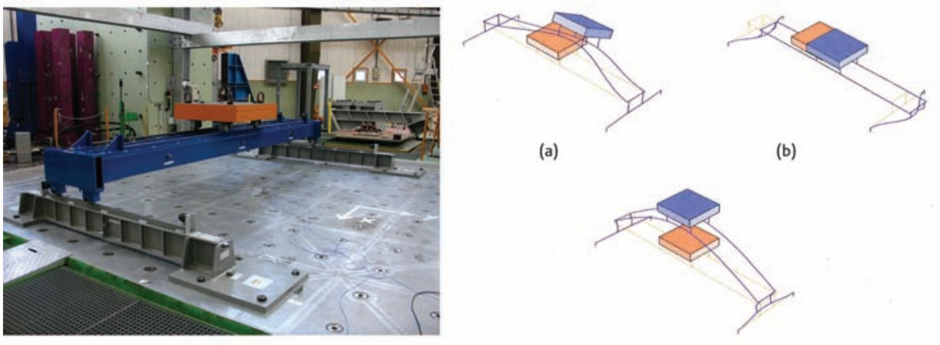

Figure 10.15 Overhead crane tested on the AZALEE table (left) and the three main modes of deformation obtained by numerical simulation. © CEA (left).

mock-up (1/5). Seismic loading was simulated by controlled jacks that moved the structure in both horizontal directions of the table. Firstly, bibliographic researches made it possible to identify the parameters that affect the behavior of overhead cranes. The mock-up's geometry and scale were defined using finite-element calculations and the objective was to find, in the mock-up, the main eigenmodes of the crane's actual structure[201]. More than 100 configurations were achieved by varying a number of influencing parameters, in particular the trolley position, the type of contact at the wheels (blocked wheels or freely rotating wheels), the "added weight" (load suspended from the trolley) and its position. A simplified numerical model able to reproduce the dynamic behavior of this structure with reduced calculation times was implemented. This model was validated for several configurations representative of the operating conditions of the overhead cranes. It may be used by IRSN for seismic risk studies, in particular to understand the behavior of overhead cranes in case of earthquake beyond the design basis earthquake (DBE).

The lessons derived from this research may subsequently enable to IRSN to propose tracks for ultimately establishing rules on analyzing the dynamic interactions between overhead cranes and civil engineering structures.

10.2. R&D on civil engineering structures

R&D work on the civil engineering structures of PWRs relates to primarily to the containment. The containment is essential, for it is the last "barrier" between the reactor and the natural environment.

We would like to remind the reader that three types of containment are used at the PWRs in France's NPPs:

- the first type (900 MWe reactors) consists of a building having a single wall of prestressed reinforced concrete lined on the inside with steel plate, referred to as a static containment;

201. The reference overhead crane is a generic crane used in nuclear facilities.

- the second type (1300 and 1450 MWe reactors) are double-wall buildings comprising an inner wall of prestressed concrete and an outer wall of reinforced concrete. Dynamic containment is ensured by ventilation and filtration of the annulus space between the two walls, thus supplementing the static containment achieved by the inner wall. Composite (resin) liners were applied to the intrados of the inner walls to improve their leaktightness;
- the third type (EPR) is a combination of the previous two: leaktightness is achieved by a metallic liner on the intrados of the inner wall and supplemented by the dynamic containment associated with double-wall containments.

The behavior of these containments under design-basis conditions and during core melt accidents is described in the IRSN document on the state of knowledge on core melt accidents in power reactors[202].

Furthermore, an important aspect regarding pre-EPR reactors in France's NPPs is that the thermo-mechanical loads on the containments during core melt accidents is more severe than the design-basis loads (around 5 bar absolute[203]), since that the "U5" device was subsequently fitted to decompress the containments should the need arise and vent filtered releases to the environment (sand-bed filter).

One of the aims of research on concrete civil engineering structures is to study the behavior of containments during accident conditions and taking aging effects into account. Its purpose is to allow the assessment of the mechanical behavior and leaktightness of the walls of the containments, which can be altered by the combined mechanisms of shrinkage, creep, and "pathologies." Some of these "pathologies," such as alkali-aggregate reactions (AAR) and delayed ettringite formation (DEF), are caused by chemical reactions that occur a few decades following construction. Research is being conducted to analyze these complex phenomena and control them in order to ensure that the safety requirements for these civil engineering structures are met.

Before discussing the R&D efforts being conducted more specifically by IRSN on nuclear civil engineering structures, three major research projects that EDF has led or is currently leading on such structures must be mentioned:

- MAI (mentioned in Section 10.1);
- the nationwide CEOS.fr project on the behavior and assessment of cracking and shrinkage defects in special structures [2008–2014] and for which the French Institute for Applied Research and Experimentation in Civil Engineering (IREX) provided administrative and logistical support. The aims of this project—which was supported by 41 partners, including IRSN—were to study cracking induced by various loads (monotonic or cyclical static loading) in walls and massive components made of reinforced concrete and the early-age behavior—extending from

202. See Chapters 6.2 and 6.3 of "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.
203. This value corresponds to the design pressure that encompasses LOCA, and it is used to conduct containment integrity tests.

10 days following casting—to the maturity under the various aforementioned loads, with special attention given to cracks formed in these structures. Recommendations for controlling these cracking phenomena were compiled in a document published in 2015 [2];

– the VERCORS project (2013–2021), which we will discuss further on in this document.

10.2.1. Development of constitutive equations for civil engineering structures

IRSN is leading research on the behavior of containments under seismic loads or and during core melt accident conditions. It uses the Cast3M numerical simulation code, which was developed by CEA with IRSN's contribution (for example, it provided CEA with a number of rheological models derived from IRSN-funded PhD work). An important point that must be emphasized is that these simulations are checked and validated by comparing them to the containment monitoring activities conducted continuously by EDF and during testing.

Since the 1980s, IRSN has been working with CEA's specialist laboratories that produce the developments necessary to establish constitutive equations for the component materials of the containments (concrete, steel), ranging from the linear behavior of the structures to their non-linear behavior and taking account of concrete cracking and structural damage. These equations are used in research on the behavior of the containments under complex loads. The objective is to be able to simulate the behavior of the containments from their construction to a potential accident.

The development of these equations is based on tests conducted on specimens that have been verified on structures (slabs, beams, gantries) and validated with the results of tests conducted on large-scale mock-ups (a particular example is the tests conducted on the RCCV[204] and PCCV[205] mock-ups at Sandia National Laboratories in the United States)[206].

These R&D efforts make it possible for IRSN to assess, using Cast3M and as part of physics research conducted in support of the development of level-2 probabilistic safety assessments (PSA), the behavior of the containments during core melt accidents. These studies began with the 900 MWe reactor containments in the 1990s. A multi-scale approach developed by IRSN, and based on the computing resources available at the time, was used to determine with a sufficient degree of accuracy, the susceptible areas of these containments (Figure 10.16). These studies showed that the equipment hatch—and more particularly the closure flanges—was a weak spot in the event of a core melt

204. Reinforced Concrete Containment Vessel.
205. Prestressed Concrete Containment Vessel.
206. Test conducted as part of a collaboration between Sandia National Laboratories (SNL), the U.S. NRC, and the Nuclear Power Engineering Center of Japan (NUPEC).

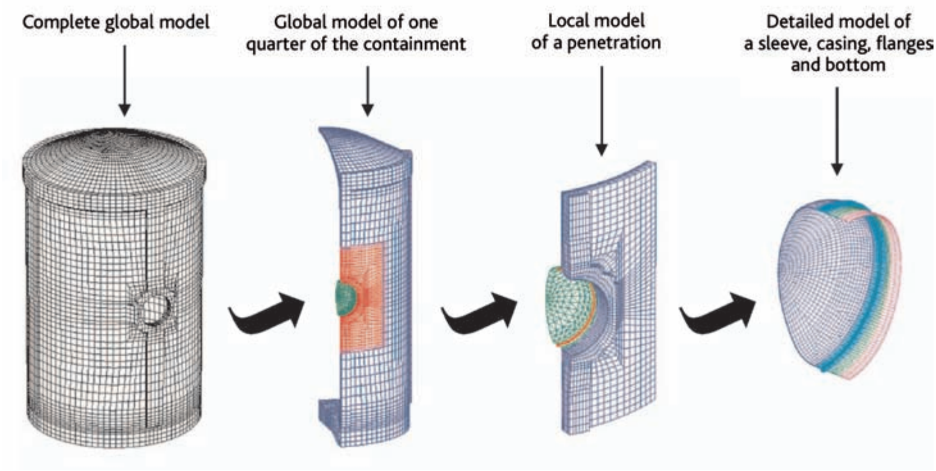

Figure 10.16 Modeling of the containments and the equipment hatch. © Georges Nahas/IRSN.

accident. This prompted EDF to schedule the closure system to be changed during the third ten-yearly outages of these reactors[207]. The system now comprises bolts that are larger in diameter and made of a different grade of steel.

These simulation tools thus offer the possibility of exploring the behavior of containments subjected to other loads during accident conditions (collisions, earthquake, etc.) and even beyond-design-basis combinations of loads during accident conditions.

10.2.2. R&D on the behavior of civil engineering structures under seismic loads

As mentioned above, the development of constitutive equations for the various components of the civil engineering structures and of rheological models makes it possible, with the Cast3M simulation code, to study the non-linear dynamic behavior of structures under seismic loads up to and beyond the design-basis earthquake.

The experimental programs were conducted at the request of IPSN (then IRSN) on simple and complex structures such as gantries and walls using the shake tables at CEA's Saclay research center (dynamic loads). At the same time, other tests on identical structures and under alternate static and increasing loads up to failure were conducted at CEBTP[208]. All these tests made it possible to acquire new knowledge about the dynamic behavior of reinforced concrete structures during earthquakes.

207. See Section 4.4.2.2 of "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.
208. French Construction and Public Works Test Center.

R&D efforts are being conducted in other areas, in particular:

– soil-structure interaction,
– evaluations of motions transferred from floors to equipment installed on different floors,
– the dynamic behavior of reinforced concrete structures.

Another focus of research conducted in collaboration with CEA pertains to seismic isolation systems for the structures, in particular regarding their application to new facilities (Jules Horowitz reactor (JHR), International Thermonuclear Experimental Reactor (ITER), both at Cadarache center).

These areas are discussed in more detail below.

A) Soil-structure interaction

Knowledge of the effects of the soil-structure interaction (SSI) is key to assessing the dynamic response, under seismic loads, of a building such as a PWR, which weighs around 50,000 tonnes. This interaction can be studied using different methods. Two were explored by IRSN—one developed at the École Centrale de Paris (integral equations method), and one developed by researchers at CEA's Saclay research center (absorbing boundary method[209] to limit the size of the field representing the soil). IRSN initiated a collaboration with the École Centrale de Paris to use its method with the MISS3D[210] code. The use of the integral equations method in the finite-element calculations of the structures was achieved by sequentially chaining the MISS3D and Cast3M codes. Both methods were validated during IRSN's participation, in collaboration with CEA, in the international KARISMA[211] benchmark set up by the IAEA following the earthquake (magnitude of 6.8 on the Richter scale) that affected the Kashiwasaki Kariwa NPP and its seven boiling water reactors on July 16, 2007. Preference was given to the method developed by CEA following numerical simulations conducted on one of the Kashiwasaki Kariwa BWRs. However, although the comparison of the calculation results obtained with the on-site measurements confirmed the ability of the model achieved with this method to reproduce the horizontal displacements, differences due to the influence of detachment of the basemats during the earthquake were observed for the vertical displacements. Additional research is being conducted to take into account, in the model, the influence of the non-linearity introduced by detachment.

B) Assessment of displacements transferred to equipment

Advances in simulation capabilities also serve to improve the prediction quality of displacements transferred by the buildings to equipment by taking into account the

209. This method makes it possible to conduct seismic soil-structure interaction calculations via finite-element analysis by limiting the size of the field representing the soil. The resolution is made in the time field for the entire soil-structure meshing and makes it possible to model the detachment and slippage of the basemat (Cast3M code manual).
210. 3D Modelling of Ground-Structure Interaction.
211. KAshiwazaki-Kariwa Research Initiative for Seismic Margin Assessment.

non-linear behavior of the structures (cracking effect of the concrete) in, for example, assessments of existing margins beyond design-basis loads for the structures and equipment.

Seismic margin assessment is a priority that emerged particularly in the United States in the 1980s. The aim was to understand the possible contribution to the overall risk of core melt accident by beyond-design-basis earthquakes that could affect NPPs located in the eastern[212] part of the nation. This resulted in different approaches[213] referred collectively as Seismic Margins Assessment (SMA).

Furthermore, after the earthquake that affected the Kashiwasaki Kariwa NPP on July 16, 2007, *in situ* observations showed that some equipment items had withstood loading beyond their design basis. More recently, an earthquake affected the North Anna NPP in Mineral, Virginia, in August 2011. The ground accelerations at the two reactors were assessed to be 0.2 g and 0.3 g, higher than the design-basis earthquake, which was 0.12 g and 0.18 g, respectively[214]. The cumulative absolute velocity (CAV) threshold, of 0.16 g.s, was exceeded during the quake, causing the reactors to automatically shut down. The generators started up following the resulting electric network failure. The reactors were returned to operation in December 2011 after investigations (including inspections by the U.S.NRC) showed that the earthquake had not caused significant damage to the NPP's safety-important equipment.

Needless to say, the issue of seismic margins returned to the forefront following the Fukushima Daiichi nuclear accident in the case of complementary safety evaluation (ECS) conducted in France, where the important concept of robustness—favoring design and construction solutions that are stablest under multiple loads with margins to cover unexplored fields—emerged.

It should be said that, in the case of Generation III reactors, France's "Technical Guidelines for the Design and Construction of the Next Generation of PWR Reactors" (2004) stipulate that *"the designer must also specify how it intends to prove the existence of sufficient design margins that are consistent with the general safety targets (...). Margin assessments shall be conducted to demonstrate that no cliff-edge effect in terms of radiological consequences could occur by assuming acceleration values greater than those specific to the site. The corresponding method shall take into account the actual behavior of representative equipment and of possible simultaneous equipment failures"*.

The aim of current R&D efforts that IRSN is participating in is to develop simplified numerical models able to simulate the non-linear behavior of structures and determine the seismic forces that act on equipment (transferred spectra). A number of tests

212. The high seismic area of the California coastline has been excluded indeed from the scope of the study.
213. See the SMA method described in U.S.NRC NUREG/CR-4334 (1985) as well as NUREG/CR-4482 and NUREG/CR-5076; the probabilistic PRA-based SMA method, which is also described in an U.S. NRC document; and the EPRI SMA method described in document EPRI NP-6041-SL Revision 1 (1991).
214. These reactors had been seismically reassessed for an earthquake with a PGA (Peak Ground Acceleration) of 0.3 g in the 1990s.

conducted with the AZALEE shake table have thus been used to validate the ability of these models to reproduce the measurement results and observations.

Among these tests are those recently conducted as part of the ENISTAT[215] program [3], which aimed to understand the robustness of structures built according to the Eurocode 8 standard[216] and fitted with thermal insulation systems between their vertical walls and floors (thermal break elements, which create discontinuity between the vertical walls and the floors).

The half-scale, asymmetrical reinforced-concrete mock-up with a total weight of around 40 tonnes was designed for a zero-frequency acceleration at 0.3 g.

The mock-up was subjected to horizontal seismic loading ranging from 0.1 g to 0.8 g. Although a wall failed at 0.8 g, cracks appeared at low loading. The test results made it possible to improve the simplified models to better take into account the displacements observed at the centers of the floors.

An important point emerged during the tests. When three-directional seismic loads (horizontal and vertical simultaneously) were applied, the mock-up's stability was affected by the displacements even at a low acceleration of 0.07 g. The tests were therefore continued beyond 0.07 g without any vertical quake. This observation led to IRSN to plan research on the adverse effects of the vertical component of earthquakes.

The simplified models will be validated in particular by making the most of the results of new tests conducted as part of the SMART program (Figure 10.17). These new tests have been conducted on a 1/4-scale mock-up designed in accordance with ASN Guide 2/01 for the structures in nuclear facilities.

C) Dynamic behavior of reinforced concrete structures

The Fukushima Daiichi accident brought back to the fore the issue of load combinations, in particular that of a core melt accident following an earthquake or of an aftershock following a core melt accident triggered by the main quake. Determining the total effects of both transient loads of different type and duration is complex, more particularly if the reinforced concrete structure enters the non-linear field or even becomes damaged. IRSN is developing and implementing various methods to meet this need. In particular, it is implementing simplified models—such as the lumped-mass stick model (Figure 10.18)—that are able to reproduce the overall behavior of the structures. The local behavior is subsequently determined for the critical times with a finer model of the structures by applying the load sets derived from the simplified analyses. Other innovative techniques are being developed with CEA and ENS Cachan[217].

215. Experimental and Numerical Investigation of Shear wall reinforced concrete buildings under Torsional effects using Advanced Techniques, conducted as part of the European SERIES program (2009–2013), led by Middle East Technical University (METU), in Turkey.
216. "Design and dimensioning of structures for earthquake resistance".
217. A prestigious French higher National School.

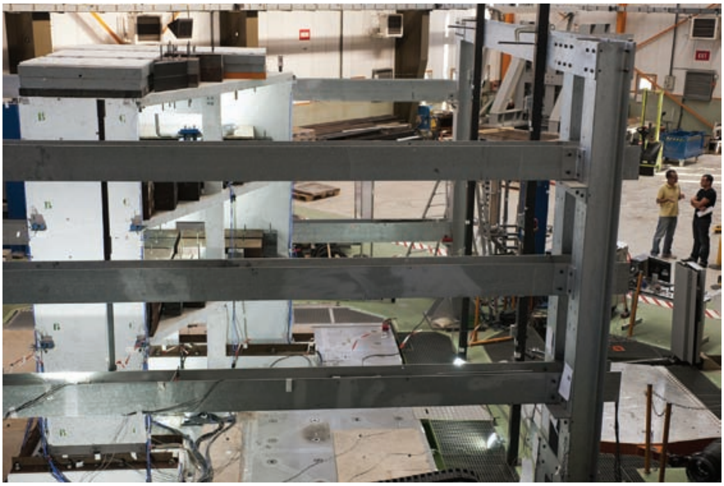

Figure 10.17 The SMART2011 instrumented mock-up (used in support of a joint research program between CEA and EDF) is representative of a 1/4-scale "nuclear" building. Weighing around 47 tonnes (including 36 tonnes of additional weights), SMART2011 was instrumented with more than 200 measurement channels (acceleration, displacement, deformation) and was subjected to a series of seven earthquakes of increasing magnitude. © P. Stroppa/CEA.

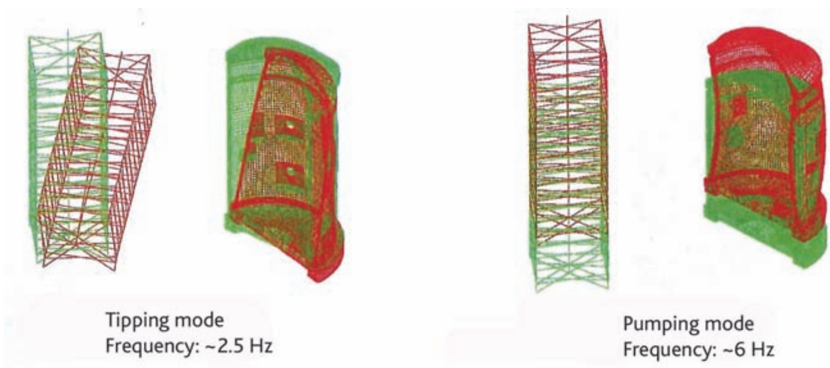

Figure 10.18 Simplified lumped-mass stick model of a containment to simulate its behavior under seismic loads: re-creation of the fundamental modes. @ IRSN.

D) Other Areas of R&D in Seismic Loading

IRSN is exploring other areas:

– seismic isolation systems. A number of safety issues have been raised during studies on the seismic isolation systems adopted for the JHR and ITER facilities (damping pads arranged underneath the buildings). Other mixed systems that combine passive isolation with active (or semi-active) control, or three-directional isolation, which is being explored by the Japanese, could also prove useful for France's NPPs. In 2013, R&D work was begun with CEA to provide knowledge with an application part of value for the expertise[218];

– the seismic behavior of nuclear structures without earthquake-resistant provisions, more particularly the floors made up of preslabs. Such floors are found in particular in the fuel cycle facilities (laboratories and plants) built between 1960 and 1990. IRSN, working with CEA, has embarked on an experimental program on this issue to study the ultimate behavior of these floors and identify their modes of collapse as a function of loading. This program assesses the relevance of checks proposed by technical regulations based on loads obtained by an analysis conducted by assuming the linear behavior. The full-scale mock-ups were thus developed by making sure replicate, as closely as possible, the building practices—in terms of concrete characteristics and implementation methods—that were in use when the facilities were built (1960–1990). These mock-ups were subjected to static tests by CEBTP and dynamic tests by CEA on the AZALEE shake table. The experimental program ended in 2013, and the results have been analyzed [4, 5]. A finite-element numerical model simulating the mechanical behavior of this assembly was used to analyze the behavior of these complex structures. The comparison of the test results with those of the simulations with finite-element models, which presupposes the monolithism and mechanical continuity of the structures, confirmed the monolithism of the behavior this complex, for the seismic loads explored, despite the absence of seismic protection provisions.

10.2.3. R&D on the behavior of civil engineering structures in cases of collision

Improving methods for assessing the impact vulnerability of nuclear civil engineering structures (following a load drop or due to a projectile) has been the focus of R&D since the beginning of France's NPPs program. The risk taken into consideration was an accidental crash by a Cessna business jet. The aircraft was assimilated with a rigid projectile colliding with the wall of a nuclear building such as a containment. This R&D work was extended to military aircraft in the late 1980s. Since the attacks of September 11, 2001, this research was further extended to assess the risks in case of a commercial wide-body aircraft impact.

These efforts can currently be split among three areas of R&D:

– an experimental area, with medium-speed collision tests (commercial aircraft) on reinforced concrete slabs as part of the IMPACT experimental program conducted at VTT in Finland. This program, which began in 2005, is being carried out in collaboration with the U.S.NRC, HSE, STUK[219], ENSI[220] and GRS;

218. A PhD thesis was started in 2014.
219. Finnish Radiation and Nuclear Safety Authority.
220. Swiss Federal Nuclear Safety Inspectorate (Eidgenössisches Nuklearsicherheitsinspektorat).

- numerical simulations conducted with fast-dynamics codes (LS-DYNA and RADIOSS). The international benchmark IRIS 2010[221] was launched and led by IRSN, under the leadership of the OECD. Its aim was to assess the ability of various codes and teams to predict the behavior of structures impacted by hard and soft projectiles. This benchmark brought together 28 teams from 11 countries. Following the findings and recommendations from this first exercise, two new benchmarks have been scheduled with OECD's partners:
  - the first took place in 2012 to calibrate the first simulations and present simplified models—by providing the partners with the test results and test conditions (actual boundary conditions), as well as the characteristics of the materials used,
  - the other began in 2014 and is being conducted to study vibration propagation in a structure. A mock-up is being built to form the experimental basis of this benchmark.

  The knowledge obtained from this area of R&D has already enabled IRSN to draw up an initial draft of recommendations for the fast-dynamics analyses conducted using numerical simulations, documented in a report by the OECD;
- characterization of impacted material, modeling, and validation. The aim of this area is to identify influencing parameters and improve the rheological model of the behavior of concrete subjected to impacts. This aspect was researched[222] with Joseph-Fourier University and CNRS and culminated, in 2013, in the confirmation of the influence of the moisture content of concrete and the rate of deformation on the dynamic behavior of impacted structures.

10.2.4. R&D on the behavior of containments during core melt accidents

A) Assessment of air and steam leaks through an idealized crack

In 1989, IPSN commissioned the Mechanical and Thermal Engineering Department (DMT) at CEA's Saclay research center to conduct an R&D program on local cracking in order to quantify leakage through a containment wall during emergency situations (air and steam) compared to dry-air leaks quantified during in-service testing. Known as SIMIBE, this program comprised tests conducted on two glass plates simulating local cracks. It resulted in the development of a numerical model integrated in the Cast3M code and able to simulate the two-phase behavior of the air-steam mixture in such cracks. SIMIBE was used to interpret the MAEVA tests described below and, in 2000, allowed IPSN to consolidate its position in relation to the air and air-steam transposition factor, i.e. use a conservative transposition factor of 1. The transposition of glass to concrete required additional research to take into account the permeability of cracked concrete and the communication between local cracks. This additional research, based on the double-porosity theory whereby exchanges occur between the two media (local cracks and the porous medium), was implemented in the ECOBA[223] project described below.

221. Improving Robustness assessment of structures Impacted by missileS.
222. Doctoral thesis.
223. Study of Reinforced Concrete Containment Structures.

B) Assessment of air and steam leaks through a crack under conditions representative of a containment

The assessment of air and steam leaks through a crack under conditions representative of a containment during emergency situations required using large-scale tests representative of such conditions. It has been conducted as part of experimental programs since the early 1980s, which include:

- the RCCV tests conducted in 1984 at Sandia National Laboratories in the United States (tests on a complete containment—made of non-prestressed concrete—of a Westinghouse 900 MWe reactor, at 1/6 scale, with a metallic liner;
- the MAEVA tests conducted in France between 1994 and 2002;
- the PCCV tests conducted at Sandia National Laboratories in 2000 on a mock-up of a Japanese PWR containment, at 1/4 scale (prestressed concrete, with steel liner);
- the VK2/2 tests conducted in 2001 at the University of Karlruhe in Germany (tests conducted on a 1.2 meter-thick pre-cracked slab extended by jacks);
- the tests in the ECOBA program (ANR project);
- the nationwide CEOS.fr project mentioned above;
- the VERCORS, also implemented by EDF.

These experimental programs were accompanied by benchmarks and workshops.

As a reminder, in single-wall containments, confinement is ensured by the steel liner (case of the RCCV and PCCV mock-ups). The goal of research programs on double-wall containments (MAEVA, VK2/2 and ECOBA) is to assess the leakage of the inner wall of prestressed reinforced concrete.

The key difference between these two types of mock-ups is the damage mechanism—a failure mode for the first and leak mode for the other.

Some of the aforementioned tests are discussed more in detail below.

► Tests conducted on the MAEVA mock-up

In 1994, EDF decided to build a containment mock-up in order to study its mechanical strength and better assess the leakage of the double-wall containments of 1300 MWe reactors. EDF's aim was to conduct an experimental study of the thermomechanical behavior of the inner prestressed concrete wall for design-basis and beyond design-basis situations (core melt) by subjecting the mock-up to sequences of increased pressure and temperature. These aims were as follows:

- assess the air and steam leakage rates under accidental conditions compared to those measured with dry air during in-service testing;
- study the behavior of the composite liners placed on the intrados of the inner wall for the in-service tests and the various accidental scenarios as well as validate their conditions of industrial implementation.

IPSN participated in the MAEVA tests, in particular by helping to define two sequences: one involving an increase in air pressure and steam (to simulate a core melt accident) up to the design-basis pressure initially intended for the EPR containment without steel liner (6.5 bar absolute), the other up to a greater pressure. IPSN also made a number of measurements (helium tracing of leakage rates, quantification of steam titers in the water-steam mixture in the core-melt simulation sequence).

The MAEVA[224] mock-up (steam and air containment mock-up) represented a straight portion of the inner wall of the containment, at 1/3 scale for the diameter and 1/1 scale for the wall thickness. The annulus space between the two walls was also represented, but the outer concrete wall on the mock-up was replaced by a steel wall (Figure 10.19). The concrete wall thus had a diameter of 16 m, a thickness of 1.2 m, and a height of 5 m. The mock-up was built at the Civaux NPP using concrete of the same characteristics as those of the concrete used for reactor No. 2 of Civaux's NPP (high-performance concrete). The top slab was supported by four prestressed concrete columns arranged at each quadrant of the surface area. The inner wall was divided into quadrants, two of which were lined with a composite material similar to that used for repairing reactors in operation.

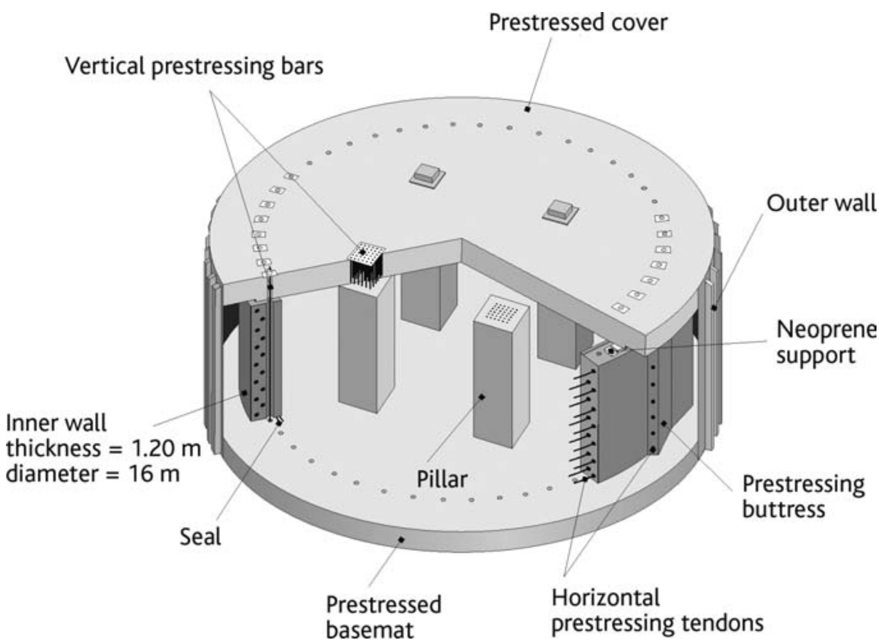

Figure 10.19 Diagram of the MAEVA mock-up. © Georges Nahas/IRSN.

224. Steam Air Exchange Model.

The prestress had been calculated to obtain, as for the containments of the 1300 MWe reactors in operation, a mean residual compression of 1 MPa for an internal pressure of 6.5 bar absolute.

Seven test sequences using air and an air-steam mixture were conducted with the mock-up. For the sequence simulating a core melt accident with air and steam, the pressure was increased in three steps (2.6 bar absolute, 5.3 bar absolute, and 6.5 bar absolute). For the air sequence conducted to study the behavior of the containments beyond their design-basis pressure, the pressure was raised to 9.8 bar absolute.

During each test, measurements were taken to determine the leaks in the annulus space—divided into four sealed quadrants (each quadrant being referred to as a chamber—for dry air and an air-steam mixture—, various temperatures and pressures, and displacements of the inner wall of the mock-up. During the air and steam tests, the extrados of the mock-up's wall was heated to 60 °C in the chambers to help vaporize the water exiting the cracks and reduce the temperature gradient in the thickness. The cracks were identified, the composite liners were auscultated visually, and pull-off strength tests of the liners were conducted. The pull-off strength test made it possible to quantify the adhesion of the liners to the concrete wall after the test sequences.

The analysis of results also made it possible to determine:

- the transposition factor to be used between the leakage rate for dry air and the leakage rate an air-steam mixture (factor of 1). This factor is then used to determine the leakage rates of the actual containments during accidental situations from the leakage rates measured during the tests;
- changes in cracking and measured leaks for a number of accidental scenarios.

These last results on the changes in cracking and measured leaks were applied to validate the leakage quantification method (nationwide CEOS.fr project and the ECOBA project).

Furthermore, the measures taken during the air and steam sequences showed that the leakage rates collected in the chambers were quite high and greater than predicted by EDF, which believed that the steam would condense in the cracks and remain in the reinforced concrete wall. This assumption was therefore invalidated.

These observations accounted for much in considerations on the robustness of the EPR. Realizing that composite liners would not ensure prolonged integrity during core melt conditions—a design basis of the EPR—EDF decided to install a steel liner on the intrados of the containment of the Flamanville 3 reactor.

► ECOBA Project

The ECOBA research project to study the confinement properties of reinforced concrete structures (2010–2014) was conducted by the French National Research Agency (ANR) as part of its "white" program (interdisciplinary projects in all fields of research). ECOBA brought together IRSN and three academic laboratories with complementary skills: GeM[225], the ENS Cachan, and the University of Pau and Pays de l'Adour (UPPA). Using a full-scale mock-up representative of an inner containment wall of a 1300 MWe (P4 series) reactor, its aim was to study the various mechanisms of concrete cracking in order to establish relations between damage, cracking, and permeability. It was based on tests conducted at the GeM laboratory of the École Centrale de Nantes on two mock-ups representative of a straight portion, referred to as the "effective area" (1.50 m wide, 1.20 m high and 0.90 m thick) of an inner wall of the containment. Each mock-up, which measured 3.90 m wide and 2.40 m high (overall dimensions) and weighed 20 tonnes, was built at the experimental laboratory of the École Centrale de Nantes. Both were instrumented with vibrating-wire strain gauges identical to those used in the actual containments and supplemented by various measurement devices to validate the results and obtain a robustness of the measurements. The mock-ups were subjected to direct tensile stress —four jacks driven load and displacement—to create cracks representative of those that could appear in a containment during accidental loading. A metallic box for injecting either air or air and steam was fitted upstream of the areas of cracking. The leakage rates through the cracks were measured using a gas tracer dilution technique developed by IRSN and which made it possible to identify the surface openings of the cracks.

The first round of tests made it possible to observe the development of cracks in the effective area—which were similar to those seen on actual structures subjected to these types of loads—and quantify the leakage rates in the sequences with air injection. The first set of results confirmed the project's relevance to progress in characterizing the parameters that affect the integrity of NPP containments and to consolidate the assessments conducted on the confinement capability of the containments in the event of core melt accidents.

► VERCORS Project

In 2013, EDF launched a far-reaching project of studies and R&D on reactor containments. Known as VERCORS (for *Vérification réaliste du confinement des réacteurs*), its aim is to acquire sufficient knowledge in order to demonstrate that NPPs can be safely operated for a period of 60 years. VERCORS is expected to continue until 2021. Its goals are as follows:

- experimentally demonstrate the strength of the containments in core melt situations (under simultaneous and maintained pressure and temperature loads);
- predict by experience changes in containment leaktightness;
- improve knowledge about leakage and the models used to predict their changes;
- find new tools of leak detection and quantification.

A number of tests will be conducted on a large-scale and thoroughly instrumented containment mock-up. In light of the feedback obtained from tests conducted on large-scale mock-ups, such as MAEVA or in Sandia National Laboratories, EDF decided to build

225. French Civil and Mechanical Engineering Research Institute. A joint research unit brings together the French National Center for Scientific Research (CNRS), the École Centrale de Nantes, and the University of Nantes.

a mock-up of a double-wall containment (P'4 series 1300 MWe reactors) at the same scale for the large dimensions and thickness (1/3). This mock-up will be built on hard soil and feature the same discontinuities (such as the ledge used for the crane runway), penetrations (in particular the equipment hatch), prestressing cables and rebars. The 1/3 scale will make it possible to accelerate some physical phenomena inherent in concrete, in particular:

– shrinkage/curing of a factor of 9,

– creep of a factor of 3,

this gives an equivalent aging factor of 7 on average.

At this scale, around nine years will suffice to simulate 60 years for the actual containments.

The ten-yearly tests will thus be conducted on the mock-up once every 14 months. The core melt accident will be simulated after the first nine years.

Because the delayed strains (shrinkage and creep) in the concrete that will be used are moderate, the extrapolation of the results will have to be examined closely in the case of the susceptible containments, for which delayed strains are greater.

C) Assessment of non-localized leaks in a concrete wall

The results of the experimental programs on the large-scale mock-ups together with the observations and the measurements of the leakage rates during the ten-yearly tests of the containments reveal the existence of non-local leaks—referred to as non-localized leaks—that contribute significantly to total leaks.

Each PWR containment in the France's NPPs has an inner surface measuring around 10,000 m^2. As a result, non-localized leaks through these containments as a function of the applied load are likely to become significant. Research, which ended in 2011, was conducted with the École Centrale de Nantes on the gas (nitrogen) and steam permeability of concrete under compressive mechanical loads. The experimental setup made it possible to measure the air-steam mixture permeability of 11 × 22 cm hollow concrete specimens. The concrete formulation was representative of that of the concrete of the containments themselves. The results revealed a difference in flow between steam and nitrogen and made it possible to identify a difference in the flow times of the gas and steam. Nitrogen flows through the accessible porous network in a matter of minutes, whereas steam takes much longer (around 10 to 25 hours depending on the injection pressure).

This observation partially explains the time difference between the air and air-steam leaks observed during the MAEVA tests.

D) Study on the early-age behavior of concrete

Early-age behavior of concrete is an important topic. During this complex phase, which spans from the setting of the concrete to around one month thereafter, exothermic physicochemical changes followed by shrinkage and creep phenomena create weak areas in the concrete's structure. Once concrete shrinks (Le Châtelier effect), the combined changes in the temperature gradients throughout the structure and the mechanical characteristics of the concrete (Young's modulus, tensile strength limit, compressive strength limit, etc.) cause the concrete to crack. The extent of these cracks depends on parameters such as the type of formwork, the outdoor temperature, the geometry of the structure, and the amount of rebars.

These cracks will affect both the durability and mechanical behavior of the structure in the event of accidental loading with propagation of these cracks. In addition, the physical and chemical changes may, under certain conditions, lead to subsequent pathologies such as delayed ettringite formation (DEF – see below).

IRSN began researching this aspect in 2007 in order to study the behavior of civil engineering structures from their construction (concrete pouring) onward. Concrete hydration reactions, with the resulting temperature gradients, water exchanges and the incompatibility of stresses of the constituents as well as the restraining caused by resumption of concreting, which causes concrete to crack, were identified as areas of research to be developed.

The first area looked at was the effect of early-age cracking of concrete on its air and gas permeability. This effect was analyzed by taking into account the rate of cooling, the rebars, and resumption of concreting. This research was begun in 2007 (PhD thesis) with the ENS Cachan. The experimental part of this thesis comprised tests on a 10 × 10 cm concrete ring representative of that in the containments and heated to a temperature representative of that measured at the core of the early-age concrete. It yielded important data on the influence of construction methods on the creation of the first structural cracks. IRSN used the results obtained to assess the method used by EDF to resume concreting of the Flamanville EPR containment.

This research was followed by another thesis (2011–2014) on the modeling of the aforementioned phenomena supplemented by shrinkage and creep, which also occur during the early-age phase. This research is currently being applied in the numerical simulation benchmark of the VERCORS mock-up.

In addition, the quantification of early-age exothermic phenomena also makes it possible to assess the temperatures reached during pouring of fresh concrete and the risk of onset of pathologies, in particular DEF.

E) Development of mesoscopic simulation models

Analysis of the results of the observations and measurements obtained from the experimental programs and ten-yearly tests of the containments show the importance of characterizing the geometry of local cracks and porosity (corresponding to non-localized leaks) as a prerequisite to realistic assessments of leakage rates. This characterization requires using mesoscopic models—i.e. models at the scale of cracking—to be able to simulate crack openings of less than 1 mm in concrete. High-performance simulation codes are required to model at the mesoscopic scale. Research conducted by IRSN, working with the University of Pau and Pays de l'Adour, have made it possible to identify important parameters, in particular initial structural stresses and their effect on the creation of cracks in concrete during accident conditions, communication between microcracks and local cracks, and residual opening of cracks after unloading. These data may be applied to the macroscopic models used by IRSN for its studies in support of the level-2 PSAs on the behavior of double-wall containments of France's NPPs, a particularly important point for the safety reviews associated with the third ten-yearly outages of 1300 MWe reactors.

10.2.5. R&D on containment aging

Aging is the process of changes over time to reinforced concrete structures such as reactor containments. Aging is caused by two phenomena: a natural phenomenon driven by delayed strains in concrete (shrinkage and creep) and a phenomenon caused by the onset of pathologies.

A) Delayed strains in concrete

In the case of containments, this aging process decreases structural strains, causing the prestressing cables to lose tension and thus decreasing the compression in the containment walls. Because this compression makes it possible ensure the necessary level of confinement required for these walls, its changes must be controlled throughout the life of the structure.

R&D on this topic are thus conducted to assess the delayed behavior of the reactors containments and its consequences on the ability to confine radioactive material in the event of an accident. This research comprises theoretical parts and experimental parts, with numerical simulations. Tests are performed on concrete specimens representative of the concrete of the containments. The main purpose of this work is to obtain the scientific knowledge and skills needed to make decisions on the risks caused by changes to delayed strains in concrete. Research is primarily conducted on the following topics:

- **Tensile creep of concrete and its effects on containment leaktightness**: research was started in 2005 with IFSTTAR and the ENS Cachan. Tests were performed on cylindrical hollow specimens to study the effects of creep on concrete leaktightness. The tensile-loading time was limited to 36 hours (in-service testing time of the containments). A numerical model was developed to simulate the mechanical behavior of the containments. The data acquired during this research made it possible to better understand the changes in cracks observed during the ten-yearly tests and the progression of cracking during consecutive pressure-hold steps.
- **Multiaxial creep of concrete and its effect on leaktightness**: the containments are biaxially prestressed structures. To take into account the effects of delayed strain (concrete creep and shrinkage), these structures were designed in accordance with the RCC-G[226] and using formulae in French standards and codes. However, feedback has shown that these formulae underestimate the effect of these strains for multiaxial loads, a fact borne out by numerical simulations

226. Rules on the Design and Construction of Civil Engineering Structures.

conducted by IRSN. The reason is that these formulae were established from observations and tests on primarily uniaxially loaded structures. Furthermore, monitoring activities conducted by EDF on the containments of its NPPs are not readily usable in the calculation models because the measured strains are due to a combination of phenomena (basic creep, drying creep, drying shrinkage, autogenous shrinkage). Research to quantify these phenomena was begun in October 2011 with the ENS Cachan. This quantification will make it possible to better simulate the delayed behavior of the containments, assess the solutions proposed by EDF to reduce the kinetic of these strains (such as wetting of the containment facings), and predict containment behavior to extend the operation life of reactors. Once data has been acquired from the other research, the final application will be to simulate a containment from its construction lift by lift with the prestressing phase, delayed behavior, ten-yearly tests, and end of operation. This model will make it possible to simulate the changes over time in the confinement capability of the containment, a decisive factor in decisions on whether to extend the service life of reactors.

B) Pathologies of reinforced concrete structures

In order to take into account all the phenomena that may alter the safety functions of nuclear facilities, and of containments in particular, IRSN deemed it essential to initiate research programs to investigate concrete and rebar pathologies. One reason in particular for this research is that EDF is seeking to extend the service life of France's NPPs to 60 years.

The development of pathologies in reinforced concrete structures can lead to damage that could affect the mechanical properties of the concrete and even undermine the confinement capability required from some structures. Swelling reactions—such as alkali-aggregate reaction (AAR) and delayed ettringite formation (DEF) in particular—are two examples of potentially harmful pathologies.

EDF conducted research on AAR (a chemical reaction between some types of aggregate and the cement matrix) and its change over time into an expansive gel and pop-outs around reactive aggregate. EDF has classified all of the containments in its fleet into five categories (from 0 to 4 and ranging from zero risk to very high risk) according to the characteristics of the concrete used and the environment of the structures.

DEF—which is related to the type of concrete, its environment, and its core temperature during concreting—is identified as a risk for nuclear structures. DEF can be triggered by temperatures above 65 °C during construction of structures as well as by certain moisture conditions. In 2009, IRSN alerted EDF (with ASN) about the importance of such a risk of DEF. This alert was based on feedback from research programs conducted by reconstituted concrete and led by IRSN in collaboration with IFSTTAR (Figure 10.20).

The temperature of concrete play an even greater role in the development of DEF, in particular in the case of structures which may, during their operation, be taken to temperatures of around 80 °C followed by a cooling cycle. Recent research conducted by

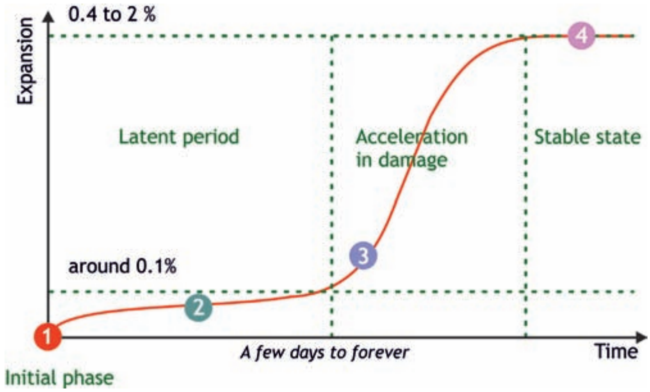

Figure 10.20 S-curve of the change in DEF proposed by X. Bruneteaud in 2005 [7]. © Georges Goué/ IRSN–Source: Xavier Bruneteaud.

the scientific community and verified by IRSN (EDF-funded tests performed on specimens at IFSTTAR [6]) have confirmed that delayed DEF can occur under some conditions even though the concrete has not been subjected to temperatures above 65 °C when it was made. Such a configuration could occur in the containments at the reactor vessel and the steam-tube penetrations, some fuel-assembly storage facilities, or after a controlled fire. IRSN is assessing the opportunity of conducting a research program on this subject.

Another parameter, related to the type of cement, affecting the development of DEF is the use of limestone filler[227] as additives. Research started in 2009 (PhD thesis) in collaboration with IFSTTAR and the ENS Cachan made it possible to study the effects of these additives as well as other parameters that influence the development of this pathology, such as the moisture content of the outdoor environment, concrete permeability, and their effects on reaction kinetics.

EDF has classified all of the containments in its fleet according to the characteristics of the concrete used and the environment of the structures. The ranking of these five categories, which range from zero DEF risk to very high DEF risk, is similar to that used for the AAR.

An analysis of both maps of the potential risk of containment pathology, derived from the classifications, reveals cases where this risk is high for both pathologies. These cases mean that R&D work must be conducted until feedback or experimental results are available. In 2014, IRSN—working with IFSTTAR and the ENS Cachan—began research (PhD thesis) on the accelerated aging of concrete affected by both pathologies together and separately.

In the current state of knowledge, there are no curative solutions for these swelling pathologies. The only recommendations are to monitor structures and, in the case of structures with high and very high risks (see, for example [8] and [9]), limit the ingress of water from external sources.

227. Dust from the cutting of limestone that is added to concrete and other building materials.

C) Predicting the effects of aging: the ODOBA project

In light of the importance pathologies have on the longevity of reinforced concrete structures and the plans to extend the service life of France's NPPs, IRSN decided the time was right to create an observatory on the longevity of nuclear civil engineering structures. This observatory will be shared with the scientific community and will yield data on the aging of these structures and the defects that could affect them.

IRSN therefore created the Observatory of the Durability of Reinforced Concrete Structures (ODOBA[228]) in 2014. ODOBA's purpose is to study pathologies found in nuclear civil engineering structures, such as rebar corrosion (due to chloride ions or carbonation), concrete swelling (DEF, AAR) and leaching, and their effects on the safety requirements for these structures.

Knowledge will be acquired concomitantly with the assessment of plans to extend the service life of France's NPPs beyond 40 years. The experimental part comprises the construction, at IRSN's Cadarache research center, of concrete structures representative of the concrete of the containments at France's NPPs. These structures—60 blocks measuring 1 m thick and several meters in height and width—will be subjected to either an accelerated aging process or the natural aging process in order to determine the equivalent durations of accelerated aging. A first difficult task was finding the quarries and cement used to build a few containments classified by EDF as having potentially high or very high risks of swelling reactions. This work was conducted with the assistance of IFSTTAR and consisted in investigating the archives of the French Central Laboratory for Bridges and Roads (LCPC). This investigation made it possible to locate nine quarries and their cement for 13 susceptible NPPs. The ODOBA project will use equivalent generic concrete for the other susceptible NPPs.

The scientific part of ODOBA is led by a scientific committee comprising IRSN, the ENS Cachan, IFSTTAR, the Construction Materials and Durability Laboratory (LMDC) in Toulouse and the Mechanics and Acoustics Laboratory (LMA) in Aix-en-Provence. The purpose of this committee is to monitor the project's scientific progress and verify the scientific relevance of the choices made as part of it.

ODOBA is divided into a number of stages that will make it possible to obtain intermediate results aligned with IRSN's assessment needs, such as analysis of the ten-year safety reviews conducted on France's NPPs.

In 2015, IRSN signed a cooperation agreement with the U.S.NRC to share the results of experiments on the pathologies of the concrete used in nuclear reactor containments. Other organizations are interested in ODOBA and discussions are under way to define the conditions of their involvement.

The first concrete blocks have been cast in 2016.

228. Monitoring Center for the Durability of Reinforced Concrete Structures.

D) Non-destructive testing of structures

Non-destructive methods of testing for defects in structures are useful tools for assessing the health of susceptible structures such as reactor containments and structures were sampling is very limited and controlled. IPSN then IRSN conducted R&D efforts with CEA's Saclay research center to study the possibilities offered by ultrasonic waves. These inspection methods have proven their effectiveness for metallic structures and make it possible to detect defects. A forward-looking development project conducted with CEA's Saclay research center consisted in adapting these methods to concrete by studying the various types of probes and the signal frequency to be emitted, and by adapting the parameters of the CIVA simulation platform and its simulation code for predicting noise and attenuation phenomena. Currently, the operational monitoring capability of this method is limited to a thickness of 40 cm and a very low amount of rebars. The results obtained by this feasibility study have shown that this method cannot be used on the containments.

In the frame of the ODOBA project, the need for non-destructive testing methods meant that other techniques and methods had to be found. These include the use of fiber optics and novel non-destructive testing techniques studied by the LMA, the LMDC and IFSTTAR. These new techniques are being developed to detect the onset of defects and pathologies in the concrete of structures. They will be validated by collecting core samples from the blocks.

10.3. Research on polymers

The main purpose of research on polymer aging (polymers are used for electrical wires, seals, and coatings) is to study the effects of radiation and temperature on the degradation of polymer properties.

For licensees, electrical wiring, which is located inside the reactor building and provides a safety function, is not easily replaceable. It must therefore perform its function throughout the life of a NPP, including in the event of an accident at the end of the facility's service life.

The insulation and jackets of wiring are made of polymers. Temperatures and irradiation can, depending on their intensity and synergies, break up the polymer chains or lead to crosslinking (creation of bonds between the chains), oxidation, or even a loss of plasticizers due to hydrochloric acid migration. Understanding these mechanisms is necessary in order to assess the relevance of the conditions of accelerated aging used during qualification or during accelerated aging simulations. This is because the electrical properties directly related to the functionality of the wires and cables generally do not vary much before significant degradation of the polymers occurs. Indicators of aging must therefore be looked for among the mechanical and physical and chemical properties of polymers.

The service life of wires and cables can then be predicted using experimental data and extrapolation models.

The purpose of a first study initiated by IRSN was to assess the aging resistance of EVA polymer cables used in the N4 series reactors. Sections of cable were aged thermally then by irradiation at dose rates between 3 and 1000 Gy/h. They were then subjected to the accident conditions that would result from rupture of a RCS pipe. The first results showed that EVA is highly stable, virtually immune to temperatures and the dose rate provided that the antioxidants in the material have not been completely consumed.

Furthermore, in order to determine the representativeness of the conditions of accelerated aging, new PVC and EPR/Hypalon[229] cables were laboratory aged using an accelerated method. To define these tests, IRSN assessed the acceleration of aging so as not to alter the aging mechanism foreseeable under actual conditions; this assessment was conducted using a similitude of the experimentally determined activation energies of the mechanisms of degradation of the materials. The mechanical properties of these aged cables were compared to those of identical cables collected from the Cruas site after seven years of operation. This comparison revealed that the properties of the cables collected from Cruas and the laboratory-aged cables were similar.

IRSN thus demonstrated that it is possible, after a study of the degradation phenomena of materials, to establish accelerated aging conditions that are representative of actual aging.

Since 2014, IRSN has been conducting a research program to study the impact of an accident on aged silicone seals of the equipment hatch. The IRMA irradiator and the EPICUR facility in particular are being used for this research. Experiments conducted on mock-ups to assess the temperature (40 °C to 150 °C) and irradiation (25 kGy to 75 kGy) effects show that the properties of the silicone are greatly affected by these exposure conditions. All the tests under LOCA conditions, after ageing, have been conducted to date; the results will be synthetized in 2017.

Accelerated aging tests of 3–4 mm coatings made of epoxy-matrix composites were conducted by CIS-bio international (CEA–Labra[230]) for IRSN between 2003 and 2008. Two epoxy resins (manufacturers: Max Perles and Chrysor) similar to those used by EDF to line the intrados of the containments of its 1300 MWe and 1450 MWe reactors were used for these tests. A variety of dose rates (1 to 100 Gy/h), doses (40 and 320 kGy) and temperatures (40 °C to 70 °C) were applied. The results showed that the epoxy resins were highly susceptible to oxidation. The adhesion of all of the coatings dropped significantly after the four postulated LOCA tests. A critical analysis of this program in relation to the tests presented by EDF in 2014 is under way. A new R&D program will be proposed in 2017 to deepen understanding on the degradation mechanisms identified and their impact on integrity.

New research on polymer aging (cable insulation, paint, resins, and anti-seismic damping pads) is being scheduled.

229. Insulation made of cross-linked ethylene/propylene with a Hypalon jacket.
230. Laboratory for Applied Radiation.

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Chapter 11
Research in the Field of Human and Organizational Factors and, More Broadly, in Human and Social Sciences

11.1. From the consideration of human factors in safety to studies in human and social sciences

Human and Organizational factors (HOF) is a relatively recent discipline born with the technological developments of the 20th century. These factors have played a central role in most nuclear accidents. Interest in HOF is growing in the nuclear power industry, and some consider that it can increase safety more than any other discipline.

While the measures implemented in the wake of the Three Mile Island accident (TMI) focused in particular on the ergonomic and cognitive aspects of workstations, the Chernobyl accident raised questions of another kind—organizational factors. The development of a safety culture in nuclear facilities has generally been considered to be the appropriate response. This concept was described in the INSAG-4 report, published in 1991 under the aegis of IAEA, just five years after the Chernobyl accident.

More recently, the 2011 Fukushima Daiichi accident revealed the importance of societal factors in the risks governance in general.

Initial research, conducted in the 1970s, attempted to better understand human "functioning" and its impact on the performance of NPP operators and technicians not just in the control room, but also during all tasks and activities carried out (tests, maintenance, in-service inspections, etc.) outside the control room and which could affect safety. The scope of this research was subsequently extended to understanding and assessing organizational factors, then societal factors. IRSN's research also extends to methodological aspects in order to build and improve baselines and approaches for conducting its assessments.

The documents of the OECD/NEA referred to herein [1] to [10] testify to the international community's continued interest in human and organizational factors in the wake of the TMI accident.

While the very first research on this aspect in nuclear reactors began in 1977 at IPSN's Department of Nuclear Safety (influence of human parameters on safety, incident analysis methodology, human reliability, control room ergonomics), it was mainly after the TMI accident that human factors were taken into account in facility safety assessments. In the early 1980s, IPSN and EDF equipped themselves with greater resources by creating dedicated structures (French Laboratory for the Study of Human Factors (LEFH) at IPSN, Human Factors Group (GFH) at EDF). The topics of interest at the time included:

- incident analysis,
- operating staff training,
- operating procedures,
- human-machine interfaces and control-room ergonomics,
- organization of control-room staff,
- communication between control-room staff,
- the use of expert systems,
- teleoperation.

In 1989, IPSN also focused on other topics, such as:

- maintenance of France's NPPs during outages,
- operation *via* computerized procedures,
- fieldwork in maintenance.

These three topics were researched in collaboration with EDF or CNRS. The adoption by EDF of a new organization of the operation of its NPPs prompted IPSN to research this topic further.

Starting in 2003, IRSN's research increased in the frame of theses and began publishing, on topics such as outsourcing, equipment or organizational modifications in facilities in terms of human and organizational factors, etc. Outsourcing appeared to be a particular subject of concern in terms of human and organizational factors, as EDF called on many contractors to carry out work during unit outages.

In 2012, with the creation of the Human and Social Sciences Laboratory (LSHS) and the post-Fukushima period, IRSN extended its research to social aspects around the governance of nuclear risks seen from the viewpoint of organizational as well as cultural factors.

Moreover, following a number of radiation therapy accidents that occurred in France, particularly in Épinal and Toulouse, IRSN extended its research to this issue, focusing on the appropriation of new technologies and conformity management[231].

Generally speaking, a particular feature of HOF research conducted by IRSN is that it is largely based on analyses in the field (interviews, observations of projects and of work situations, such as during "sensitive" activities, etc.)—which, when these analyses are conducted as part of safety expertise, are set out in protocols established between IRSN and the nuclear installation operator. The very nature of the topics researched and the specific conditions of the assessments carried out thus provide IRSN with a cross-disciplinary and hands-on view of the maintenance, organizational and management practices at NPPs. Research may also draw on simulations, as we will see further on with the Halden Reactor Project.

Research is conducted with partners from academia, research organizations (such as CNRS aforementioned), potentially with industrial companies, and/or as part of national or international projects (such as the Halden project–see "Focus" further on). IRSN, in collaboration with AREVA and French naval contractor DCNS[232], created the Chair of Safety, Organization, and Human Research (RESOH[233]) at the École des Mines de Nantes in 2012. This five-year chair is dedicated to the management of safety in high-risk industries, particularly the nuclear industry. Moreover, in 2013, ANR decided to fund the AGORAS[234] project led by the École des Mines in Paris and in Nantes and the project led by the Center for the Sociology of Organizations of Sciences-Po on risk governance and emergency management.

A few of the most significant projects involving[235] IPSN (then IRSN), both past and current, are presented below. As we will see, this research is basically intended to understand the mechanisms behind the various activities that involve people and organizations in order to bring out, as much as possible, relevant levers to the safety of nuclear facilities.

231. In collaboration with Paris 8 University for the first topic and La Pitié-Salpêtrière hospital for the second topic.
232. French hi-tech company specializing in defense naval systems.
233. Research on Safety, Organization and Humans.
234. Improving Governance of Organizations and Networks involved in Nuclear Safety.
235. Including thesis work.

11.2. Studies and research on reactor control room design

11.2.1. The post-TMI period

A) Background

The 1970s were marked by the implementation of centralized instrumentation and control systems in industries referred to as "continuous process": iron and steel production, fine chemicals, food processing, etc. France's nuclear program benefited from these technological advances. During this decade, IPSN was confronted with the issue of analyzing the safety not only of the control rooms in France's PWRs (900 and 1300 MWe reactors), but also those in the SUPERPHENIX fast neutron reactor and the La Hague plants.

The design of the control rooms raised questions about the work of operating crews. Industrial companies thus wondered about what monitoring in the control room entailed, how information was displayed on the control room consoles, how information was passed between roundsmen and the operators in the control room, and many other aspects. Researchers in ergonomics (specialists in human-machine interfaces and human-system interactions), especially in Europe, rallied to answer these questions[236]. Their research influenced and contributed to IPSN's research.

It was in this scientific and technical context that the TMI accident occurred in 1979. TMI showed that the design of reactor control rooms plays a central role in the control of accidents, in terms of their prevention, the mitigation of their consequences and the recovery of the accident. The TMI accident was caused by a pressurizer relief valve that had remained stuck open after automatically opening to reduce a pressure peak in the RCS. This valve received a closure order, but did not close completely. However, the control interface indicated that the valve was closed because the information used by it was the order to close the valve, not its actual position—which could have been determined by a position sensor. As a result, there was confusion in the control room about the facility's actual status. Furthermore, the emergency shutdown of the reactor and the problems affecting the secondary system activated many alarms, causing the control console to light up like a Christmas tree. Because there was no way to rank the information displayed, the operating crew was quickly overwhelmed and unable to detect and extract the relevant information.

Following the TMI accident, EDF took a number of actions to improve the control rooms in its 900 and 1300 MWe reactors. These improvements were made to the control panels. The controls for functionally related equipment were grouped into color-coded areas, the main areas corresponding to overall functions were defined, and secondary areas related to sub-functions were defined within these main areas. A safety-parameter display system for controlling the reactor in emergency situations was also added to the

236. Especially V. De Keyser in Belgium, L. Bainbridge in the U.K., J. Rasmussen in Denmark, and A. Wisner and F. Daniellou in France.

control room. Other improvements were made to the organization of the operating crew. The position of "safety and radiation protection engineer" was created to independently monitor the situation from a viewpoint located away from the operating crew. The safety and radiation protection engineer makes a diagnosis of the state of the reactor (state-oriented diagnosis) whiled the operating crew uses procedures aimed at diagnosing events. The organizational logic of the operating procedures to be applied during emergency situations was changed as well (see Section Section 11.2.2 A further on).

In 1983, IPSN conducted a two-pronged study to analyze the improvement plan implemented by EDF. First, it compiled a summary of the results of research conducted on continuous processes and transposed them to the nuclear industry. Then, it conducted a study based on tests carried out on an operating simulator as part of a four-party agreement involving CEA, EDF, Framatome, and Westinghouse. IRSN used the results of these studies to issue an opinion in the improvement plan implemented by EDF.

B) Main findings from research on the operation of continuous processes

In the early 1980s, a number of studies were conducted on the activity of operators working in "continuous process" control rooms (refineries, cement works, iron and steel works). These studies focused primarily on operation under normal conditions. In its initial (internal) reports established in 1983, IPSN summarized the key results of these studies in order to raise awareness amongengineers in charge of conducting safety assessments or expertise on human-machine interactions at NPPs.

These studies revealed **the central role played by anticipation in control room monitoring**. For example, in the continuous-process industries studied, it was found that operators do not wait for an alarm or fault to appear in the control room. Instead, they seek to anticipate the physical and chemical changes of processes in order to act before problems occur. These operator interventions help to smooth out changes in the process, which is beneficial for production quality. They also make it possible to anticipate some transients that, although not risky in themselves, may put the operating crew in a difficult position or even put challenges of varying intensity on the facility.

These studies also showed that **monitoring is an activity that is focused as well as all-encompassing**. It is focused because operators have to concentrate all their attention on the actions that they are doing to avoid making mistakes. They have to direct their attention on certain parts of the facilities to be able to clearly assess various phenomena. At the same time, however, they have to maintain a bird’s eye view so as not to overlook malfunctions or in order to be able to notice overall changes in the process.

They also showed that monitoring is an activity that is training and experience-oriented. It is widely acknowledged that operators will prefer to monitor parameters related to functions important to production quality and operating safety. However, the study results enabled to qualify this statement. Indeed, operators integrate their operating experience and tend to prefer to monitor systems that are proned to malfunction or have adjustment issues. Equally, they tend to interpret changes in observed parameters first by referring to changes to parameters of the same type that they have previously encountered.

C) Four-party study program involving CEA, EDF , Framatome, and Westinghouse Operator action

Initially, the goal of this four-party program was to study the response time of operators during accident simulations in order to provide data for the probabilistic safety assessments (PSA). However, in 1981, EDF decided to supplement this quantitative approach with a qualitative analysis of the difficulties encountered by operators during simulation tests. Various baseline accident scenarios for PWRs (large RCS break, steam line break inside the containment, steam generator tube break, etc.) were thus "acted out" by operating crews on the simulator at the Bugey training center in April and May 1982.

IPSN conducted its own qualitative analysis of the data collected during these simulation tests. This analysis revealed that the recovery phase of accident is the most problematic for operating crews. The simultaneous management of several concurrent or conflicting objectives was identified as a source of difficulty. The crews that best succeed in overcoming these difficulties are those who have a good ability to anticipate changes in physical phenomena. In particular, it enables them to not focus on instantaneous changes in parameters. Crews who lack this ability tend to gather information from multiple sources and perform multiple checks, leading them to work "in fits and starts" and have a fragmented view of the facility. Team cohesion and proper coordination of the activities of each member also appear to be factors that can promote the development of foresight capacity.

Furthermore, at the end of the analysis of the simulation tests, EDF and IPSN considered that more specific behavioral data needed to be collected in order to better identify operator strategies and reasonings. Although collecting oral data during the interviews conducted at the end of the tests was envisaged, IPSN considered that it was important to couple the statement with objective data. This led to a feasibility study on the use of electrooculography[237] as a means of studying visual-exploration strategies during work in the control room.

11.2.2. The N4 series: the first computerized control room

A) New interfaces and new questions

Following the TMI accident, EDF decided to install a computerized control room in its future 1450 MWe reactor (N4 series). EDF wanted to benefit from advancements in computing that made it possible to filter and rank information and structure its presentation in control rooms. For example, there were now alarm processing and ranking systems, automatic operating sequences, and screens for displaying views of facilities by system, by function or by other logics.

EDF also decided to modify the design of the emergency operating procedures of its reactors by adopting the state-oriented approach (SOA)[238]. The purpose of SOA is to

237. A technique for following eye movements.
238. See "Elements of Nuclear Safety", J. Libmann, chapter 14–IPSN/Les éditions de la physique – 1996.

characterize the operational state of the reactor[239] and gradually bring it back within a safe operating range regardless of the events that caused the degraded conditions. An SOA procedure is a flow chart consisting primarily of questions that the operator must answer by "yes" or "no" and which provides the operations to be carried out depending on the answer. In the case of the N4 series reactors, these procedures are computerized. Using a trackball and the arrow keys of his/her keyboard, the operator follows the "steps" in the image of the flow chart. Information that the operator needs to follow the procedure (parameter values, excerpts of circuit diagrams, etc.) is displayed on the image. The key operating parameters are displayed in a "dashboard" on a second screen. On a third screen, the operator can display more detailed views of circuits, systems, and parameters (e.g. pressure/temperature diagram) and make the necessary adjustments. A number of steps in the flow chart are controlled by the computerized procedure. The procedure compares the operator's answers with those calculated by the computerized control system using data provided by sensors. If a deviation is detected, the relationship with the previous step in the flow chart is displayed in red. These are retroactive checks that the operator can either accept or refuse to take into account. This is referred to as "overriding the step." In addition, most of the procedures are monitored by the computer system, which periodically checks whether the answers given by the operator as he/she proceeds through the procedure are in line with the changes in the parameters. The system alerts the operator when an answer is no longer valid so that he/she can directly go to the relevant step.

The impact of these new operating arrangements on the activity of plant operators raised new questions. To validate its design choices, EDF equipped itself with a full-scale simulator comprising a mock-up of the future control room coupled with a "process" simulator to allow the crews to "experiment" with operation under various scenarios. Dozens of tests were conducted during campaigns held between 1987 and 1996. These consecutive campaigns made it possible to assess the control room's suitability to the requirements of the operating crew's activities. Operation under normal conditions was assessed during the first two test campaigns. Emergency operation was tested between 1994 and 1995, with the development of the computerized operating procedures.

Although the knowledge acquired by IPSN during the aforementioned research enabled to expertise the NPP operation under normal conditions, the computerization of the emergency operating procedures raised new questions, prompting IPSN to conduct new research.

B) Study on operator guidance during emergency operating procedures

In 1994–1995, EDF conducted a series of tests with a simulator to validate the use of the computerized procedures. The purpose of these tests was to ensure that the means provided to operators effectively enabled them to operate a nuclear facility in acceptable safety conditions, in particular during emergency situations. They consisted in closely observing how operators did their job, i.e. the suitability of their operations in the control room. At the end of the tests, IPSN considered that, in order to better understand how

239. The term "process" is also commonly used.

the operators performed, it was necessary to study their cognitive activities (perceptions, interpretations, reasoning, etc.):

- On what basis does an operator identify deviations?
- How do operators construct their vision of changes in reactor operation state?
- How do they understand how the procedure constructs its "own vision" of these changes?
- How do operators call on the rest of the crew to confirm their opinions?

Working with researchers at CNRS, IPSN analyzed data that it had collected during several tests that it had followed. These data consisted of:

- the facility parameters recorded by the simulator;
- audio recordings of the tests (alarms, information provided orally or by telephone, orders, keylogging (to follow operators as they proceeded through each step in the computerized procedures), private discussions in order to identify areas of confusion or hesitations, etc.);
- video recordings;
- comments provided by two control room experts: one with extensive experience in non-computerized control rooms and emergency operation using simulation technology, the other with solid technical expertise in computerized procedures.

This research made it possible to better understand the impact of step-by-step guidance of operators by the procedures [11]. Considering the possibility that operators could carry out unacceptable actions, it can appear appropriate to develop a form of guidance that would tell operators what actions to take. The computerized SOA procedures appear to have been engineered with this view. Indeed, they were designed to break down facility operation to a maximum, both in terms of diagnostics and the operations themselves. The horizon of each action is greatly circumscribed by the steps in the procedure, forcing operators to adopt a step-by-step approach to control.

Close analysis of the activities performed by operators to respond to the steps in the procedures revealed that they use a multitude of skills that are not made explicit by the procedures. The use of these skills assumes that the operators are trained. However, there are so many skills that it is unrealistic to imagine that operators will ever know everything. In addition, some skills are specifically tied to emergency situations and thus will only be used very rarely. To overcome these difficulties, the form of guidance must help operators to enlist their skills when following procedures.

The study showed that the visualization of the overall structure of procedures helps operators to understand the logic behind the current operation and helps them to enlist the necessary skills. Indeed, interpreting the previous steps in a procedure is almost always necessary to interpret a subsequent step. Equally, the relevance of the step's interpretation will be reinforced if operators can see the consequences of their choices. Conversely, strictly following each step in a procedure promotes operator lock-in that appears to lead to a type of withdrawal and passivity, making it sometimes difficult for operators to take initiative when necessary. Operators who passively follow procedures are not able to effectively enlist their skills. These observations concur with the results of two studies conducted in other sectors of industry. A 1987 study [12] on the interaction with an expert system for diagnostic in maintenance activities showed that this interaction resulted in inadequate results if operators passively follow the system's recommendations without constructing in parallel its own diagnostic. Likewise, a 1996 book [13] cites an experimental study on the interaction with a flight planning system. The study shows that *"the pilots allowed themselves to be influenced by the system and lost a certain level of critical thinking. They accepted illogical choices that they never would have considered without the system."* In order for guidance to be efficient, we must get better at taking into account the need for active involvement of the operator in control operations. Guidance thus must allow operators to take a certain distance from recommendations given in procedures.

This research also makes it possible to better understand **operators' positions with regard to guidance that they find unsuitable**.

These differences between the operator's viewpoint of the procedure's "viewpoint" occur in various situations. For instance, an operator may take changes trend in the reactor's state into account, whereas the step in the procedure reflects the reactor's instantaneous state. The operator may find it hard to follow actions that are recommended by the procedure but appear not to be optimized. The operator may find it hard to follow the recommended order of multiple actions whereas one of these actions is perceived as urgent. These differences in viewpoint reveal the operator's active position in relation to the recommendations in the procedure. They show his/her ability to resist the lock-in effects caused by the step-by-step guidance of the procedures discussed above. They can be beneficial to operators, as they provide opportunities for operators to check the validity of their interpretation of changes in reactor state.

Two outcomes are possible when an operator notices a difference between his/her point of view on changes in the process and that of the procedure: the operator either overcomes this difference and continues to follow the procedure or he/she stops following the guidance and risks becoming completely disoriented.

An operator moves past a difference when he/she factors in additional information allowing him/her to understand the reason for the difference. In this case, this difference becomes understandable and the operator can accept it—at least for a certain time. Take the example of an operator who wants to carry out an action not prompted by the procedure. This difference in viewpoint can be overcome if the operator sees that the action in question is dealt with by the procedure a few steps later in the flow chart. The operator can also overcome this difference, without understanding the logic behind it, provided he/she deems that it is acceptable to continue following the procedure. This acceptability will, in part, depend on the conflict that may appear between the recommended action and the operator's knowledge. But the fact that the procedure causes the operator to act against his/her knowledge—to put it aside—can lead to a loss of credibility in the procedure or encourage the operator to adopt a passive attitude toward the procedure and result in the negative effects presented above.

11.2.3. The EPR project: the move toward increased automation

A) Design principles

As for the N4 series, EDF decided to fit the EPR control room with a digital instrumentation and control system driven by a computerized man-machine interface. A goal of the project from the very outset was to increase the level of automation in particular to help to reduce the number of actions to be performed by the operating crew, lighten their workload, and establish lines of defense regarding inappropriate actions which could be taken by the crew.

This automation applied to the start-up and shutdown of equipment (e.g. automatic connection of the residual heat removal system [RHRS] during normal operation), the introduction of new regulations (e.g. for the auxiliary feedwater supply of the steam generators in shutdown states, the supply of borated water) and of new operator support functions intended to introduce automatic corrective actions to avoid activating the reactor protection system. In addition, an automatic reactor state diagnostics system was introduced to aid in emergency operation. Using the main parameters for the reactor state and the primary and secondary systems, this facility-monitoring system suggests the operating strategy to be implemented by the operating crew during accidental situations.

In light of these trends, IPSN began conducting a literature review in 1995 in order to make a status of research on human-automation interaction. This work made it possible to characterize two approaches to automation. In what is referred to as the technician-centric approach, automation is a solution of choice for reducing operator intervention and thus reduces the risks of human error. Operators thus carry out a set of residual functions that are often imprecise in definition and can be detrimental to the overall effectiveness of the human-machine system. An alternative approach consists in viewing automation as a tool for enhancing operator performance. This means that automated systems must therefore be designed to be compatible with the characteristics of human activity (perception, reasoning, cooperation, interaction, etc.). This activity-centric approach calls into question systematic automation based on the analysis of its negative impacts on this activity.

This literature review also made it possible to identify areas to be examined more closely—the impact of automation on the control of the operation of a process by operating crews, the ability to acquire knowledge through operating experience, modes of cooperation between operating-crew members, the level of trust in automation, the degree of independence, and operator involvement in reactor operation.

B) Study of the impact of automation on operator performance

Between 1996 and 1998, IPSN led a research project and the Halden Man-Machine Laboratory (HAMMLAB) in Norway—which is affiliated with the Halden Reactor Project (see "Focus" below) — to study the impact of automation on operator performance (Figure 11.1). This research was conducted on the laboratory's NOkia Research Simulator (NORS) (PWR version of the VVER 440).

Figure 11.1 The facility at HAMMLAB during IPSN's experimental work. © DR.

The purpose of the research project was to understand how operators integrate the effects of automated actions when they manage a situation and must anticipate its progression. An experiment taking account the type of automation (extended vs. limited) and type of operations activity (diagnostic or operating sequence) was conducted at HAMMLAB to answer this question. Diagnostics situations were characterized by their gradual deterioration, with operators having a certain latitude and having to foresee future changes. The operating sequences had to be performed either manually or automatically in order to achieve a change of state. Six scenarios lasting around one hour were acted out by six operating crews.

Several independent variables were measured during the tests for quantitative analysis purposes. These included:

- measuring the overall performance (crew & automated systems) based on the achievement of some specific reactor parameter values;
- measuring operator performance during time windows predefined for each scenario and which ranged from the occurrence of a failure signal to the completion of a response action by the operating crew;
- measuring situation awareness through the operators' responses to a questionnaire about the values of certain parameters. The questionnaire was given several times during a test while the simulator was in pause mode;
- measuring trust in the automated systems *via* a questionnaire at the end of each test.

This analysis yielded a number of findings, such as:

- during diagnostics situations, operators tend to place more trust in automated systems when they have a limited scope of action;

– during operating sequence situations, extensive automation increases overall performance and reduces operator workload;
– human intervention remains preferable in complex diagnostic situations, for in this case extensive automation does not lead to good overall performance.

#FOCUS

The Halden Reactor Project

The Halden Reactor Project was created in 1958 under the joint leadership of the OECD and the NEA (www.oecd-nea.org/jointproj/halden.html). It is being carried out at the Institute for Energy Technology (IFE), in Norway, and is supported by 19 countries, each which funds research in areas such as nuclear fuel, the behavior of materials in nuclear environments, organizational and human factors, human-machine interfaces, etc. Some of this work is conducted by making direct use of a small 20 MW experimental reactor (the Halden reactor is a boiling heavy water reactor—see Figure 11.2) that regularly contains more than 30 experimental devices at the same time. Research on organizational and human factors is based on simulations conducted within the Halden Man-Machine Laboratory (HAMMLAB).

Figure 11.2 View of the Halden reactor hall. © IFE (Institutt for energiteknikk).

11.2.4. Contributions and outlook

The knowledge acquired through this research constituted as many "points of attention" for IPSN's expertises. It was therefore considered that EDF's decision to create a "safety and radiation protection engineer" position would make it possible to limit the focus effects, and even lock-in effects, created by unexpected situations and the importance of which has been highlighted by the studies. In another example, IPSN considered that including coordination points in the accident procedures should offset the risk of the operating crew being split up during unexpected high-stakes situations. Likewise, the results of the study on use of computerized procedures were used by IRSN in its assessment of the EPR control room, in particular to emphasize the importance of providing operators with an overview of the current procedure in order to allow them to understand the operating strategy.

Given the constantly evolving technologies that provide operating crews with new functionalities, this research on the operational activities of reactors shall be continued. One example is the development of embedded systems that will allow field workers to locally obtain multiple data on the status of systems and provide control room operators with much more accurate information about real field conditions.

They must also be continued to derive lessons from the Fukushima Daiichi accident. IRSN thus conducted an in-depth investigation of all of the official reports and accounts that were published about the Fukushima accident. It published its findings in a report titled "A Human and Organizational Factors Perspective on the Fukushima Nuclear Accident" [14]. This report highlights a number of consequences of the total loss of power to the control room: *"At 3:37 p.m., the control room [of reactors 1 and 2] lost all electrical power and was suddenly plunged into silence and darkness, leaving the operators to use flashlights to read the emergency procedures. These procedures were of no assistance in managing the nuclear reactor, however, as the indicators used to monitor its operation were out of action. It became impossible to check the parameters essential in cooling the reactor: the water level and the vessel and containment pressures."* This study has also made it possible to identify issues that deserve closer examination. For example, what knowledge should be used to collect and interpret information on the state of the systems when indicators and procedures are no longer operational following total loss of power to the control room? How can real cooperation be maintained between the control room and emergency-response center once means of communication are no longer operational?

11.3. Studies and research on the organization and management of safety at EDF's NPPs

It was not until the early 1990s that IPSN conducted its first study on the organization and operation of reactors (human factors and organization of unit outages[240], see

240. Term used to designate reactor shutdowns during which the reactor is reloaded with fuel assemblies, maintenance is carried out, and other operations are performed.

Section 11.3.1 below). Previous studies had only focused on operating crews and control room design. Although these organizational studies were prompted by the emergence of issues on the safety of maintenance operations, they also benefited from research conducted by U.S. sociologists on organizational reliability in the wake of the TMI accident.

In 1984, a book with the provocative title of "Normal Accidents: Living with High-Risk Technologies" [15] was published. Providing an in-depth analysis of several industrial accidents, including TMI, it uses the term "normal accident" to emphasize accidents that are related to the very nature of high-risk systems. Adding protections to the systems in fact increases their complexity, which increases their possibility of failure, reduces the ability of operators to understand their overall operation, and in turn increases their vulnerability. Furthermore, the components of these complex organizations are tightly coupled, meaning that what affects one component may affect its related components as well.

Following the publication of *Normal Accidents, living with high risk technology,* some researchers considered that the viewpoint presented in it was not borne out by actual accidents affecting high-risk systems, which were extremely rare. They thus sought to understand how high-risk industries were able to maintain a high level of reliability. The movement known as High Reliability Organizations (HRO) was created by a group of researchers at the University of California, Berkeley[241]. Their work explains the organizational measures and methods implemented daily by HROs to maintain a high degree of reliability. It emphasizes the positive role of organizational flexibility, i.e. rallying several levels of operation in the organization depending on the situation, the redundancy of checks by people, strong consistency between the organization's goals and individual goals, recognition of the importance of the skills of people whatever their level in the hierarchy, and the continuous organization of training and refresher training courses, the co-existence of centralized decision-making and decentralized operational decision-making.

This research remained relatively unknown in France in the 1980s. It was not until a 1999 thesis titled *Le nucléaire à l'épreuve de l'organisation* [16], that information about the HRO paradigm spread. This thesis also emphasized the paradigm's limits and proposed new developments.

Another work of interest is Diane Vaughan's "The Challenger Launch Decision—Risky Technology, Culture, & Deviance at NASA" [17]. Published in 1996, it explores the reasons behind the Challenger accident and analyzes several aspects of the organization at various time spans. It thus looks at the interactions between engineers in the hours preceding the launch of the shuttle as well as the relationship dynamics between NASA and the U.S. Government over several years. By focusing on the micro-decisions made at the time, Vaughan shows that what in hindsight appears to be a series of clearly identifiable errors is actually a series of decisions and interpretations that are perfectly understandable in the context in which they were made, but which are in fact slight

241. Todd LaPorte, Karlene Roberts, and Gene Rochlin, joined by Paul Schulman and Karl Weick in particular.

deviations from normal limits and lead imperceptibly to "normalization of deviance". This is one of the key contributions of this book.

Also worth mentioning is the PhD thesis "When designers anticipate the organization to control risk: two installation modification projects on two SEVESO 2 classified sites" completed in 2008 [18]. This thesis, which contains many sociological lessons that cannot all be listed herein, brought to light or confirmed a few of the "workings" that form the basis of organizational reliability:

- the involvement, right from the design of facility retrofits, of operators (in all the relevant fields) and contractors (such as maintenance subcontractors) who will all ultimately participate in the operation of the retrofitted facilities, is a factor of success;
- corrections or additions to the initial design of the facilities is a factor of risks in complex systems (as is a nuclear reactor and its operation).

All of this work will influence the research carried out by IPSN (then IRSN) detailed hereafter.

11.3.1. Organization of maintenance activities

In the summer of 1989, three incidents occurred while maintenance was being conducted on reactors in France's NPPs. These incidents highlighted the fact that safety could be compromised by a forgotten or incorrectly basic operation (gaseous releases beyond the threshold, a latent failure that could have caused an engineered safeguards system to fail during an accident sequence). EDF reinforced its organization starting in 1990. It created new positions (checkers and contract managers), reinforced site engineering and preparation, provided contractors with training on quality and safety, and introduced the performance of a risk analysis prior to all maintenance activities. Although these arrangements may, in principle, reinforce the reliability of maintenance activities, IPSN and EDF agree that a deeper understanding of the conditions of unit outages was necessary to be able to better assess the relevance of these measures.

A) "Human factors and the organization of unit outages: safety challenges"

In 1991, IPSN undertook a study to understand the maintenance work in order to be able to assess the organizational changes proposed by the nuclear reactors operator. An ergonomic analysis of the activities of various workers during two unit outages—in July 1992 and July 1993—on the same site was implemented. In concrete terms, more than 20 projects (essentially related to mechanical work) and the activities of more than 50 workers with various positions were closely tracked. Two requalification tests, among the most complex, of the safety-injection system items were observed. An ergonomist took part in the daily outage meetings during which progress on the maintenance work was checked and rescheduled where necessary. This ergonomic analysis revealed the specific aspects of each worker's job, its specific constraints, and the associated risks, and described how the workers coped with contingencies in their established organization.

Both of these aspects made it possible to identify possibilities of organizational failure and failure in individual work, as well as the implicit strategies and skills developed by workers to mitigate these risks of failure.

The study revealed that **the conditions under which maintenance was performed were insufficiently taken into account** during the maintenance-preparation activities and when drawing up operating experience feedback on unit outages. The little interaction between contractors and preparers was identified as a contributing factor to this situation. Information about the difficulties in carrying out maintenance work did not rise, despite the fact that these difficulties led to unexpected events, lost time, and workers making decisions themselves. All this had a human "cost" and could compromise safety.

The management of unexpected events is a primary concern in the life of workers. Nothing happens exactly as planned and unforeseen events can crop up. When they occur, they must be detected and maintenance work must be adjusted accordingly. Their impact on the safety of this readjusted maintenance work must be assessed in real time. The time and resources needed to do this are often underestimated.

The study also revealed that jobs and skills of diverse nature are involved in the maintenance activities performed during a unit outage. The **comparison of viewpoints** in real time, when organized, helps to ensure maintenance activities are carried out smoothly. With this in mind, skills management and time constraints, attention to collaboration among various maintenance workers, and the creation of structures where limits encountered during maintenance can be openly discussed are elements important to safety.

B) "The organization of unit outages in a period of reform"

In 1994, a study by the Center for the Sociology of Organizations (CSO) was carried out in French NPPs as part of a PhD thesis being written on the organization of work in high-risk systems. Titled "The organization of unit outages in a period of reform," the thesis subsequently found an echo in the 1999 thesis [16] referred to herein. It provided a comparative analysis of the preparation and completion of four unit outages at two NPPs in the United States and two NPPs in France.

The main objective of the thesis was to assess changes initiated by EDF, starting in 1991, to reinforce the organization of its maintenance activities during unit outages. It sought to identify the difficulties encountered during outages. However, it exceeded and renewed the scope of analysis of the previous study (see Section A). On the one hand, it attempted to understand how a number of outage constraints occur during the lead-up to the outage itself. On the other, it sought to understand how some characteristics of the intervention situations are related to overall constraints that burden nuclear sites. In terms of the methodology, the thesis made use of strategic analysis, developed in the sociology of organizations (see, for example, the work referred to herein [19]). It used data collected from 68 interviews to trace back to the "games" that take place between the members of an organization and the strategies that they use to perform their activities and achieve their goals.

The thesis revealed that **setting up a project organization** ("permanent structure of a unit outage") dedicated to the completion of unit outages facilitates the preparation of maintenance operations. However, if the project team is formed around six months before an outage, each department gradually assigns its representatives and some people, such as inspectors, are assigned late in the process. Furthermore, the people assigned to the project team are chosen by their respective departments. Outage project managers therefore do not choose their "troops." The thesis also showed that the "ramping up" of the outage structure was accompanied by a withdrawal of the departments in the preparation of outages. Broadly speaking, the departments did not appear concerned with outage projects, considering that they were solely a matter for outage workers.

Another prominent change observed on the site was **how work is awarded to contractors**. Buyers participate in drawing up work orders and business aspects (the definition of technical requirements, competitive tendering, etc.) play a larger role. The level of detail required in specifications is a source of difficulty for EDF staff, especially for those who lack experience or deal with operations for which it is difficult to anticipate the working conditions. Furthermore, although competitive tendering can help contain maintenance costs, it creates contractor instability and the possibility of having to work with new contractors whose technical expertise remains to be seen over time.

The study also revealed the complexity in the **scheduling of outage work**. A first draft of the master outage schedule is established based on aspects such as the preventive maintenance programs, the retrofits to be made, and constraints related to the provision of tools. The coordinators then define the work packages and send the corresponding orders to contractors. This preparation essentially takes place for each individual package with relatively little overall coordination. The result is that cross-disciplinary activities, such as the procurement of spare parts or routine radiation protection equipment, are insufficiently taken into account.

C) Study on the implementation of EDF's outage control centers

Since 2012, EDF has been implementing COPAT (*Centres opérationnels de pilotage des arrêts de tranche*), its own version of the outage control centers (OCC) in use in North America. IRSN conducted a benchmark study in June 2011 to assess the impact of this new organization of safety and radiation protection. It did so by consulting licensees in the United States and Canada as well as the Canadian Nuclear Safety Commission (CNSC). Prior to this, EDF had conducted its own study in 2007.

EDF's COPATs are designed to coordinate outage activities 24 hours a day to ensure that planned outage schedules are better adhered to. Sensitive phases and sequences of critical activities, which are identified beforehand, are closely monitored. The COPAT is alerted of probable or confirmed deviations of 30 minutes in the critical path in order to take the appropriate arrangements without affecting outages. This assistance makes it possible to solve difficulties encountered during outages more quickly and share real-time information among teams. This organization requires having quick response teams. Its aim is to be able to anticipate potential difficulties and imagine scenarios for managing them by foreseeing the sufficient necessary resources.

IRSN's study made it possible to capitalize the facts, observations, and lessons, particularly regarding the key factors of success, the limits, and the challenges of managing unit outages via outage control centers. Its objective was more to identify key issues than collect organizational solutions. It thus helped to establish a method for assessing EDF's proposal for implementing COPATs. Although the study validated a number of topics for in-depth study that had been identified previously by IRSN (complexity of the management of interfaces between hundreds of people and within projects comprising thousands of activities, forecasting of human resources in a period of massive retirement, integration of hundreds of contractors, operating experience feedback, change management, etc.), it revealed other important topics to be investigated in detail during IRSN's assessment of the measures adopted by EDF.

The interviews conducted during the study revealed that OCCs were seen as structures that were added to solve problems of coordination and cooperation between operating crews and maintenance crews and bridge cultural gaps. It was thus deemed wise to investigate how COPATs could increase the complexity of unit outage management.

The study also showed the variety of project structures in North America that adhere to OCC principles, have different staffing levels (10 to 20 people), and place various levels of importance of operation and maintenance. The question was thus raised about whether COPATs should be tailored to each French NPP.

It was noted that North American facilities placed importance on preparation (longer by a few months and standardized) and controlling the forecast volume of maintenance. Licensees in North America tolerate only 10% additional unexpected[242] maintenance over the initial volume determined six months prior to the start of an outage. In the case of France's NPPs, this additional maintenance can rise to 100% (50% during the six months prior the outage and 50 % during the outage). This observation raised a number of questions:

- Was there a lack of foresight detrimental to the organization of the preparation of maintenance activities in France's NPPs?
- Was the restriction of unexpected maintenance by North American licensees conducive to organizational success but detrimental to equipement reliability and safety?

The study found that OCC staff had a great amount of expertise (three years as an assistant in an OCC, then eight years in that position) that enabled them to calmly deal with situations, particularly with unexpected events. The situation in Canada and France was vastly different. It was found that COPATs could not be training centers, which related to issues of human resource management.

242. The term unexpected maintenance work encompasses both that which may be overlooked during the inventory made six months before the outage and that which is caused by anomalies found during the six months before or during the outage.

The study also showed that there was a relationship between foresight and responsiveness at licensees in North America. There was a high priority on foreseeing potential failures and scheduling slippages and on being prepared to manage potential unexpected events (capitalization of past unexpected events managed, prepared crews to face similar events). This aspect had to be looked at in the case of EDF's COPATs.

Lastly, the study revealed that North American licensees (30%) significantly outsourced fewer maintenance activities than EDF (80%), pooling of resources among sites, and greater internalization of logistical means (ensuring that maintenance activities are carried out under the right conditions).

11.3.2. Contractor management

The studies conducted between 1992 and 1994 (Sections 11.3.1 A and B) chiefly focused on the organization and performance of maintenance activities during unit outages. In the early 2000s, IPSN decided to conduct research on the use of contractors. There were two reasons for this: contractors are increasingly being used, in particular for maintenance activities, and outsourcing in the nuclear industry is a subject of intense debate that, in the end, has little basis on extensive studies.

A) Outsourcing relationships and their consequences on security and safety

Some research emphasizes the difficult working conditions of some employees of contractors and directly link these difficulties to their subcontractor status. The use of contractors is thus seen as a factor that deteriorates working conditions and even the very quality of work and thus safety. However, some arguments cast doubt on this negative effect of outsourcing. For example, some types of sensitive equipment (pumps, valves, etc.) have always been maintained by the employees of their manufacturers ever since the first NPPs were commissioned. This situation is widely viewed as an assurance of quality. Likewise, some maintenance tasks are extremely difficult regardless of which employee is performing them. In order to more clearly identify the effects of outsourcing, the research focused on the outsourcing relationship, how this relationship is built through the joint action of the project owner and the contractor, and how this relationship affects not just the work of contractors' employees, but that of the project owner's employees as well.

Such research was conducted as part of a sociology PhD thesis in collaboration with the French rail operator SNCF and the French Gas, Network management, Distribution Company GrDF [20]. This partnership made it possible to access maintenance projects to replace railroad tracks and track ballast (SNCF) and to replace gas-distribution networks (GrDF). These projects were carried out mainly by the employees of contractors. More than 50 interviews were conducted with the local, regional, and national employees of SNCF and GrDF. These employees were in charge of preparing and conducting the maintenance projects and providing feedback on them. These interviews made it possible to collect data on the meaning each employee ascribed to their work practices, how they saw their work, and the ties linking them to the contractors. Several projects were also observed as they took place. For example, a project to replace track ballast and railway ties in Northern France (November 2007 to January 2008) was followed, making it possible to attend initial project planning and scheduling meetings, follow an employee in charge of bringing in and clearing out the work trains on the railtracks, and more.

Although the findings yielded by this research are too numerous to mention here, it provided some important observations and important lessons.

First of all, this research made it possible to provide a typology of the various types of outsourcing activities[243], including in the case of insourcing by a contractor. Aspects of "cascading" outsourcing were also discussed in this research.

The research also revealed that there are many types of relationships between project owners and contractors and that they depend in particular on the extent of outsourcing, how much technical skills are shared between project owners and contractors, their degree of mutual dependence, and the duration of outsourcing agreements. The 2003 collapse of the gangway leading to the Queen Mary 2 liner at the Saint-Nazaire shipyard is cited as an example highlighting a number of aspects of the relationships between the project owner (Les Chantiers de l'Atlantique) and its contractor (SAS Endel) that jeopardized safety at the shipyard and ultimately led to the accident[244].

The thesis showed how much the premise that "*the market' quality, cost, and turnaround time results—always better than those obtained internally—make it easy to overlook the transaction costs required to remain in control of facilities*". Indeed, outsourcing often leads to "virtualization" of safety or security where hands-on monitoring may be severely decreased (or even prohibited) in favor of supervision based on the validation of technical studies and the verification of data used by contractors with reduced hands-on checks ("paper-based" safety or security). "Cascading" outsourcing can compound this situation and cause "*risk to migrate to the weakest links in the outsourcing chain*."

The thesis furthermore emphasized that outsourcing affects the project owner's internal organization, thus challenging the "*idea that outsourcing is simply a "transfer" of operations to the contractor without any deep-rooted changes to the organization in which the change occurs*."

Another aspect highlighted is that "*it is impossible to study the link between outsourcing and safety or security with a rationalistic vision consisting in identifying factors that would directly affect safety or security and lead to the conclusion that managerial intervention on such-and-such factor would, with such-and-such percentage of odds, guarantee safety or security*". The thesis shows the oversimplification of the outsourcing "approach" from the sole perspective of the "sustainability of the skills" of the project owner and "contractor monitoring." Outsourcing must be considered and analyzed as a relationship or partnership.

243. Technology-driven or capacity-driven insourcing, technology-driven or capacity-driven external subcontracting, and outsourcing.
244. These factors were clearly established by the criminal investigation into the accident.

The thesis also pointed out that "established" contractors—thanks to their acquired (and well-earned) reputation—can become elevated to favored status: "*contractors who know that they are well positioned create a lot of leeway for themselves*".

These findings and lessons are points of attention in IRSN's assessments of outsourcing practices used by licensees, in particular those conducted on EDF in 2015. They also opened the way to other tracks of research, such as that carried out as part of RESOH research chair.

B) Outsourcing relationships—the RESOH research chair

Inter-organizational relationships, such as outsourcing, were seen as a source of advances in safety following the progress made in human and organizational factors of safety since the 1980s, a fact attested as much by the conclusions of the complementary safety evaluation (ECS) conducted by IRSN and ASN following the Fukushima accident. The Chair of Safety, Organization, and Human Research (RESOH) is dedicated to studying organizational and human factors of safety at high-risk industrial facilities throughout their life cycle (design to dismantling and waste management). It focuses in particular on these inter-organizational relationships.

RESOH was created in March 2012 by its four founding partners—the École des mines de Nantes, IRSN, AREVA, and DCNS—for a five years duration.

The RESOH research chair focuses on two areas of research:

- inter-organizational relationships, in particular *via* the development of outsourcing and co-contracting relationships[245];
- the integration of safety in all the constraints and management systems related to research on industrial competitiveness.

Safety is a collective "construct" that brings into play the liability and individual activity of not just each contractor, but of teams and institutions as a whole as well (project owners, contractors and co-contractors, inspectors, etc.). RESOH's objective is to analyze the construction of safety at the scale of the system formed by the complex relationships between all of these parties and by taking their environment (economic, legal, social, etc.) into account.

Its aim is to provide answers to these issues by identifying the vulnerabilities, robustness and resilience[246] of these contracting and co-contracting networks and by studying the managerial practices and management systems that can enhance safety within them.

RESOH uses two research methods:

- qualitative field studies that call on techniques used in the sociology of work and organizational ethnography;

245. This term describes the pooling of contractors (consortium) for the purposes of providing service for a licensee.
246. Adaptability to unexpected situations.

– an observatory on outsourcing practices, for quantitative and cross-disciplinary purposes.

RESOH aspires to bring in additional partners and thus explore other areas of industry.

Its research on the management of complex projects and the use of outsourcing covers two aspects:

– the contribution of unit outage schedules to improving time management and coordination between parties;

– the role agreements play in mutual commitments between project owners and contractors.

This research is grounded in field studies conducted with AREVA (La Hague) and DCNS (Cherbourg).

11.4. Research on human and social sciences: the AGORAS project

In his introductory message to official report by the Fukushima Nuclear Accident Independent Investigation Commission commissioned by the National Diet of Japan, Chairman Kiyoshi Kurokawa mentioned a number of societal factors "*that could and should have been foreseen and prevented*". He wrote of "*mindset that supported the negligence behind this disaster*" *(...)* "*made in Japan*" Chairman Kurokawa's reflections have a general scope and should be of concern to any party involved in the "governance" of risks of any type. It was in this field of human and social sciences that IRSN decided, in 2012, to conduct new research, in particular as part of the AGORAS[247] project. Rather than investigate the human and organizational factors that contribute to the operational safety of nuclear facilities, it focused on the general functioning of all stakeholders involved in ensuring safety.

In France, the complementary safety evaluation (ECS) conducted in wake of the Fukushima accident pointed up the importance of studying the link between safety and inter-organizational relationships. It was this as yet little-explored topic that became the focus of the AGORAS project to improve governance of nuclear safety organizations and networks. AGORAS was selected in late 2013 as part of a call for nuclear safety and radiation protection projects launched by the French National Research Agency (ANR) and spurred by the highest levels of France's Government. Its goal is to understand how institutional equilibriums among licensees, subcontractors, partners, and also safety regulators and public assessment bodies are built and evolve in a world where civil society is playing an increasingly greater role. Emphasis will be placed on the dialog that is established between stakeholders and which partially forms the basis of safety in the nuclear industry as a whole.

247. *Amélioration de la gouvernance des organisations et des réseaux d'acteurs pour la sûreté nucléaire*.

Expected to last six years, AGORAS is structured around two aspects—accident prevention and emergency management. The goal of the first aspect is to analyze the impact of the Fukushima accident on the safety "approach" for nuclear facilities and the relationships among those involved in the governance of nuclear risks. The goal of the second aspect is to analyze how this accident has helped to change the perception of nuclear accidents and the methods used to manage the preparation of accident and post-accident situations. It will look at changes in emergency-response organizations and doctrines since the TMI accident by analyzing feedback obtained for emergency exercises and actual emergencies (the flooding of the Blayais NPP in late 1999 is a possible example). The expected results should reveal vulnerability factors related to the growing complexity among emergency-response teams and institutions, tracks for improvement and opportunities making simulation exercises more realistic.

IRSN is coordinating two efforts as part of the project. The first (2014–2018) will identify organizational and cultural conditions that influenced decisions about the Fukushima NPP (technical choices, design options, etc.) and which ultimately proved to be inappropriate. This effort comprises two areas of research:

- the first, the study of assessment processes and technical decisions, consists in understanding, as part of a socio-historical analysis developed to analyze a number of major accidents (such as the Challenger Shuttle disaster), the dynamics and factors that may explain certain concepts, tools, and data that are poorly known, ignored, or inadequately taken into account by all stakeholders involved in the governance of nuclear risks. It is based on an analysis of French cases;
- the second consists in analyzing the preparation and implementation of risk-control documents, such as the "extreme cold weather baseline" or the "flood guide" (see Section 8.2 herein). The aim is to analyze the dynamics and representations at work during their preparation and how they are able to either strengthen or weaken the legitimacy of the documents. How are some proposed tools, data, recommendations, or approaches interpreted and used? Which deviations or amendments in relation to the initial intentions of the designers of these documents are ultimately accepted by regulators? What are the limits, checks and impact assessments?

The second action (2013–2018) relates to technical dialog during the post-accident complementary safety evaluation (ECS) to understand how the occurrence of a major accident can lead to a reassessment of previous safety practices. This involves identifying the conditions that both favor and impede reassessment. Complementary safety evaluation (ECS) has effectively opened a period of intense technical dialog among licensees, ASN and IRSN. In concrete terms, the goal will be to answer the following questions: What new inter-organizational dynamics will be brought to light by the lines of reasoning and potential controversies among the stakeholders? Which "paradigm breaks" (the unforeseen accident has happened) or, on the contrary, what lines of "defense" will underpin these lines of reasoning? How will media coverage of some debates and the particularity of extending the service life of NPPs affect the positions and arguments of key players of nuclear safety?

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Chapter 12
Other Research and Tracks of New Research

IRSN also conducts research of varying scope in many other technical areas related to PWR safety, including:

- the reliability of critical codes in the code-based digital instrumentation and control systems used in N4 series and EPR reactors;
- the behavior of materials subjected to seismic loads with the proprietary development, *via* numerical simulation, of material "fragility" curves that can be implemented in probabilistic assessments of seismic hazards. Determination of fragility curves *via* numerical simulation is an alternative to the American method (EPRI curves based on a lognormal model and predetermined parameter sets) and could also make it possible to determine in a most suitable way the parameters of EPRI's lognormal model. To achieve this, IRSN developed Interactive Seismic Analysis of Fragilities of Equipment and Structures (ISAFES), a simulation code that makes it possible to conduct dynamic time simulations using representation objects of structural components referred to in France as MARC (Weight, Damper, Stiffness, Criterion). This development thus addresses the fields of dynamic numerical simulation of equipment with uncertainties as well as soil-structure interaction with seismic excitation;
- in-vessel mixing phenomena in PWRs, which could occur, for example, if borated water is diluted with clear water, a situation that can lead to reactivity accidents;

– airborne contamination. This research, apart from that on filtration systems, is more specifically geared to the issues encountered in fuel cycle facilities and dismantling operations.

Furthermore, the designers of the Generation III[248] and III+ reactors, as well as of projects for integral or modular reactors (SMR[249]), are increasingly emphasizing passive systems[250]. There has been renewed interest in passive systems since the Fukushima Daiichi accident. Needless to say, the accident's consequences led researchers to look at prolonged loss of electrical power and LOCA. Passive systems may be solutions for handling such situations. However, the increasing use of passive systems reveals that more needs to be known about their use in deterministic safety demonstrations and PSAs. Indeed, questions remain to be answered about their reliability, their effectiveness, the possibility of "recovery" in case of failure, and the verification of their performance from reactor design to reactor operation.

IPSN (and after IRSN) addressed the subject of passive systems during assessments conducted in the 1980s and 1990s on the PHENIX and SUPERPHENIX fast neutron reactors and, more recently, for the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) project. These systems related specifically to the possibility of removing the residual heat from the reactor *via* natural convection of the sodium in the reactor vessel—primary circuit—and in the secondary circuits, which also contained sodium. At the same time, IRSN was involved in the European Thermal-Hydraulics of Innovative Nuclear Systems project (THINS), particularly for the numerical simulation of natural convection tests conducted at PHENIX in the late 1990s (as part of its "end-of-life tests") and for the qualification of the CESAR module of the ASTEC simulation code. However, proposals for research projects (extending beyond sodium-cooled reactors and spanning prestandardization research, testing of components, numerical simulation, etc.) are being prepared following the European Commission's H2020 call for projects. These projects include REPASS[251] and NUSMOR[252], in which IRSN may participate with partners such as ENEA[253], KIT and GRS.

248. The use of passive systems in an EPR is limited to gravity drop of the control rods, pressurized accumulators to reflood the core, the possibility of thermosiphon cooling of the reactor following failure or voluntary shutdown of the primary pumps—to be checked in the reactor itself as well as in the case of reactors in operation—, hydrogen recombiners, and the core catcher in the event of a core melt accident.

249. Small Modular Reactors.

250. The concept of a passive system is broad: spanning systems that make limited used of components requiring mechanical movement to perform the systems' functions, the supply of outside power or support functions, human intervention (to activate and operate the system for the appropriate period), and systems that use natural phenomena (gravity, heat transfer by conduction, natural convection, or radiation, pressure differences, etc.).

251. Reliability Evaluation of Passive Safety Systems.

252. NUgenia Small Modular Reactor with passive safety features.

253. Italian National Agency for New Technologies, Energy and Sustainable Economic Development.

Science and Technology Series

Current State of Research on Pressurized Water Reactor Safety

Jean Couturier, Michel Schwarz

For more than 40 years, IPSN then IRSN has conducted research and development on nuclear safety, specifically concerning pressurized water reactors, which are the reactor type used in France. This publication reports on the progress of this research and development in each area of study – loss-of-coolant accidents, core melt accidents, fires and external hazards, component aging, etc. –, the remaining uncertainties and, in some cases, new measures that should be developed to consolidate the safety of today's reactors and also those of tomorrow. A chapter of this report is also devoted to research into human and organizational factors, and the human and social sciences more generally. All of the work is reviewed in the light of the safety issues raised by feedback from major accidents such as Chernobyl and Fukushima Daiichi, as well as the issues raised by assessments conducted, for example, as part of the ten-year reviews of safety at French nuclear reactors. Finally, through the subjects it discusses, this report illustrates the many partnerships and exchanges forged by IRSN with public, industrial and academic bodies both within Europe and internationally.

This publication reflects IRSN's desire to keep an enduring record of its results and to share its knowledge.

IRSN is a public industrial and commercial body undertaking research and consultancy in the field of nuclear safety and radiation protection. It provides the public authorities with technical support. It also carries out various public service missions entrusted to it under national regulations. In particular, these include radiological monitoring of the environment and workers in France, managing emergency situations, and keeping the public informed. IRSN makes its expertise available to partners and customers both in France and worldwide.

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