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.).

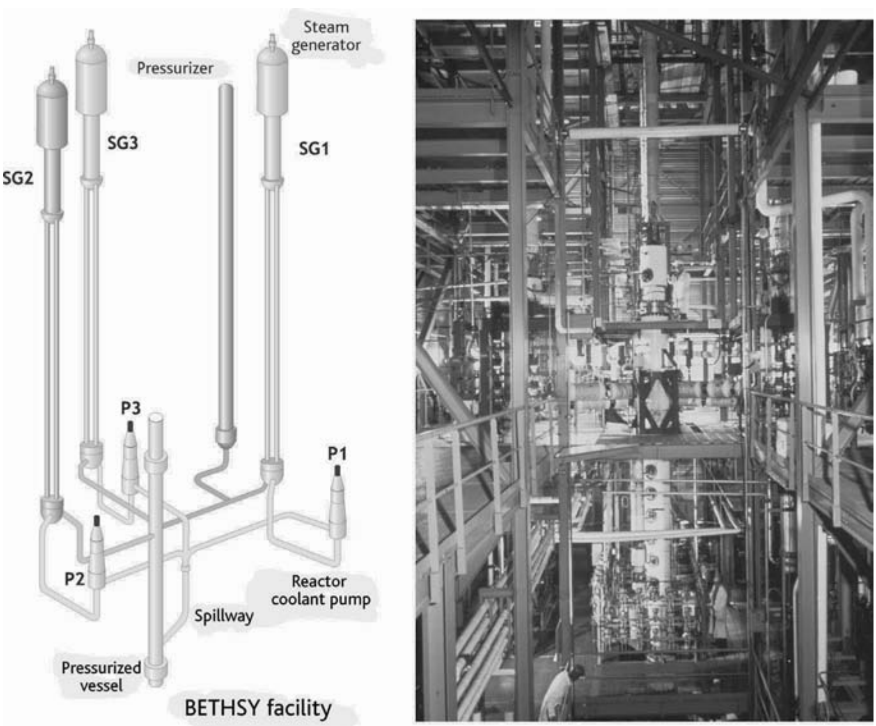


Figure 3.1 The BETHSY loop (900 MWe PWR, three loops). © Georges Goué/IRSN-Source CEA (left), CEA (right).

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).

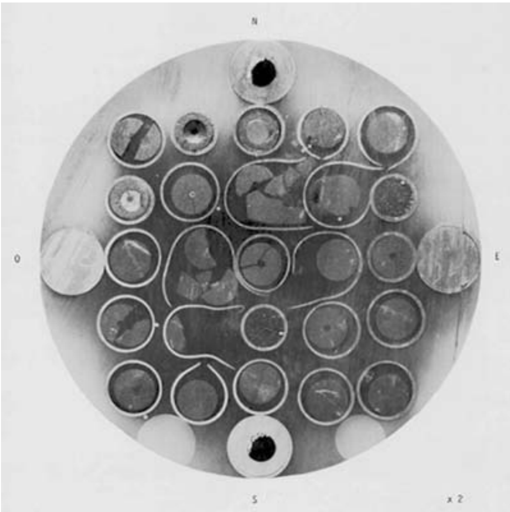
Analysis of all these results showed that the effects of in-service corrosion on the cladding were slight, in terms of both oxidation kinetics and mechanical strength during rod reflooding.

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.

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