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).

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™, and M5™) 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.