Contribution of Research Reactors to the Programmes for Research and Technological Development on the Safety

J. Couturier, F. Pichereau, C. Getrey, J. Papin, B. Clément

Institut de Radioprotection et de Sûreté Nucléaire,
Fontenay-aux-Roses, France

Abstract. The objective of the present article is to illustrate the contribution of the research reactors to the programmes for research and technological development led to increase the safety of the nuclear power plants, more particularly in the field of the of Reactivity Insertion Accidents (RIA), Loss of Coolant Accident (LOCA) and partial or total melting of the core. Indeed, the physical phenomena occurring during a severe accident are extremely complex. Research aims to better understand these phenomena and to reduce the associated uncertainties in order to assess the extent to which the state of acquired knowledge can be used to make reliable predictions. The examples will concern accidents that may occur in Pressurized Water Reactors (PWR) and in Sodium Fast Reactors (SFR).
Some given examples will show the fundamental importance of experiments in research reactors to establish and support the safety demonstrations; indeed, such in-pile tests :

— allow to characterize badly known physical phenomena;
— can put in evidence some not foreseen phenomena;
— contribute *in fine* to the development and in the global validation of the codes used in the safety demonstrations, such as ASTEC (Accident Source Term Evaluation Code) used to simulate an entire accident from the initiating event to the possible radionuclide release outside the containment.

1. Introduction

From the very first, nuclear reactors have been used in a wide range of applications. These include the generation of electricity in 'traditional' power plants using, for example, pressurised water reactors, the production of neutrons for the characterisation of materials, and research into new applications including solutions to the problems of radioactive waste management. Experimental reactors are now essential tools in scientific and technological research, and in support of operational reactors used for electrical power generation.

Experimental reactors may be classified into a number of categories including :

— Reactors used to generate beams of neutrons for scientific purposes. Examples include the high flux reactor (RHF) in Grenoble, the ORPHEE reactor in Saclay, the high flux reactor (HFR) in Petten and the FRM II reactor near Munich.
— Reactors used for technological irradiation, such as the OSIRIS reactor in Saclay and the Jules Horowitz reactor in Cadarache (under construction). These reactors are used to develop and characterise the behaviour of materials (fuel and cladding) when exposed to radiation as part of research into the acceleration of fuel burn-up. They are also occasionally used to characterise the behaviour of fuel during slow power transients, etc.
— Reactors specifically designed for use in safety studies, such as the CABRI, PHEBUS and the (now shut down) SCARABEE reactors in Cadarache.

- Critical mockups and oscillating reactors, together with specific mockups for studying criticality, core physics and fuel. These include the EOLE, MINERVE, and MASURCA systems in Cadarache, and SILENE, CALIBAN, PROSPERO and the B Apparatus in Valduc.
- Demonstration reactors used to validate and qualify designs, systems and equipment such as fuel, heat exchangers and steam generators. These include the RAPSODIE reactor in Cadarache (shut down in 1982) and the sodium-cooled fast neutron PHENIX reactor in Marcoule.

The aim of this paper is to illustrate the contribution of research reactors to research and technological development programmes aimed at increasing the safety of the nuclear power plants, especially in relation to Reactivity Insertion Accidents (RIA), LOss of Coolant Accidents (LOCA) and partial or total core meltdown. Pressurised water and sodium-cooled fast neutron reactors are used as examples.

2. Contributions in relation to pressurised water reactors

2.1. Rod ejection accident

Rod ejection accident is one of accidents occurring in PWR that require specific studies to be carried out at the design stage. This type of accident is initiated by a sudden breach in the housing of the Rod Cluster Control Assembly (RCCA) latch mechanism. The resulting pressure difference forces the ejection of the RCCA. The reactivity in the region of the core adjacent to the RCCA rises extremely rapidly (of the order of a few tens of milliseconds). Within the fuel rods, the increase in power results in sudden heating of the fuel pellets which rapidly expand. In some cases, this can also lead to fission gas release. The expansion of the fuel pellets and the pressure of the gas put the fuel rod cladding under stress. If the cladding fails, very hot fragments of solid fuel may be ejected into the primary coolant circuit, causing a pressure wave to form due to the vaporisation of the coolant. This type of phenomenon is especially likely with finely fragmented high burn-up UO_2 fuel or with MOX fuel. The studies associated with this type of accident are mainly aimed at guaranteeing sufficient core cooling and the integrity of the second containment barrier, i.e. the reactor vessel and the main primary coolant circuit. The decoupling criterion associated to this objective was initialy expressed in terms of maximum enthalpy reached based on the results of the SPERT tests carried out in the USA on fresh and lightly-irradiated fuel.

The first experiments on heavily-irradiated fuel, carried out in the 1990s using the sodium loop of the CABRI reactor in France and the NSRR[1] reactor in Japan revealed a range of unexpected behaviours, highlighting the importance and relevance of these integral tests. Since then, considerable modelling work has been carried out on the interpretation of these tests with the aim of understanding the underlying physical mechanisms within a high burn-up fuel rod during a RCCA ejection accident.

In particular, fourteen reactivity insertion tests have been carried out using the CABRI reactor in Cadarache since 1993. These tests were carried out on sections of commercial fuel rods that had previously been irradiated in PWR. The first test, known as CABRI REP-Na1, was carried out on a highly corroded fuel rod of an old design. During this test, some of the fuel was ejected out of the rod at very low enthalpy by comparison to the peak enthalpy wich could be reaches during a RCCA ejection accident occurring in a PWR. The results of this test demonstrated the importance of being able to guarantee the correct mechanical properties of the cladding. The adhesion of the zirconium oxide layer resulting from the in-reactor oxidation of the cladding is particularly important, given that the thickness of this layer, and hence the clad fragility, increases with the time spent in the reactor. The REP-Na7 test, carried out on a MOX fuel rod irradiated at 55 GWd/tU, showed the effects of a fission gas release during a reactivity insertion accident on the ejection of finely dispersed fuel following rupture of the cladding. The REP-Na11 and CIP02 tests, carried out on AREVA designed

[1] Nuclear Safety Research Reactor.

fuel rods with an M5^{TM} cladding (an 1% niobium zirconium alloy) demonstrated the excellent performance of this type of cladding at high burn-up. The long-terme aim is to establish a new and more physical safety criteria that are applicable at all burn-up rates. In the meantime, the french utility : *Electricité de France* safety demonstration for high burn-up fuels is based on decoupling criteria derived directly from the interpreted results of tests carried out on the sodium loop of the CABRI reactor in Cadarache. These decoupling criteria are expressed in terms of burn-up, corrosion thickness, pulse, enthalpy deposit, and maximum cladding temperature.

The water loop currently under construction as part of the renovation of the CABRI installation will enable test results to be obtained in 2010 that are more representative of actual reactor situations. These will include the thermal coupling between the fuel rod and the primary coolant and will be used to consolidate the safety criteria and to define criteria for new fuel products, especially those intended for use by *Electricité de France* in the EPR reactor.

2.2. Loss of coolant accident (LOCA)

The Loss of Coolant Accident is another of the accidents considered during the design of the PWR. The associated safety requirement is that it should be possible to guarantee both the short term and long term cooling of the core. This type of accident may be caused by a breach in one of the pipes carrying the main primary coolant and initial effects include the drying out of the fuel rods and a transfer of stored energy from the fuel pellets to the cladding. During this initial phase, the cladding of the fuel rods is forced outwards by the increase in internal temperature and pressure. This reduces the effective cross section of the channels through the core, reducing the available cooling. Initialy, This ballooning of the fuel rods was studied in tests on new and lightly-irradiated fuel. The results of these tests are summarised in US nuclear safety authority NRC document NUREG 630. An R&D programme is currently being developed with the aim of quantifying this reduction in the case of highly-irradiated fuel rod assemblies. This may include tests on fuel rods in the CEA[2] PHEBUS reactor in Cadarache. If the cladding actually bursts, fuel relocalisation in the the ballooned area may occur resulting in a local rise in the heat flux due to the residual power in the fuel. This will then result in a rise in temperature adjacent to the ballooning.

The cladding may become very fragile at temperatures exceeding 850 °C during a loss of coolant accident due to oxidation by water vapour and the absorption and resulting diffusion of hydrogen in the metal. If the oxidation of the cladding is severe, the resulting embrittlement may cause the cladding to fragment, especially during the thermal shock occurring when the core is flooded by the safety injection system. The resulting loss of core geometry may also adversely affect cooling.

Fragmentation of the cladding may be avoided by ensuring compliance with criteria relating to the maximum cladding temperature and maximum level of corrosion. These criteria, set by the NRC in 1973, are currently applied in France.

The phenomena of ballooning and resulting migration of fuel into the ballooned areas (Figure 1) were demonstrated during the PBF-LOC, FR2 and FLASH-5 tests carried out in the experimental PBF reactor in the USA, FR2 in Germany and the CEA SILOE reactor in Grenoble respectively. In addition, a specific test programme has been running since 2005 at the Halden experimental reactor in Norway with the aim of improving the characterisation of the migration and hydridation phenomena adjacent to the ballooned areas. These tests are mainly concerned with fuel rods that have previously been irradiated in power reactors at high burn-up. One of the tests on a highly-irradiated fuel rod (the IFA 650-4 test) has shown a considerable migration of finely divided fuel in the ballooned area and cooling channel despite the chemical adhesion of the pellets and the cladding.

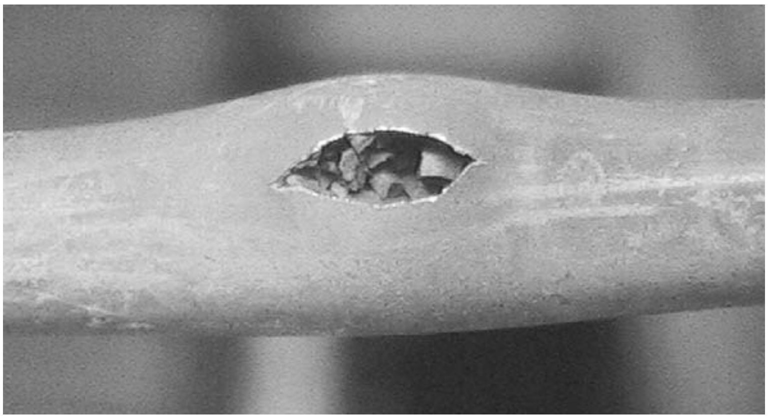
---

[2] Commissariat à l’Energie Atomique

These experimental reactor tests will provide useful data for the LOCA safety standards revision programme currently in progress in the USA and in France. In France, this revision will be discussed at two meetings of the Reactor Standing Group planned for 2008 and 2010.

FIG. 1.

Ballooning and relocalisation of the fuel in the ballooned area during the ICL#2 integral test carried out at the Argonne National Laboratory (USA)



2.3. *Pellet-cladding interaction (PCI)*

Accidents involving sudden load increases, uncontrolled withdrawal of control rods under power, control rod falls and uncontrolled dilution all lead to core power surges lasting between a few tens of seconds and several minutes. The safety requirement associated with these types of accident is the maintenance of the integrity of the first barrier, i.e. the fuel rod cladding. This is achieved by a reactor protection system which limits the power surges, and by the application of the technical operating specifications. Within the fuel rod, the power increase not only carries the risk of a loss of water in the core cooling channels when the critical flux is reached, but also results in mechanical deformation of the cladding due to thermal expansion of the fuel pellets. In this case, this deformation is known as the Pellet-Cladding Interaction (PCI). In France, the corresponding safety demonstration is based on a comparison of the maximum mechanical stress applied to the cladding during the transient with a technological limit at which a breach can be guaranteed not to occur. This limit is determined from the results of power ramp tests carried out in experimental reactors using commercial fuel rods previously irradiated in a PWR. The phenomenology used to account for a cladding breach is that of stress corrosion due to the presence of fission products, especially iodine.

The power ramp tests were carried out in the Swedish Studsvik R2 reactor and in the OSIRIS reactor at the CEA Saclay centre using sections of fuel rods previously irradiated in a PWR. These tests have enabled *Electricité de France* to define two technological limits, one for Zircaloy-4 fuel rods, and one for M5^{TM} fuel rods in . These limits were then used to define linear power thresholds for the PWR.

As part of a programme aimed at understanding the behaviour of PWR fuel rods during these power transients, these tests have shown that an increase of the burn-up does not necessarily increase the risk of a cladding breach due to PCI. The embrittlement of the cladding due to the time spent in the reactor is compensated for by a more even distribution of stresses within the cladding due to fracturing of the fuel pellets and an increase in visco-plasticity caused by irradiation. The tests carried out on AREVA M5^{TM} fuel rods have also shown that these fuel rods are capable of withstanding much more severe power surges that Zircaloy-4 fuel rods. This will enable *Electricité de France* to relax the operating conditions for its reactors in the near future.

2.4. *Severe accidents: The value of the PHEBUS PF tests in understanding corium progression and fission products behaviour during a severe accident*

Since the Three Mile Island accident in 1978, extensive research on an international scale has been carried out into severe accidents. This work has included integral experiments, single effect tests and the development of models.

The PHEBUS PF programme began in 1988, initially led by the IPSN and then taken over by the IRSN in 2002 as part of a European partnership with the international community. The value and originality of this unique programme resides in the integral character of the experiments carried out. PHEBUS PF is the only programme that produces scaled-down, but representative, simulations of a core meltdown in a pressurised water reactor.

Extending over some twenty years, the PHEBUS PF programme has included a series of five integral tests aimed at studying the main physical phenomena involved during a severe accident. The main objective of the research programme is to contribute to a better understanding of the radioactive discharges that may be released into the environment during such an accident. Investigations have focussed on the nature and quantities of radioactive products released outside the plant, and whether these products are released in the form of dust or gas.

Each of the tests used a different set of parameters, including the type of fuel (burn-up and geometry), the thermal and hydraulic conditions (flow rates and whether conditions were oxidising or reducing), the conditions in the reactor sump (temperature and whether acid or alkali), and the composition of the control rods.

The first four experiments, carried out between 1993 and 2000, provided new data on the mechanisms leading to the degradation of a nuclear reactor core, the radioactive products released, and their behaviour in the various circuits and within the containment.

The first, second and fourth tests (FPT-0, FPT-1 and FPT-2) were specifically intended to study the degradation of fuel assemblies and silver-indium-cadmium alloy control rods typical of those used in the core of a French 900 MWe PWR. The third test (FPT-4) aimed to study the release of low-volatility fission products (barium, molybdenum and ruthenium) and actinides (uranium, neptunium and americium) from the debris bed of irradiated fuel heated until it formed a molten pool of fuel. The formation of a debris bed, as occurred at Three Mile Island, may make it possible to attempt a flooding of the core.

These tests have provided new data in the following areas :

- The reactor core degradation process, with melting occurring at temperatures of 400 to 600°C which are lower than expected.
- The hydrogen production kinetics in an environment rich in water vapour, which are more rapid than those predicted by most existing computer simulations.
- The physical and chemical behaviour of the iodine (more volatile than expected) and caesium (less volatile than expected) carried through the various circuits.
- The complex behaviour of iodine within the containment, including being trapped by the silver in the sump water originating when the rod simulating the silver-indium-cadmium control rods in the reactor melts.

These tests have also confirmed some of the knowledge gained from laboratory experiments in relation to the release of volatile fission products including iodine, caesium and tellurium, and the physical characteristics of the radioactive aerosols carried through the circuits.

The Phébus tests have also been used to validate and improve the severe accident computer simulation software used throughout the world. In France, this is mainly the ASTEC[3] code system developed in close collaboration between the IRSN and the German institute, GRS[4], the central link in the European severe accident network SARNET[5]. The MAAP[6] code used by *Electricité de France* has also been tested during the PHEBUS PF programme.

Some of the results have also been used in the evaluation of source terms[7], and the research carried out during this programme has contributed to the optimisation of the actions and procedures to be implemented in the event of a severe nuclear accident with the aim of protecting the population and the environment.

The final PHEBUS PF test, known as FPT-3, was carried out on the 18th November 2004 with the aim of refining existing data in relation to the degradation of a PWR core fitted with boron carbide control rods as used in the 1300 MWe PWR and the EPR. An additional objective was to test and compare different catalytic recombiners[8]. These are used to reduce the risk of an explosion due to the presence of hydrogen inside the containment during a severe accident involving a reactor core meltdown. It was also observed that the proportion of gaseous iodine within the containment reached 80 % during the fuel degradation phase, a figure significantly higher than that assumed at the time in source term evaluations. The efficiency of samples from the recombiners was also found to be lower than expected. An understanding of the physical basis of these phenomena should result from the CHIP[9] experiments carried out as part of the International Source Term Programme (see below). The results may also be transposed to the reactor case.

A better understanding of the new phenomena revealed by the experimental results from the PHEBUS PF programme will be obtained from the International Source Term Programme led by the IRSN in partnership with the CEA and *Electricité de France* in association with a number of non-French partners. In this programme, the approach will be more global and analytical than the PHEBUS PF programme, and it should provide a more accurate analysis of the accident process. The experimental aspects of this programme should be complete by 2010.

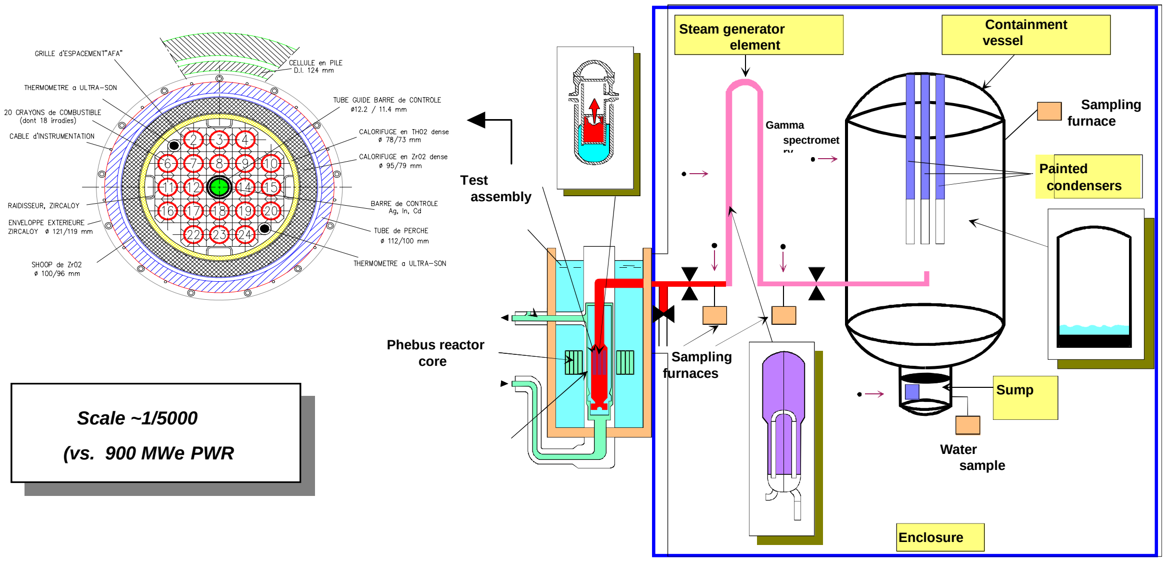
A study is currently in progress with the aim of identifying those knowledge requirements to which integral experiments of the type carried out in the PHEBUS reactor would be relevant. Examples of potential themes include the consequences of the entry of air, flooding degraded fuel, MOX fuel and highly enriched fuel. Work is also in progress to examine the possible contributions of the PHEBUS reactor (see Figure 2) to severe accident safety studies in relation to fourth generation reactors.

---

[3] Accident Source Term Evaluation Code.
[4] Gesellschaft für Anlagen und Reaktorsicherheit mbH (Germany).
[5] Severe Accident Research NETwork of excellence.
[6] Modular Accident Analysis Program.
[7] Source term: The expression 'source term' is used to refer to those radioactive substances that could potentially be released into the environment in the event of a severe accident in a nuclear power plant. The source term associated with the greatest short term risk is iodine 131.
[8] Catalytic recombiners serve to reduce hydrogen levels in the containment by causing the hydrogen to recombine with the oxygen in the air within the containment.
[9] An experimental programme carried out with the aim of reducing the uncertainties associated with the carriage of iodine through the primary coolant circuit.

FIG. 2.

The PHEBUS installation



3. Contributions in relation to sodium-cooled fast neutron reactors

A major test and study programme has been carried out in support of the safety studies for the commissioning and operation of the PHENIX and SUPERPHENIX reactors. This programme focussed on the problems of degradation of the fuel ($UPuO_2$) and the core of sodium-cooled fast neutron reactors. The CABRI and SCARABEE reactors have contributed to many experimental programmes in this field with the aim of gaining knowledge about the phenomena involved in sequences capable of causing boiling of the sodium in an assembly, melting of fuel in the pins, melting of an assembly or even meltdown of the entire core. This knowledge has been used to qualify the computer software tools used in the safety demonstrations and studies.

It should be noted that reactors such as PHENIX and SUPERPHENIX have been designed to contain a core meltdown involving both the pins and the assemblies and to withstand the release of mechanical energy that could arise from the bursting of a bubble of vaporised fuel or sodium generated by a thermodynamic interaction between molten materials and sodium. These energy levels were specified in the decrees authorising the construction of PHENIX (500 MJ) and SUPERPHENIX (800 MJ).

The experimental programmes carried out between 1971 and 2001 in the CABRI and SCARABEE reactors addressed the safety questions raised during assessments of the prevention, detection and consequences of the following three sequences or events :

— Total core meltdown.
— Instantaneous total blockage of an assembly leading to melting.
— Sudden withdrawal of the control rods.

A total core meltdown accident, generally associated with a sequence involving the shutting down of the primary coolant pumps without insertion of the control rods and without any power being absorbed by the generating plant, has been studied as part of the CABRI-1, SCARABEE-N, CABRI-2, CABRI-

FAST and CABRI-RAFT programmes (54 tests in all) using both fresh and irradiated fuel. The conclusions drawn from these tests may be summarised as follows :

— The initial or primary phase of the sequence does not disperse the fuel to a degree sufficient to prevent a later return to criticality.
— Further work is required on the subsequent transition and secondary surge phases in order to guarantee that recompacting mechanisms do not result in releases on energy greater than those occurring during the first phase.

The instantaneous total blockage of an assembly accident has been studied as part of the SCARABEE-N programme (10 tests) using fresh fuel only. The essential requirement associated with this type of accident is to ensure that the reactor detection systems (variation in reactivity, cladding breaches, and heating of neighbouring assemblies) can react sufficiently quickly before the meltdown propagates to the rest of the core. The results of these tests showed that the meltdown will propagate to neighbouring assemblies unless the accident is detected within fifteen seconds. The development of effective and wide ranging detection systems is therefore essential. Further work is required using irradiated fuel as the meltdown is likely to propagate faster in this case due to pressurisation of the melt pool by the fission gasses.

The sudden withdrawal of the control rods was studied in the CABRI-2, CABRI-FAST and CABRI-RAFT programmes (9 tests in all). One of the main questions raised during development of the technical instructions for the SUPERPHENIX reactor was whether it was permissible to accept the partial melting of the pins when the bars are withdrawn, given that one of the pins may incorporate a manufacturing fault. Test RB2 in the CABRI-FAST programme was carried out in order to address this question. During the test, an irradiated fuel pin with reduced cladding thickness in order to simulate a fault was subjected to gradually increasing power until the fuel in the pellets partially melted. This test showed that molten fuel was ejected when the fault gave way. This provided justification for the decision to include a system to limit the withdrawal of the control rods in the SUPERPHENIX, thereby reducing the risk of the fuel melting when the control rods were withdrawn.

In conclusion, the CABRI and SCARABEE reactors have made a significant contribution to safety research in relation to sodium-cooled fast neutron reactors, and a number of avenues for future research have been identified for further consideration should the development of this type of reactor ever be restarted.

4. Conclusion

The examples given above demonstrate the fundamental importance of experimental reactor tests alongside modelling and analytical tests in the design and development of safety demonstrations. These tests are useful in the following areas :

— Characterisation of poorly-understood phenomena.
— Revelation of previously unknown phenomena.
— Contribution to the development and overall qualification of the simulation tools used in the safety demonstrations.

ACKNOWLEDGEMENTS

The authors thank Georges Hache, Sophie Pignet, Sandrine Boutin and Aude Taisne from IRSN for theirs participations.