

Phénix

30 years of history:
the heart of a reactor

Phénix" means phoenix. The phoenix: a mythical bird which, being the only one of its kind, was unable to breed. After living for hundreds of years, it would burn itself on a pyre, then to be reborn from its own ashes. This was the name given, by analogy, to the nuclear power plant whose story is told in this book: a reactor capable of producing energy anew from by-products of the burn-up of its own core.

An original reactor: a sodium-cooled, fast neutron breeder reactor.

Important scientific and technical results.

Serious technical problems successfully overcome.

A key role in R&D on burning of radioactive waste.

Construction; the first years; performances; problems; safety upgrading; renovation; return to power operation - each period in the history of the Phénix power plant, though essentially a technological and scientific story, is also a tremendous human adventure in which everyone concerned committed themselves wholeheartedly to the success of the enterprise.

Jean-François SAUVAGE has worked as an engineer for the French nuclear safety authority, the French electricity utility (Électricité de France) and the French atomic energy commission (Commissariat à l'Énergie Atomique). He has worked on fast neutron reactors in several functions. From 1984 to 1990 he worked as a safety analyst on these reactors. He was then head of the safety department and subsequently head of engineering at the Creys-Malville nuclear power plant. From 1998 to 2002 he directed the Phénix plant, first as deputy and then as director of the facility.



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EDF
Electricité
de France

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*Jean-François
SAUVAGE*



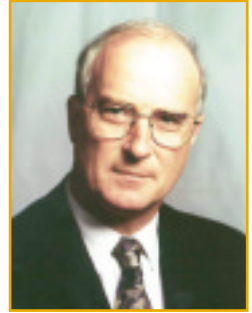
Translated from French
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To all who work in fast breeder reactors



Jacques BOUCHARD

*Head of the Nuclear Energy Commission
of the French Atomic Energy Commission (CEA)*



Thirty years already ! When Phénix started up in 1973, no one doubted the need to develop, and rapidly, the breeder reactor type. Our oil-based civilisation was facing the first economic shocks and the resulting early geopolitical warnings. In full growth, the western world feared the threat, clearly none too far in the distance, of a shortage of the oil which was so indispensable to continued progress. All the calculations showed that thermal neutron reactor development potential would remain too limited to provide for future needs. And though at the time the concept of sustainable development had not yet influenced thinking, mid and long-term forecasts were indeed the foundations for the choices to be made. In a climate of fierce industrial and international competition, there was no time to be wasted. Less than eight years passed between the first design project specifications and reactor criticality. The teams from the CEA, EDF and the industrial community moved fast, efficiently and dynamically. This enthusiasm for an ambitious objective would later be found in the early studies for a 1000-MWe power plant and in the Superphénix pre-project.

For fifteen long years, Phénix met every expectation. The plant started up flawlessly and was connected to the grid in record time. The reactor worked beautifully, despite a few incidents which did not challenge the basic design choices. The first demonstration was the viability of using sodium as the coolant for a 300 MWe-type reactor. At that time, the only water reactor producing electricity in France in the same power category was the reactor at the Ardennes power plant. A second major result of the early years of Phénix operation was the effective demonstration of breeding. The reactor produced approximately twelve percent more fissile matter than it consumed. The validity of the estimated calculations was first proved by the experiments in the reactor core, then confirmed by the recycling of the plutonium produced. The core was actually completely recycled three times, and this experience remains today the only industrial demonstration worldwide of the possibility of using all the uranium through breeding.

Among the other achievements attained by Phénix, stand the flexibility of reactor use and the excellent radioprotection results, two undeniable advantages of the pool type reactor concept, and of course the impressive amount of experimental results obtained from the many in-core irradiation devices. The main incidents which occurred during the first fifteen years involved the heat exchangers (intermediate heat exchangers and steam generators). They confirmed the conclusion reached on all the other reactor systems for the same power range – that the choice of materials and the design detail are delicate and rarely succeed the first time around. Fortunately, the modularity and the overall design allowed for rapid solutions to the first failures. Other incidents, this time characteristic of the choice of coolant, were the sodium leaks. These only affected the secondary circuit, and thus did not involve radioactivity, but did each time cause localised, rapidly controlled fires. There were over



PREFACE

twenty throughout the reactor life, which confirmed the need for a powerful detection system and the ability to intervene rapidly to control the incident.

The late 1980's saw storm clouds gather over Phénix. Worldwide, protests against fast breeder reactors had reached a peak. Superphénix, the focal point of these protests, was facing technical problems which, once again, occurred at the worst possible time, though they did not question the overall concept. And Phénix, which had operated smoothly up until then, was suddenly shaken by the well-known negative reactivity trips, whose origins would be searched for, in vain, for months to come. The final conclusion deemed that the phenomenon was harmless and could have had at least two plausible causes, signal interference or mechanical movement by the sub-assemblies in the core. The reactor was loaded with additional instrumentation to record an event ... which never occurred again.


The decade which followed was marked by two major orientations – the research program on very high activity waste which implied experimentation for transmutation of the waste and created a new objective for the reactor – and the need to upgrade the reactor safety level. This safety upgrade, which quickly proved itself more challenging than originally thought, was a unique experience which provided in-depth knowledge for future projects. After thirty years, the reactor is once again operating, and there is no doubt that it will continue to provide major results for the future of fast neutron reactors.

Because there is a future for fast neutron reactors, and more aptly than ever before, the symbol of the phoenix comes to the fore. After all the western countries had more or less obstinately stopped all form of development of fast neutron reactors – first the Americans, then the Germans, the British and finally the French with the shutdown of Superphénix – the concept is now returning in force, in the fourth generation studies, on the grounds of sustainable development and the need to burn the actinides rather than placing them in waste. Phénix is once again on centre stage, part of the international scene where it has always had its place, having continuously been courted by many foreign collaborators and visitors whose presence and interest were always warmly welcome.

In the upcoming years, this international cooperation should focus on other tools, such as Monju, a great hope, BN 600, if its lifetime allows for continued work, and other prototypes, in India, China

In retracing the life of this reactor, author Jean-François Sauvage has written important history, emphasising the significance of the context and of the people involved in every step of the way. I would like to join him in paying homage both to the teams, from the visionaries at the outset to the artisans behind the renovation work, whose human qualities made such a success possible, and to the local populations and their elected leaders who always kept an open mind, and whose reception shall remain unforgettable for the many French and foreign colleagues who had the pleasure of working with Phénix.





It is not because things are difficult that we do not dare,
it is because we do not dare that they are difficult.

Seneca



ACKNOWLEDGEMENTS

- This book would never have seen the light of day without the agreement of Loïck MARTIN DEIDIER, Director of the Centre d'Études de la Vallée du Rhône, who grasped the opportunity.
- I am particularly grateful for the honour done to me by Jacques BOUCHARD, Director of Nuclear Energy at the Atomic Energy Commission, in authoring the preface to this book.
- Jean MÉGY and Yves CHELET, well-known members of the French nuclear community and both excellent teachers, inspired me to follow in their footsteps and were generous with their excellent advice.
- I am most grateful to the witnesses who answered my appeal for testimony. In doing so they showed not only their attachment to the Phénix plant, but also great kindness and interest in the book.
- My wife Odile MÉGY showed endless patience in reading each successive version of the manuscript, and once again put up with having her private life invaded by fast neutron reactors.
- My thanks to everyone who helped me produce this book by providing information or suggesting corrections. Special thanks to Pierre GRENET for the illustrations, and to Laurent MARTIN for checking the English translation.
- To conclude, this assessment owes a great deal to all those who, over the past thirty years, have written up synthesis reports on events occurring in the life of the Phénix plant. The effort of writing things down is essential if the experience gained is to benefit our successors.



"Phénix" means "phoenix". The phoenix: a mythical bird which, being the only one of its kind, was unable to breed. After living for hundreds of years, it would burn itself on a pyre, then to be reborn from its own ashes. This was the name given, by analogy, to the nuclear power plant whose story is told in this book: a reactor capable of producing energy anew from by-products of the burn-up of its own core.

That was the practical aim of the reactor's designers and those who put France (and more specifically its atomic energy commission Commissariat à l'Énergie Atomique and its national electricity utility Électricité de France) on the road to fast neutron reactors. In 1974, on the symbolic date of 14 July (Bastille day), its design and construction phases completed, the Phénix power plant went into power operation.

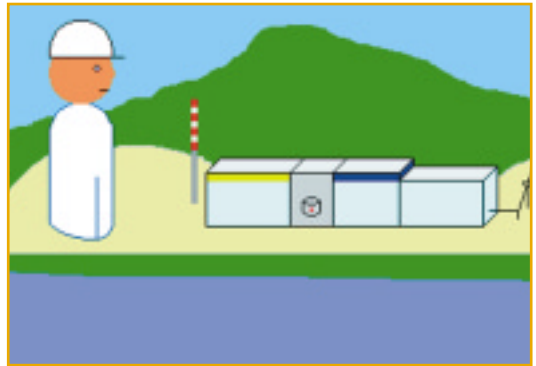
In this book we look back over the thirty years that have passed since then, to consider the history of the plant and the contribution it has made. Because often the past fades as it recedes in time and important facts are overshadowed by more recent events. Furthermore, objectives and working methods have changed in thirty years.

"*The past is always golden!*" Perhaps, but although the Phénix plant's results were often better than its designers had hoped, it sometimes took considerable effort to achieve that. On the other hand, if one only remembers the trials and tribulations, one forgets how innovative solutions were deployed to overcome them. For a balanced assessment we need to look at the whole story, moments of elation and times of trouble alike.

Highly motivated men and women succeeded in tackling each successive challenge. Of those who started up the Phénix plant, there

are few who have not yet reached retirement age. Some, alas, have already passed away. Writing this book involved assembling the available written information and profiting from the memories of those involved. This assessment is also a way of paying homage to all those who have put their hearts and their energies into the tremendous human adventure of running a power plant that is, in many respects, unique.

The story of the Phénix plant is essentially a scientific and technological tale, even if humans – the people who ran it – are ever-present. To tell that story, therefore, we have divided it into **seven periods** of five to six years, each centred on a specific issue. The little man and his power plant on the drawing below will be with us all the way through.



- From 1968 to 1974, once France had opted for fast neutron reactors and the main options of the project were decided on, construction of the plant and the commissioning tests ran smoothly,
- From 1974 to 1980 were the first years in power production. The project's characteristics were verified in practical terms, its limits began to be extended, the first major incidents on the intermediate heat exchangers were overcome.



- From 1980 to 1986, after repair of the steam generators where sodium-water reaction had occurred, the prosperous phase continued: the reactor gave complete satisfaction and the records were beaten.
- From 1986 to 1992, all continued as well as before, but there were warning signs of the trouble to come, both in the Phénix plant and in other facilities.
- From 1992 to 1998, safety upgrading studies were in the news. Plant operation varied in inverse proportion to the scale of the studies.
- From 1998 to 2003, production was out of the question: it was time for a thorough renovation of the entire facility.
- From 2003, when power operation was restored, preparations began for the end of the power plant's life: the last irradiations and the final dismantling.

To avoid weighing down the main text of the book, the **technical explanations** regarding the Phénix plant's design and operation have been put together at the end, in a descriptive appendix and a glossary of abbreviations and technical terms specific to the power plant. More detailed information can be found in documents and conference proceedings, of which there is a partial list at the end of the book.

The author had free rein in writing the book; despite the great care taken by those who kindly agreed to read the manuscript, the author assumes full responsibility for the content and any errors or omissions the book may contain.



INTRODUCTION



CONSTRUCTION
AND TESTS
(1968 - 1974)



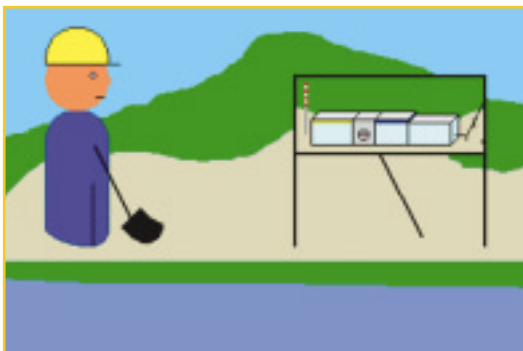
Landmarks

28 January 1967	First divergence of Rapsodie
5 May 1969	First concrete poured at the Phénix site
5 March 1970	Non-proliferation treaty comes into force
2, 12 and 25 November 1970	Installation of primary, safety and main vessels
28 July 1971	Discharging of first sodium
1 September 1971	Installation of reactor slab
25 September 1972	Discovery of natural fossil reactor at Oklo in Gabon announced at the French Academy of Science
29 November 1972	First divergence of BN 350 (USSR)
29 and 30 November 1972	Storage drum filled with sodium
12, 18 and 19 December 1972	Secondary cooling circuits filled (numbers 3, 2 and 1)
3 - 10 January 1973	Primary sodium cooling circuit filled
13 March 1973	Conseil Supérieur de la Sûreté Nucléaire (high council for nuclear safety) and Service Central de Sûreté des Installations Nucléaires (nuclear installations safety central service) formed
3 August 1973	Core fuelling begins
31 August 1973	First divergence of the Phénix reactor
17 October 1973	First energy crisis: OPEP raises prices by 70%
13 December 1973	First connection to electrical grid
1 March 1974	First divergence of PFR (United Kingdom)
5 March 1974	1st multi-year contract to build 16 standardised 900 MWe PWR units
12 March 1974	Reactor reaches rated capacity
14 July 1974	Phénix plant commissioned



The first kilowatt-hour of nuclear electricity was produced on December 20, 1951. It came from the small prototype reactor EBR 1, built at the Idaho Falls (United States) research center. It was a fast neutron reactor cooled by a liquid alloy made of sodium and potassium (NaK).

As early as in 1945, Enrico FERMI explained in Los Alamos: “*The first country to develop a fast breeder reactor will have a commercial advantage for the exploitation of nuclear energy*”^[1]. It was in this context that the United States, USSR and United Kingdom started to develop fast neutron reactors as early as the end of the war. The Americans were the first, with the Clementine reactor, going critical in 1946, continuing with the construction of a succession of fast neutron reactors of increasing power. In 1956, the electricity company Detroit Edison started the construction of a commercial power plant, called Enrico Fermi, in Michigan. In Obninsk, the Soviets started up their test reactor, with increasingly powerful configurations. They then continued with the construction of an experimental reactor equipped with a turbo-generator set, the BOR 60, and later with a prototype power plant on the river of the Caspian Sea (BN 350). The British went directly into the construction of an electrogenic experimental reactor in Scotland (Dounreay) before starting the construction of the Prototype Fast Reactor (PFR) on the same site.



France joined this fascinating scientific and technical adventure during the sixties. The aim was to establish the basis for a new range of reactors, called fast neutron reactors or fast breeder reactors^[2], which would make it possible to optimize the exploitation of uranium reserves to supply enough electricity for several centuries. Studies were started for the construction of a demonstration reactor of significant power, called Phénix. This was with a view to constructing large commercial reactors, at a time when the experimental reactor Rapsodie had not yet been started up. They were obliged to construct non stop, with no break between projects, in order to absorb new knowledge quickly. This allowed them to take advantage of the skills of the teams involved and the synergy which drove them, in order to make up for lost time as regards the foreign countries that had been part of the adventure for several years already. The teams involved in the projects took up the challenge, galvanized by what was at stake.

In this chapter, the context in which the fast neutron type of reactor was developed should not be forgotten. More specifically, in France, all this was concretized with the experimental reactor Rapsodie. The main studies concerning the Phénix plant will be described below, followed by an account of the building work and construction of the facilities. The chapter will close with the tests and commissioning of the power plant.

1.1. The context

At the end of the fifties, the Atomic Energy Commission (CEA) developed natural uranium gas graphite reactors (UNGG in French) to produce electricity and plutonium for military

[1] Quoted by Georges VENDRYES in “*SuperPhénix pourquoi?*” (“*Why Superphénix?*”) to whom the first chapters of this report owe much.

[2] These were also called “*super fast breeders*” (“*surrégénérateurs*” in French), although the term was progressively abandoned in France, particularly because of the difficulties involved in pronunciation.



purposes (the G1, G2 and G3 reactors in Marcoule, diverged for the first time in 1956, 1958 and 1959 respectively). From 1962 to 1971, EDF maintained the same policy regarding electricity generation, building and operating six reactors in the towns of Chinon, Saint-Laurent-des-Eaux and Bugey. In 1970, when the time came for EDF to choose the main (and in the end only) type of reactor to equip its industrial basis for producing nuclear energy, they opted for the pressurized water reactors, which had proved their solidity in the US, a first 300 MWe unit being already in operation in France, at Chooz, since 1966.

At the same time, the **fast neutron** type of reactor looked very promising. Fast neutron reactors can effectively breeders, because of the excess of neutrons produced by the chain reaction in their core, which means they produce more fuel than they use. In fact, the excess neutrons are used to trigger nuclear capture reactions by the uranium 238 (a breeder element which makes up more



The American EBR 1 reactor

A tale of damp wood

Here is an image to explain fast breeding: after a long day's walk in the rain, at last we reach the refuge. We have a look at the pile of wood: it is very big but all the faggots are damp, so they will not burn! Nevertheless, when we take a closer look, we notice the presence of a few dry twigs in the middle of each faggot. First solution: painstakingly sort the dry bits from each faggot: there will be enough to cook the meal and warm up for an hour or two.

But if we shrewdly lay out the faggots of wet wood around the fire we have started like this, without undoing them, we will gradually dry them until they are dry enough to catch fire and feed the flames. In this way, not only will we be able to cook our meal, but we will keep warm all night, we will dry our clothes and we will prepare a stock of dry wood for the hikers who arrive after us.

than 99 % of natural uranium) which is transformed first into uranium 239 and then, after a few days and two successive β^- decay into plutonium 239 (a fissile element).

Unlike what happens in moderate reactors (water or graphite reactors), the production of plutonium in fast neutron reactors may be higher than the destruction of fissile nuclei. Moreover, the other isotopes of plutonium created (mass number greater than or equal to 240) do not prevent the chain reaction or even take part in it. A certain amount of fissile matter is thus created in the core of the reactor and, on the scale of a fleet of fast neutron reactors, almost all of the uranium nuclei are used progressively to produce energy, by means of a process that transforms breeder uranium 238 nuclei into fissile plutonium 239 nuclei. The energy resources contained in the uranium ore can thus be multiplied by a factor commonly described as close to 70.

All these elements factors raised hopes for a prosperous future in which electricity consumption, and energy independence, could be guaranteed for centuries to come, at least as far as electricity were concerned. The 1974 oil crisis, the sharp increase in nuclear energy and the risk of tension on the uranium market that can be feared had not yet made their contribution to the cause of the supporters of fast breeder reactors^[3]. Nevertheless, these supporters pleaded their cause with enthusiasm and conviction, making the directors of the CEA persuade their supervising ministries that they should invest the necessary funds.

During the late fifties and early sixties, the **means for studies and trials** necessary for the research and development into developing fast neutron types of reactor were being made available. Many systems were considered in order to make the fast breeder program successful. They had to develop fuel containing large quantities of fissile matter (²³⁵U or ²³⁹Pu), find a way to evacuate the considerable amount of heat produced per unit volume in the core, ban moderator materials, and so on.

In a fast neutron reactor, it is impossible to obtain the chain reaction with natural or slightly enriched uranium. There must be a high proportion of fissile matter in the fuel in order to compensate for the lower probability that fast neutrons will trigger fission. Plutonium stood out as the best fuel because it is produced from uranium 238, which is abundant in natural uranium. It has the advantage of producing an average of more than three neutrons for each fission, thus making fast breeding easier. In addition, France, which had mastered the reprocessing of spent fuels from natural uranium gas graphite reactors, thus had at its disposition the plutonium necessary for starting a cycle. The fuel chosen came in mixed oxides



Liquid sodium

UO₂ - PuO₂ pellet form. It was thus necessary to master the manufacturing process for these pellets and implementation. The pellets are clad in stainless steel pins. The power density produced in these reactors is so high that very small diameter pins must be used.

The performance required, considering the technological capacities of this time, led to the choice of sodium as coolant. It has many advantages: it plays only a small role in slowing down neutrons (which is essential for a fast neutron reactor), its properties in thermal transfer are excellent, it can circulate easily, its boiling point is high (883 °C at air pressure), it is only slightly corrosive provided it is pure, it is a common industrial product, it is produced in large quantities in France (several thousand tons per year), and so on. It nevertheless has two disadvantages: it is opaque, hence the need to develop highly sophisticated techniques as a replacement to direct vision when handling and inspecting, and it produces a chemical reaction when in contact with air or water. This means that a wide range of precautions become obligatory to maintain the tightness of the circuits which contain it and limit the consequences of possible leaks.

1.2. Rapsodie

At the end of 1958, a draft for a pre-project for an experimental breeder reactor with fast neutrons and sodium was drawn up. Its name was to be Rapsodie. Its characteristics were as representative as possible of the industrial reactors envisaged for the future (mixed UO₂ - PuO₂ fuel, sodium, materials, power density,

[3] No one foresaw either the anti-nuclear backlash, or the end of the civil nuclear program in the US.





The Rapsodie reactor at Cadarache

temperatures and so on), with the exception of electrical production. Similarly, certain choices had already been made and would not be questioned for the following reactors: the hexagonal design of the fuel sub-assemblies, suspended vessel, intermediate heat exchangers, mechanical pumps for the main cooling systems and electromagnetic pumps for the auxiliary cooling systems, and so on.

The **Cadarache site** was chosen for the reactor, as well as for other facilities necessary for the development of the type, which would be added progressively over the years: test rigs stands for the sodium technology and steam generators, the reactors Cabri and Scarabée (1964, studies on core accidents), Harmonie (1965, study of materials) and Masurca (1966, study of cores), and the Esmeralda facility (1982, study of sodium fires). The project was financed by a collaboration between the CEA (65 %) and Euratom (35 %).

The Rapsodie reactor is a loop reactor: two primary cooling systems are connected to the reactor's vessel. Each system transfers its heat to a secondary system, also in sodium, which evacuates the energy via a sodium-air exchanger. Around sixty fuel sub-assemblies make up the core, which has a volume of less than 50 liters. The temperature of the sodium when it leaves the core can reach 500 °C. The

fuel retained was, from the outset, mixed oxide with uranium and plutonium.

The facility was built between 1962 and 1966. It was in 1966 that the vessel and cooling systems were filled with sodium. The first criticality took place on January 28, 1967 and, two months later, the reactor reached the power of 20 MWth for which it had been designed. At the end of the same year, its power was increased to 24 MWth. In 1970, full recasting of the core made it possible to increase the reactor's power to 40 MWth (the "Fortissimo" operation). This transformation considerably increased Rapsodie's possibilities as an experimental tool, particularly for performing the irradiation tests that would be essential for designing the fuels for future reactors.

The **good running** of Rapsodie, which has a level of unprogrammed unavailability of around 4 %, shows that, in a context where the only example of international collaboration is within the European Union (and then, only

Choosing the fuel

"Choosing the type of fuel to be used was particularly important. Contrary to all the other fast neutron reactors in existence at the time, which used highly enriched uranium in the form of a metal alloy, the decision was made from the beginning of the Rapsodie project to opt for a plutonium-based fuel, which we considered to be obviously the best for the future of fast breeders. After many tests conducted in parallel on a range of variants, the formula finally chosen at the end of 1962 was a mixture of plutonium and uranium oxides.

At the time, using this type of fuel may have seemed rather daring. With hindsight, we can see that virtually all the fast neutron reactors built later, throughout the world, use this type of fuel. Rapsodie's other specifications were modest for the most part but this essential point gave it a good headstart."

Georges VENDRYES, op. cit.





six of its members), the CEA teams know how to design, construct and run a fast neutron reactor, using plutonium fuel and cooled by a liquid metal (sodium). The main components (primary pumps, sodium exchangers, purification systems, security logics and so on) have shown themselves to be highly reliable. This does not mean that there were not problems, although it was possible in every case to find a solution within a reasonable amount of time and under satisfactory conditions. This, in itself, is an excellent way of acquiring experience, as faults are identified, stimulating the need to reconsider. Finally, the experiments on the irradiation of fuels and materials produced important results, particularly in terms of the fundamental fuel parameters (pellet-fuel gaps, density of the mixed oxide, diameter of the pins and so on), which made it possible to prepare the choices to be made for the Phénix plant and, ultimately, for the development of the fast neutron type of reactor.

In 1971, the irradiations performed on the Rapsodie reactor revealed a phenomenon of irradiation swelling in the stainless steel of the fuel pin cladding. This was the result of the damage caused in the structure of the steel in the high neutron flow. The swelling forecast extrapolated for the level of burn up targeted in the Phénix reactor (and the corresponding damage in the steel) included margins of uncertainty such that a pessimistic analysis could have resulted in the project being abandoned: deformities in the fuel sub-assemblies at the end of their life span

would be incompatible with the requirements for handling them.

Despite all this, it was decided that the Phénix project should continue, at least until more precise results were obtained. In the very worst case scenario, limited combustion rates would be obtained, but this would at least have the advantage of revealing the capacities of this type of reactor. The more precise results that were obtained at a later date made it possible to understand the swelling phenomenon better. A full research program was set up to choose as cladding material the steel that was the least sensitive to swelling.

1.3. Studies of the Phénix plant

Following on from American and British success, the timetable of studies essential for building a prototype reactor was established, without waiting for the Rapsodie reactor to produce the results necessary for continuing development of fast neutron reactors. It was thought better not to have too much of a time lag between the construction of successive plants, as a means of keeping the skills and enthusiasm of the CEA and industrial teams at their highest level.

A preliminary study of a 1000 MWé plant was conducted in 1964, prior to determining the characteristics of the Phénix reactor. The aim of the study was to ensure that it would be possible to extrapolate the options chosen at a later date. The next stage was preliminary studies on a demonstration power plant. As the choice of a name that meets with general approval is essential for bringing a project to life, everyone was systematically asked to give their opinion. Eventually,



the name “Phénix” was mentioned, and finally adopted once it had received unanimous approval. At the same time, EDF started showing an interest in fast neutron reactors and expressed its desire to play an active role in their development. This resulted in an effective partnership based on trust, at a time when the relationship between the two organizations was somewhat strained because of the choice of reactor type to be retained (natural uranium gas graphite or pressurized water reactors) for the national program for the production of nuclear energy.

In the course of 1965, the main technical characteristics of the Phénix plant were determined, in close collaboration with all the CEA departments involved. A common agreement was soon reached, without the need for prolonged debates to decide between rival plans. The essential choice of the fuel to be used was thus made as a direct follow-on of the choice already made successfully for the Rapsodie reactor, given the results obtained by the irradiations in this reactor. The quantities of plutonium available in France were perhaps insufficient for a complete reactor core. It was also part of the plan to use enriched uranium oxide on a temporary basis to make up part of the first core. After hesitating between 80 MWé (like the Brennilis heavy water reactor) and 125 MWé (the EDF limit in the 1950s), the electrical power was set at 250 MWé. This was to make it possible, without any major adaptation, to use a recently designed turbo-generator set that had already proved its worth in EDF’s conventional coal or oil power plants.

The sodium was kept as coolant. Finally, an integrated design was adopted. All the elements of the primary cooling system (core, pumps and intermediate heat exchangers) were housed inside a large-scale vessel with no penetrations. This solution, used by the Americans in the experimental EBR 2 reactor, had certain advantages in terms of safety in particular: the large quantity of

sodium made it possible to deal with thermal transients or accidental situations better, and leakages of primary sodium were easier to control in a vessel (surrounded by a safety vessel) than in several systems circulating in a variety of places. In addition, as sodium does not have to be pressurized, the structures were not thick. It was reassuring to discover, at a later date, that the British had made the same choice, and as discreetly as the French, for their Prototype Fast Reactor which resembles the Phénix plant in many respects.

In 1967, the CEA drew up a **detailed pre-project**, in liaison with the constructors and in association with the future operators. The power of the reactor resulted in determining the number of both the secondary cooling systems and the main components: three primary pumps and six intermediate heat exchangers. The thermal gradients in the primary cooling system justified the design of certain structures for organizing the sodium flow. Operating at temperatures of 400 to 600 °C implied using austenitic stainless steel, which was chosen for its mechanical properties, satisfactory at high temperatures over long periods of time, and its good corrosion resistance when sodium is present.

A maximum accident was defined to take into account the fact that the core was not in its most reactive configuration (compaction would have increased its reactivity) and that the molten fuel could react violently with the sodium. The purpose of this accident was to define the scale of the resist-

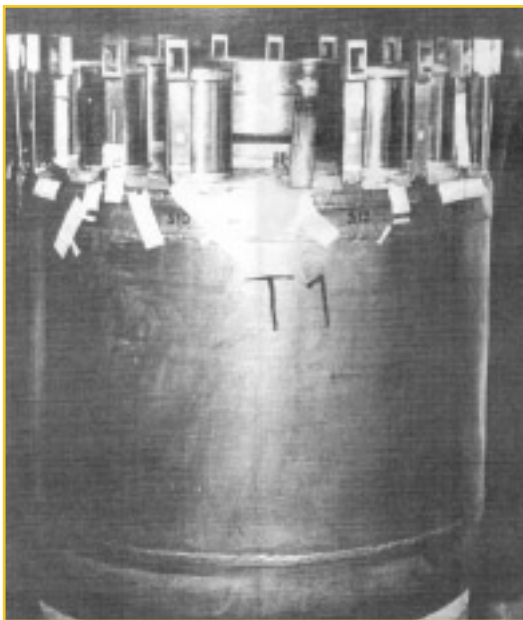
A fantastic human adventure

The people who played a part in this period remember it as a fantastic adventure. They worked enthusiastically, under the management of people who knew how to make a team, composed of people from diverse organisations, work together. They also shared a feeling of working for “humanity’s future energy supply”. This balance between ethics and day-to-day work is a highly valuable asset.



ance of the reactor block, in order to guarantee that the radioactive products would be contained. For this reason, it is commonly referred to as a containment design basis accident.

As it is impossible to describe the causes of this type of accident, given the complexity of the phenomena involved, none of which taken individually would be capable of resulting in major propagation, a global, conservative approach is adopted. Independently of any potential causes (which are all insufficient), it is supposed that there is a sudden increase in the reactivity of the core (power excursion). With the help of mathematical models and tests on scale models (with specially adapted explosives)^[4], the resistance of the vessels, structures and components of the reactor can be shown to have appreciable safety margins.



Mock-up of the reactor block undergoes explosive test

The decision was taken to operate with a permanently low level of sodium impurities in order to prevent any small openings and pipes from becoming clogged, as well as to limit structural corrosion^[5]. Unscheduled drainage of the primary sodium was made impossible. The systems were

designed to make easy operation in natural convection. The fuel is handled in sodium thanks to transfer arms. Internal storage inside the vessel plus external storage were included in the plans to make it possible to cool the spent sub-assemblies before dismantling and evacuating them.

By interposing an intermediate sodium heat exchange circuit between the reactor and the steam generators, it was possible to prevent any accidental contact between the radioactive sodium in the primary cooling system and the water in the electricity generating system. In this way, should one of the tubes on the steam generator leak, the sodium-water reaction that would follow would only concern the non-radioactive secondary sodium. This type of accident is classified as a chemical accident, involving the non-nuclear part of the facility.

The steam generators, which are not used in the Rapsodie reactor, are modular, thus making them easier to replace. Each evaporator is associated with a superheater and a reheater. The steam generators have been designed to be cooled with water or air during shutdown periods for the power plant. Several prevention and protection measures concerning possible leakage from the water or steam pipes have been developed. In addition to the means employed on the Cadarache site and those of the constructor, a test installation of the steam generator modules was made by EDF's "Etudes et Recherches" (studies and research) department. The test was conducted in EDF's Renardières center with the support of the CEA. The 50 MWth facility was in particular composed of a system with 100 tons of sodium heated by means of a gas boiler.

[4] In particular, a model at a scale of 1:25 of the reactor was subjected without damage to explosions corresponding to energy release three times higher (in relation to the scale of the model) than those retained for the containment design basis accident.

[5] At the same period, the British chose a higher level of impurity in order to make it easier to lubricate the moving parts.



In 1969, the CEA and EDF signed a protocol for the common construction and operation of the Phénix plant. Costs were to be split 80 % for the CEA and 20 % for EDF. For the construction of the power plant, an original organization was developed to reduce as much as possible any unforeseen technical and contractual events, given the CEA previous experience in such matters: the construction work was entrusted to an integrated team, combining engineers and technicians from the CEA, EDF and the firm GAAA^[6], in which the role of each was decided in relation to the skills they could provide, independently of their belonging to one or other of the three organizations involved. In particular, certain key functions in the project (study manager, construction manager) were given to engineers from GAAA. The team was based in Saclay, except for the agents who were supervising the work directly. It was composed of a little less than 200 people, including approximately 80 engineers. The project and operation teams were managed by people “targeted” from teams that had already successfully completed the other great projects before Phénix: the G2 and G3 reactors, Rapsodie, Célestin 1 and 2, natural uranium gas graphite reactors and so on.

[6] Groupement Atomique Alsacienne Atlantique, a subsidiary of:
- the Société Alsacienne de Constructions Mécaniques (SACM), a subsidiary of the Compagnie Générale d'Electricité (CGE),
- the Chantiers de l'Atlantique, a subsidiary of Babcock Wilcox.

In the course of the studies and associated tests, certain problems arose that meant that the design had to be modified. **The way in which the main vessel was suspended under the slab had to be changed to guarantee that the upper hangers would hold should there be an accident in the core.** A hydraulic baffle was added against the main vessel to channel the cold sodium (that is, at 400 °C in power) and to keep the vessel at a temperature of around 420 °C. This was to decrease the risks of creep, as validation of the acceptability would have necessitated very long experiments.

The leaktightness of the penetrations of the primary vessel baffles by the intermediate heat exchangers was finally guaranteed by an argon seal. For the leaktightness of the rotating plug, it was finally decided to use a fusible metal seal as this solution was deemed more reliable than inflatable seals on the basis of the experience acquired with Rapsodie. On the secondary cooling systems, the expansion tanks were made spherical in order to give them greater resistance to overpressure in case of a violent sodium-water reaction in the steam generator. Regarding the same potential accident, the number of burst diaphragms was increased from four to six per steam generator.

1.4. The worksite

The work prior to starting the worksite began in February 1968, coming hot on the heels of the decision to put the Phénix plant to the north of



Main excavations



The reactor site in 1970

the Marcoule site rather than on the Cadarache or La Hague sites. La Hague was considered to be too far from Cadarache, where all the R&D facilities associated with sodium technology are situated, Cadarache itself not having a sufficient cold water supply. A campaign of reconnaissance surveys made it possible to obtain details concerning the geology and hydrology of the area, and thus to settle the definitive implantation of the works and determine the type of foundations for the building. A 750-meter dike protected with a layer of broken stones was built along the Rhône to guarantee protection from flooding. The works on the infrastructure of the site were done at the same time: access roads, electricity and industrial water supplies, drainage of rain and waste water, reception area for companies and so on.

The site itself was opened in October 1968. The first stages consisted of digging the **main excavation** inside which the foundations and infrastructures of the buildings would be placed. The excavation site was enormous: 180 meters long, 50 meters wide, and 11.5 meters deep. Its boundaries were side walls moulded in the

ground to retain the natural soil. A network of pumping pits in the water table made it possible to control the water supply and uplift caused by the removal of river deposits and rock. The earth work took eighteen months to complete.

One of the particularities of the civil work involved in the Phénix plant was the metallic liner. Its function was to guarantee the leak-tightness of the underground parts of the buildings containing the sodium circuits, the reactor and handling buildings on the one hand and the steam generators building on the other. It took four months to complete these two liners. The work required a level of care and careful supervision that were relatively unusual at the time on a civil work site. The prefabricated metal sheets were 14 meters square and 10 mm thick for the base, 5 mm for the walls. They had stiffeners and connectors and were assembled on the edge of the excavation. They were held in position by a system of rigid rods propped against the moulded wall of the excavation.

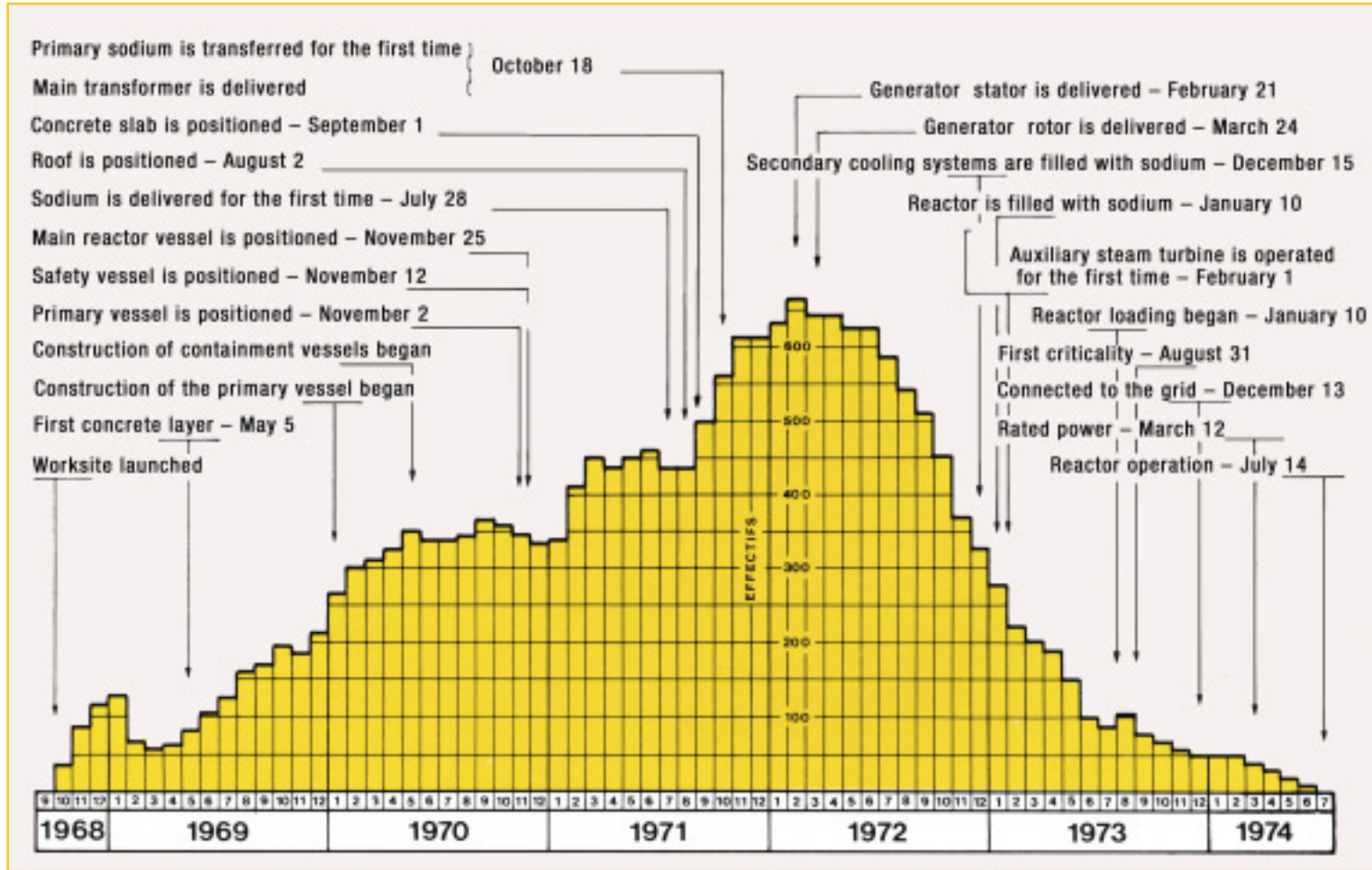
The assembly procedures took into account the shrinkage from the welding and temperature variations during the day. Most of the welding was radiographed and verified by dye penetration. The concrete in the base slab of the buildings was cast once the liner had been completed. Injections of cement and resin then filled the spaces beneath the metal sheets, particularly the recesses that had made it possible to position the radiography films.

The **superstructure of the reactor building** was composed of 25 cm thick prefabricated panels, propped against the studs cast on site. There were approximately 270 panels, horizontally prestressed, which had been manufactured sufficiently in advance to allow them to dry for at least five months. They were put in place in successive layers, resting on the steel whilst waiting for the studs, before the concrete of these studs was cast as high as a panel. Once the panels had been fully stacked, the parapet





Sequence of events on the worksite



was cast and vertical prestressed cables were put into position across the studs and panels. The metal frame of the roof was then installed, with its main beams slotted into the tops of the studs.

Throughout the installation period for the reactor block, the west façade remained open and the track of the travelling crane, capable of handling a load of 260 tons during the work site phase (150 tons in the operating phase), was extended by an external pier so that the large parts (vessel, slab and so on) could be introduced into the building comfortably.

The **other buildings**, the superstructures of which are composed of steel frames, were built at the same time. The turbine hall is identical to that of EDF's conventional power plants of the 250 MWe series. As for the warehouse and maintenance building, it was made large in size so that it could be used as an

assembly area for the large factory-made and mechano-welded sub-systems.

The only major anomaly was observed in November 1972. The water pipes bringing water from the Rhône to cool the condenser were cracked and all the manholes that made it possible to inspect them from the river were broken. The cause of the damage was found to be the repeated, unscheduled closure of the free valves on the suction baskets of the circulating pumps. The valves were removed and the pipes repaired with no major effect on the schedule of the other work and tests, despite the delicate nature of this operation in the bed of the Rhône.

The number of people working on the site reached a relatively modest maximum of 700. This number evolved, however, in a regular manner throughout the works period. The arrival of the first deliveries of sodium, and putting their circuits and equipment into service in mid-1971 corresponded to both the height of activity on the site and the start of the operation period. For the first time, it was necessary to implement the operating regulations and instructions whose aim was to impose a collective way of acting that would be very different from the standard way of acting on a construction site.

1.5. The construction

The **large vessels**^[7] were manufactured by Neyrpic and Creusot-Loire in an on site workshop, the future warehouse and maintenance building, which was ready as early as October 1969, making possible the construction of this sensitive equipment without too much strain on the timetable. The distribution of the pre-fabricated elements and equipment was studied in detail between the factories and workshops on site. The operating methods used on site were as mechanical as possible.

[7] See the diagram of the reactor block in the appendix, page 203.

Visits by the Project Manager

Everything must be perfectly clean on the worksite, particularly for mounting in the reactor block. Before each monthly meeting, the Project Manager used to pay a long visit to the worksite, most often on his own, to see with his own eyes how the work was progressing. During the assembly of the reactor block, and after the diagrid had been installed in the main vessel, gaining access to the bottom of the vessel, under the diagrid, while still possible, resembled "potholing in white coats" more than anything. When he was visiting the bottom of the vessel, the Project Manager stopped short at the sight of a nice little round pebble, right in the centre. Given the layout of the place, it was difficult to imagine that it had got there on its own or by mistake! He made a quick decision: take the pebble and... keep mum!

Two months...and two pebbles of increasing size later, the last pebble was discretely placed on the desk of the worksite engineer responsible for the reactor block, with a ribbon around... In the afternoon, behaving as if nothing had happened, the engineer was grinning from ear to ear!





The primary containment enters the reactor building

Considerable handling means were employed. Particular care was taken with surveillance, checks and verification of assemblies, including with the presence, unusual at this time, of the contracting authority's inspectors at the manufacturers'. Works progress and quality control operations were checked constantly, thus making it possible to stick to the timetable and guarantee harmonious progress, despite a certain number of setbacks, inevitable in a project of such scale.

The primary vessel was made of carbon steel and delivered to Marcoule as formed metal sheets presented for blank assembly in the workshop prior to delivery. The vessel was mounted in 18 mm thick conical shells. These shells were then stacked by manual welding. The twelve cooling sub-circuits, with a total length of 4,300 meters of pipes, were welded on to the outside. After checking the welds and leaktightness, the vessel was ready in April 1970. It was installed in the reactor building on November 2, 1970, in an operation requiring many delicate maneuvers because of the mass of the vessel (130 tons) and the narrowness of the gaps in the concrete pits into which it was placed on jacks.

The other vessels (safety, main and primary) and the internal structures of the reactor block were made from stainless steel. They were prefabricated in factories using 15 to 25 mm thick metal sheets, in a manner that was more advanced than for the primary vessel. At every stage of the process, extremely precise checks were made on the dimensions making it possible to reproduce with precision the limits set on the plans. Dye check and radiographic checks were made, as well as leaktightness tests using a suction-grip. After a final cleaning *in situ*, the safety and main vessels (with the matting) were installed in their positions in November 1970. Precise topographic checks were made after each installation so that the interconnection of the successive tolerances (including the building's future settling) and possible settings made it possible to produce a correct sub-assembly.

Inside the main vessel, the support for the lateral matting support was installed, then the diagrid with its inlet nozzles was positioned on the stellite coated plates wedged on the matting. Finally, the fully mounted primary vessel was descended into the reactor block and welded on to the lateral shielding support.

The **roof** of the main vessel was another large-sized structure, 60 mm thick. This is because it drives the mobile components during dilatation movements. It was prefabricated in a factory in three parts (a semi-circle, a 28° sector and its complement to make the second semi-circle), which were equipped with the circular penetration and connection shells of the main vessel. Assembly of such a large part requires, for example, the use of special reversing tools to make possible symmetrical welding all over the metal sheet. The roof was installed on August 2, 1971, and was then welded to the shell of the main vessel on to which was also connected the safety vessel.



Installing the primary vessel

The slab was a mechano-welded unit 15 meters in diameter and 1.5 meter high. It was assembled from six prefabricated elements. The metal parts alone weighed 200 tons and the slab was installed on September 1, 1971, on 22 ball-and-socket type pads, and then 600 tons of concrete were poured inside. The vessel's connection support was installed and the upper hangers attached to it. The reactor's vessel was thus suspended from the slab. The next stage consisted of jacking up the primary vessel and connecting it in turn to the slab.

The six elements of the lateral matting support were introduced into the reactor through the opening of the rotating plug. The lateral neutron shielding billets were implanted into the lateral shielding support, and then the steel, breeder and dummy sub-assemblies (steel sub-assemblies that reproduce the size and hydraulic characteristics of fuel sub-assemblies) were handled in ambient air in order to be plugged into the diagrid. This thus resulted in a "dummy core" that would be used for the inactive tests in air, and then in sodium before fuelling the reactor.

During this period, the real fuel sub-assemblies were manufactured in Cadarache and were

gradually delivered to the site. The rotating plug, propped against its ball roller raceway, closed the reactor in a leakproof manner thanks to its fusible metal seal. The installation of the core cover plug and the rod mechanisms completed the mechanical assembly of the reactor, which was ready to be preheated prior to being filled with sodium at the end of 1972.

The **main components** (primary and secondary pumps, intermediate heat exchangers, steam generators and so on) were factory-made (Hispano-Suiza, Stein Industrie, ...) and deliv-

Installing the workshops

In 1971, the first operators arrived from the various units of the CEA and EDF, mainly from the G2 and G3 reactors in Marcoule. The future workshops of the Phénix plant had not yet been built. The present maintenance building was essentially composed of a huge hall for assembling the large elements and could not at that point be used by the newly-arrived staff of the power plant. The staff were thus housed exclusively in the control-office building. Every month, new employees arrived and the lack of space began to be a major problem. The improvised offices were moved gradually, as each definitive structure was installed.

The technical staff housed temporarily in this single building were soon confronted with major administrative difficulties: almost no tables for studying plans and documents, few chairs, no tools, even less equipment for measures or checks... In short, there was a chronic lack of everything. What was needed was resourcefulness. Everyone, with more or less success, contacted their former workplace in Marcoule to pick up abandoned desks and chairs, pens and rulers, screwdrivers and monkey wrenches, small electronic components, pipe connections and so on.

Soon, the Marcoule employees, some of whom were amused, others less so, welcomed their former colleagues with a sardonic smile and remarks of the "*nail your things to the floor, the Phénix boys are on their way!*" Their precious assistance and patience nevertheless made it possible for the Phénix workshops to take shape. Little by little, working conditions improved, allowing everyone to play a role in the adventure of operating the plant.



ered to the site for installation. Prototypes for each piece of sensitive equipment were tested in conditions as representative as possible of the plant's future operation, including a certain number in sodium filled rigs in Cadarache. Only drilling the tube plate of the intermediate heat exchangers gave the manufacturer serious problems that could have proved insurmountable (the drill bit deviated beyond specifications). The manufacturer nevertheless managed to solve the problems once the CEA had demonstrated the feasibility of the operations requested.

The mechanical, electromechanical, electrical, electronic and so on assemblies all proceeded without incident. The turbo-generator set was installed by the Compagnie Électro-Mécanique. The secondary cooling circuits were supplied jointly by Stein Industrie and Babcock Atlantique. The Ateliers et Chantiers de Bretagne built the handling line for irradiated sub-assemblies. The sodium (manufactured by Ugine Kuhlmann) was delivered to the site between July 1971 and February 1973, in 20 ton tanks. The sodium was so-called nuclear-quality sodium, the impurity level of which is particularly low so as to prevent pollution of the cold traps during filling procedures and the first temperature build-ups. The sodium was stored in drainage tanks (primary and secondary) until the cooling circuits were ready to be filled.

The construction of a large, dense unit that uses concrete, sheet metal work, electrical equipment and fine mechanics simultaneously requires strict **organization** and timetabling. The very short deadlines imposed on the construction of the Phénix plant were obtained by drawing up a timetable that allowed as many tasks as possible to overlap, as well as the intervention of various professions simultaneously. These deadlines were respected. For the first time at the CEA, a timetable was drawn-up on the basis of a simplified PERT schedule (without

the financial resources), specially developed for the occasion.

Despite the publicity surrounding the Phénix project, several technological aspects were kept secret. These essentially concerned the fuel used, on the basis of the irradiations conducted in the Rapsodie reactor. In particular, it was the radial gap between the mox pellets and the cladding of the fuel (to prevent any interaction between the two materials during the power deformations, whilst simultaneously encouraging the evacuation of the fuel's heat) and the oxide density (high density pellets turning out to behave better than low density ones).

1.6. The tests

The importance of the start up tests for Phénix was the result of the prototype nature of the power plant and its size. The aim of these tests was to guarantee progressive operation whilst minimizing the risks inherent to a prototype situation. The complexity of the tests, the desire to use as many skills as possible and the need for as progressive a transition as possible between the construction and operation phases led to a special structure being developed in 1970 and implemented in 1971. This structure was to be used for the start up tests. The process took place over 3 years. The program was irregular, it is true, but a duration of this length is indicative of the importance given to this vital aspect of the construction.

The tests were grouped together in four phases, determined by the following important events: putting the sodium into the main cooling circuits, fuelling the core, and the start of the power build-up.

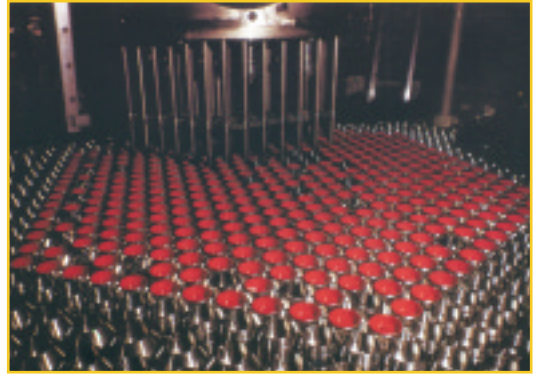
The **first phase of the tests** was a verification of all the general auxiliaries and associated circuits. All these circuits had to be in good working order before the sodium could be placed in the main cooling systems. While the reactor was being flushed with nitrogen before being pre-



heated and filled with sodium, an oil valve on the temporary preheating circuit drained into the reactor. It was necessary to return to air and intervene in order to recover with a cloth most of the twenty-five liters of oil that had soiled the dummy sub-assemblies, diagrid, matting and bottom of the vessel. At the same time, tests were conducted at Cadarache which revealed that any traces of oil that remain would be decomposed by the sodium at a high temperature.

The power plant's staff was integrated into the test teams. At the end of each series of tests, operating notes were immediately given to the operators, allowing them to draw up a variety of instructions, produce booklets of diagrams and verify the facility in practical terms.

Although the tests were approximately five months behind the original 1969 schedule, and despite the fact that any time that could be saved was done so with determination, the tests for permuting an intermediate heat exchanger with the component handling cask were maintained. There was nevertheless a heated debate between those skeptical of the utility of such tests, and those who thought it better to do this



The mock core

handling once in the inactive stage rather than later, under pressure of an incident, with an active pump or exchanger. During the tests, many adjustments were made. They paid off a few years later (see § 2.4) when the intermediate heat exchangers were permuted without difficulty.

The **second phase of the tests** started at the end of 1972 when the storage drum, sodium cooling circuits and reactor block were filled. It continued until the end of July 1973, and was composed of tests with isothermic and inactive sodium. The temperature of the sodium was maintained by the pumping power. As soon as the sodium was first put into the secondary cooling circuits, small leaks were observed at the level of the drainage valves, at the place where their joints had not been definitively welded because of modifications planned for the future. Anomalies were then observed in the displacement of the pipes in the secondary sodium cooling systems. Throughout January 1973, while the reactor was being filled with sodium, the supports for these pipes were adjusted and the drainage valves modified.

The first temperature build-up to 450 °C was done in March. It made it possible to purify the sodium and, in a more general manner, examine how the materials functioned in terms of flow and temperature, in order to detect any

We all pull together

One Monday morning, during a weekly meeting for the tests: "This weekend, the oil from the valve of the reactor block was aspirated. It's a catastrophe! Who did it?". The person responsible for the reactor block tests answered immediately, "It was me". As everyone present knew full well that the engineer in question had been to Paris for the weekend, the question was asked again and again, but always with the same answer.

A plan of action to resolve the problem as quickly as usual was soon developed and approved. But, over the weeks that followed, the question, "Who did it?" came up repeatedly... and the answer was always the same. Despite requests for investigations, the truth never came out and the directors had to get used to the idea of the solidarity and friendship present in the team.

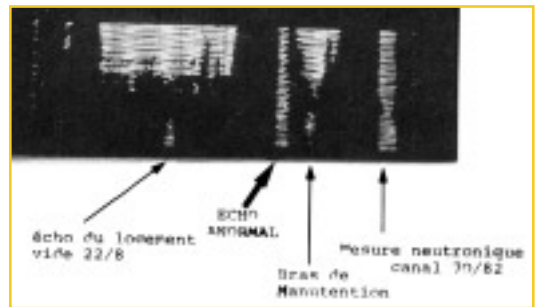


problems as early as possible. The pumps were seen to be functioning satisfactorily and no abnormal vibrations caused by the considerable number of instruments used (accelerometers and strain gauges) were detected in the structures of the reactor. The only dysfunction was found in the check valves of the primary pumps: they did not close again after the pumps were stopped when running at slow speed. This observation made it possible to modify the valves and test the first modified valve installed on a primary pump as early as July.

At the end of March 1973, after completion of the tests on the isolation valves for the steam generators, a pressure surge on the pipes, caused by the particular conditions of the tests, burst the diaphragm that protected against overpressure in case of a water-sodium reaction in the superheater of steam generator N. 3. After draining the associated secondary cooling circuit, the sodium that had penetrated the separator tank was transferred into the drainage tank, and then any residual sodium was cleaned from the bottom of the separator tank. Ten protection diaphragms for the steam generator were changed, although only three of them had been punctured or deformed.

Several different temperature levels made it possible to test the detection of burst cladding or hydrogen in the steam generators, the measurement of pressure and flow, as well as very complete tests on fuel handling at 250 °C. The rotating plug was successfully installed in April 1973, followed by tests for the development of Visus. This is an ultrasonic visualization device, making it possible to detect, under the level of sodium, the presence of any object liable to block the rotation of the rotating plug or the handling arm. This device had not yet been tested in a large-sized vessel, and it took several days to adjust the electronics to the environment of the reactor block by interpreting the signals.

The **third phase of the tests** started when the dummy sub-assemblies were replaced with fuel sub-assemblies. Loading took place progressively, starting from the center. It was interrupted when obstacles were observed on the images delivered by the Visus (visualization in sodium ultrasonic device): echos, characteristic in appearance of one or more sub-assemblies raised above the network, appeared on the control monitors. The handling procedure did not allow operations to continue while they were still present.



Phantom echo in the reactor

After a considerable number of control operations, and in the hope of the disappearance of the echos, in vain, the decision was made to lower the sodium level in order to perform a periscope examination. Part of the primary sodium was thus transferred into the drainage tanks. It was then observed that the echo did not in fact correspond to any real object. The handling operations could thus restart. The phenomenon was later explained and an electronic device was set up to eliminate the problem. These “phantom” echos were caused by the multiple reflections on the walls of the vessel for the particular positions of the Visus, enhanced by a particularly high transmission coefficient for the ultrasound waves, in turn caused by the high level of purity of the sodium and the low level of argon emulsion in the sodium.

A first batch of 46 sub-assemblies was loaded into the reactor, followed by a second of 12 sub-assemblies. The neutron counting was



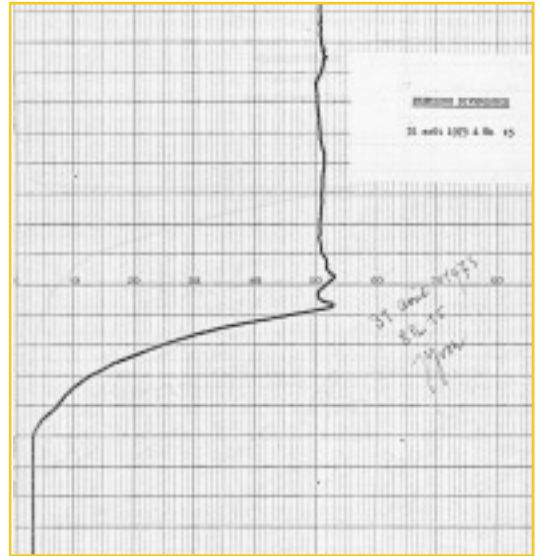
then performed each time a new batch of 6 sub-assemblies was loaded. Problems concerning interpretation of evolution in the counting rate appeared in the subcritical run-up^[8]: the neutron measuring chambers, situated on average one meter from the periphery of the core, were more sensitive to the geometric effects (site of a new sub-assembly in relation to the sources and measuring chambers) than to those of reactivity. Conducting special simulations, plus the installation of a special counter under the vessel, made it possible to check the corrections to be made to the measurements. The seventh, and final, batch was composed of 9 sub-assemblies. Once loaded, the critical mass of the core was thus estimated at 87.5 sub-assemblies.

In parallel, and in a relatively independent manner, the tests on the electricity generating system started in 1972. Using a temporary boiler made it possible to launch the turbine and perform an initial connection of the alternator to the grid, at a low power level, as early as February 1973. The feed water and condenser part was tested as completely as possible in its various configurations, and endurance testing was carried out with the condenser under vacuum.

1.7. Commissioning

The **first criticality** took place on August 31, 1973 at 8:15 am, with a core containing 91 sub-assemblies. The day before, there was a firework display from the banks of the Rhône that all the staff on duty that night were able to enjoy from the corridor that runs along the control room. The criticality took place in the presence of the High Commissioner for Atomic Energy and a large number of directors of the CEA.

By then, the core had fully reached its definitive size. Low power neutron tests were conducted



First criticality

throughout September: reactivity testing of the control rods and sub-assemblies, measurement of the distribution of the neutron flux, reaction rates, temperature coefficients and so on. The characteristics of the core were shown to be exactly conform to those established in the calculations made two years earlier.

Several short duration (one or two hour) irradiations were performed in order to determine the distribution of power and flux in the core, breeder blankets and neutron shieldings. Several hundred detectors (^{235}U , ^{238}U , Pu, Au, ...) were arranged inside the rigs introduced into experimental sub-assemblies which were loaded in a variety of positions in the core. Once the irradiation was completed and the rigs recovered, the detectors were sent to Cadarache to be counted on the Masurca and Harmonie facilities.

Double irradiation of the same detectors was also conducted in the Phénix reactor and the Harmonie reactor in Cadarache. Controlled by telephone, the criticality, power values and shut-down at the end of the irradiation were performed simultaneously on both reactors to the great satisfaction of the physicists who were

[8] The subcritical run-up consists of taking successive readings of the reactivity of the core by progressively lifting the control rods in order to slowly approach the critical elevation, to record it with increasing precision. When the reactor was loaded for the first time, this operation was performed each time a batch of sub-assemblies was introduced.



thus able to compare the results and make any adjustments necessary in their computer codes for fast neutron reactor cores.

The steam generators which had, up until then, been tested either in sodium or in water, were put into service with sodium and water simultaneously on October 15, 1973. The signals for the detection of hydrogen were monitored particularly closely but did not show any significant variations.

The power build-up proper was the fourth phase of the opening tests. It was started in November 1973. It was conducted with a double aim: to move forward in stages, permanently monitoring the conditions required for nuclear safety, and to obtain, as quickly as possible, an operating period of significant length under conditions similar to those of the rated capacity.

The fast breeder

"This year, the Atomic Energy Commission and Électricité de France have installed Phénix, a prototype of great power, in Marcoule on the Rhône. Phénix, the fabulous mythological bird that rises from its ashes, will provide 250 megawatts.

This nuclear power station is the cream of French technology. It is the first in a program of 8,000 megawatts planned for the VIIth plan. It will make it possible to partially satisfy requirements in electrical energy, forecast by EDF at 200 billion kilowatts - hour by 1975.

With the generation of 40 billion kilowatts - hour of nuclear energy, France, with its large reserves of uranium, is the third power in the world after Great Britain and the United States.

Thanks to its leading techniques, our country can hope to free itself in the years to come of its unfortunate dependence on foreign countries for its energy supplies, as underlined by the recent oil crisis."

Card accompanying the 1st day edition of the "Phénix fast breeder" stamp, September 21, 1974

The **first connection** to the electricity grid took place on December 13, 1973 at 5 pm^[9] in the presence of the French Minister for Industry (who pressed the button on the telephone network to immortalize the event for the press cameras while the machine operator did the real work discreetly in the background), the General Secretary for Energy, the General Administrator of the CEA, the President and Managing Director of EDF and many other personalities.

A test of natural convection for the primary and secondary sodium circuits was conducted. After several preliminary tests, all the pumps were shutdown while the thermal power of the reactor was maintained at 4 MWth by positioning the control rods adequately, thus making it possible to simulate residual power. A stable operating regime in natural convection was observed after only a few minutes, first in the primary circuit and then in the secondary circuits. The results obtained made it possible to check the calculation programs developed to demonstrate the reactor's safety.

Lack of tension was provoked deliberately at different temperatures in order to confirm the behavior of the equipment. The systems for detecting and locating clad failures were calibrated by means of an experimental sub-assembly with a surface of bare uranium, introduced into the core especially for this operation. The variations in the counting rate were studied in relation to temperature and power in order to establish the background noise diagram and to specify the sensitivity of the detection.

Automatic shutdowns were also provoked as a means of analyzing the transient corresponding to the different power levels in the reactor. After

[9] This was in fact the third connection as the first two took place on 11 December at 9:42 pm and 12 December to check that everything would run smoothly on the appointed day. The turbo-generator set thus remained connected to the electricity grid for respectively half an hour and five and three quarter hours.





one such shutdown, it was observed that three control rod mechanisms of the same type (without leaktightness bellows) had stopped before reaching their lowest position. When these mechanisms were dismantled, it was noticed that a deposit of sodium compounds had formed in the place of the free level of sodium. This deposit was in fact the impurities conveyed by the flushing argon which had been caught in this spot. The flow of the argon injection was reduced to limit the input of impurities and to increase the free level of sodium in the sheath of the rod mechanism in a zone where the gap is more significant.

By going through successive levels, the **nominal power** was reached on March 12, 1974. The main operating parameters for the reactor were perfectly comparable with those in the pre-project. Additional tests in operation then continued until the month of June. This long and complex final test phase made it possible to acquire good knowledge of the operating regimes possible and the actions to perform in case of incident. The abnormal operating regimes were provoked and controlled. The safety actions, either automatic or manual, were tested for each configuration and the corresponding instructions were written up from experience. The tests of operation with two primary pumps or with two secondary circuits, revealed the considerable possibilities of an integrated primary circuit under asymmetric conditions.

After a month of shutdown for works, modifications and definitive configuring, the power plant was declared fit for service in industrial operations. The start-up took place on July 12, with the power build-up on July 14, 1974. The industrial commissioning was declared the same day, thus officializing the transfer of responsibility for the power plant from the project manager to the manager of the power plant.

1.8. Ready for operating

Throughout the pre-project and project stages, a number of problems were encountered and resolved in time, to a great extent because of the innovative nature of the work, particularly:

- extrapolation from Rapsodie, a 40 MWth loop reactor, to 563 MWth for Phénix, with (a new concept) the primary circuit integrated into the main vessel,
- mastery of sodium technology at a scale of hundreds of tons, of neutronics in a large fast neutron core and its thermohydraulics, the design and manufacture of the fuels, and so on,
- the safety studies specific to fast neutron reactors, in particular for defining the maximum accident admissible and the size of the reactor block needed to confine it,
- the studies, testing and production of large components, all prototypes, mastery of the materials used and the thermomechanical stresses and so on.

A certain number of new problems almost significantly disrupted the good running of the project. When, during operation of Rapsodie, swelling was observed in the stainless steel cladding of the fuel pins in neutron irradiation, obtaining the targeted fuel burn-up was put seriously at risk. Similarly, when the tubular plates of the intermediate heat exchangers were bored there was, for a time, a serious risk that the schedule would not be respected.



The construction work of the Phénix plant was conducted briskly. The first stages started in November 1968. Five years later, the power plant was connected to the electricity grid. The initial timetable forecast the first criticality for April 1, 1973. In reality, it took place on August 31. But this was soon made up thanks to the swift, and almost unhindered, running of the power build-up tests. The nominal power was forecast for April 1, 1974 and was achieved, in reality, 18 days ahead of schedule. The combined work effort of the CEA, EDF and manufacturers made it possible to associate rigor and flexibility. This was a key element in the success of this construction work and its respect of both costs and timetables. The technical modifications made during the study and production phase increased the initial budget by less than 10 %.

This success won its chief architects honors on the occasion of a special promotion in the order of the Legion of Honor and in that of Merit decided by the President of the Republic. Abroad, the start-up of the Phénix

plant (which occurred before that of the British Prototype Fast Reactor, despite its construction starting two years earlier) was also praised: "French lead world in fast reactor technology"^[10].

The Phénix plant had all the technical characteristics determined for the project. It was ready to start running in order to generate electricity and, above all, to provide key information for the future of the fast neutron reactor field: behavior of the fuel, different materials, operation of the circuits and so on.



[10] Title of an article in the Financial Times, dated 15 March 1974.



Jean Mégy

Deputy, then director of reactor construction department and Phénix project manager from April 1968 to July 1975

From design to commissioning

With hindsight, one can see the Phénix project from the design stage through to commercial operation as a fine technical success in a new field and above all, for those who lived through it, a remarkable human adventure. Around the engineering team in charge of building the plant there were teams from the CEA, EDF and the manufacturers involved, all working with a cohesion, rigour and enthusiasm that has left nostalgic memories for those who experienced it.

True, during the preliminary design and the project itself numerous difficulties cropped up and were solved in due time. Some new problems almost called the continuation of the project into question: for example, during the operation of Rapsodie it was found that the stainless steel fuel pin cladding swelled under neutron bombardment – a major threat to achieving the intended burn-up rate.

All in all, we kept to schedule. But rather than the technical aspects, which are well-known, it is worth looking at how the different teams involved were organised.

Right from the preliminary design (1966-1968), the engineering teams who were later to form the Phénix project team were involved in the design work. That work was mainly done at Cadarache and Saclay, using a strong R&D structure:

- labs and test halls for the technology of the sodium and components,
- the Rapsodie reactor and the Harmonie and Masurca experimental reactors for neutron research,
- fuel development units and plutonium fuel fabrication facility,

and the links with these units continued during construction, testing and industrial operation - close, efficient links with very fast responses to solve problems as they arose.



One vital decision taken in 1968 had very positive consequences: the decision to have a single engineering team handle the tasks of both owner and prime contractor. This was the "integrated team" made up of staff from CEA, EDF and GAAA (the Groupement Atomique Alsacienne Atlantique, which had already acted as industrial architect for earlier reactors, particularly Rhapsodie and Célestin).

The team's engineers and technicians were chosen for the experience they had acquired on Rhapsodie, G2, G3 and Célestin, and most of them had already worked together at some point. A representative of the department in charge of safety was allocated to the project full time and associated with all the activities in real time; in this way the opinions of the appropriate experts could be taken into account immediately. The Phénix project's "integrated team" was placed under the leadership of the CEA project leader. Having a single authority did much to ensure quick decision making and smooth management of interface problems.

Furthermore, to facilitate the transfer of the knowledge and experience acquired with Phénix to the Superphénix project and, conversely, to make sure the Phénix operator had as much support as possible from those who had helped build it and had since moved on to Superphénix, the single-authority system was maintained until the end of 1977: the operator of the Phénix plant and the new Superphénix project were both under the responsibility of the Superphénix project director, who had formerly been director of the Phénix project. This particularly facilitated interventions following the sodium leak on an intermediate heat exchanger in July 1976.

The initial management team was set up in 1968 and was involved in the project research. The full operating team was formed gradually between then and the trials, which it implemented in line with the programmes and directives, under the supervision of the project's test groups.

The manufacturers were also involved very early on, under study contracts. This way the project benefited from their particular experience, especially for drawing up the technical specifications, and the suppliers became responsible partners and not just manufacturers. There was a twofold purpose here: efficiency for the project in hand and preparation of a competent, specialised industrial group for subsequent developments.



As often as possible, the main components underwent tests on full-scale prototypes: the main pump on a hydraulic loop, handling device on a hot argon rig, valves, control mechanisms, sensors, etc. in sodium loop. This minimised adjustments during reactor tests and in operation.

Particularly close attention was paid to planning, so as to manage the sequence of tasks as well as possible. A programme derived from the American PERT was specially developed for the project's particular needs. The initial analysis of tasks and the permanent monitoring of the critical path and several sub-critical paths made it possible to integrate, optimally and in real time, the divergences from predictions due to the hazards that are unavoidable in a prototype.

Preparation of the start-up and power build-up tests began in March 1970 and a special organisation was set up for this, using the same methods. In particular, managing the test programme involved organising systematic validation by all involved (research and engineering departments, suppliers, operator etc.). This rigour, combined with the excellent training the operating team received, probably largely explains why the power build-up tests were completed faster than expected and without any major hitch.

That was quality assurance - before the EDF's Equipment division formalised quality assurance for building nuclear power plants for the large French program.

But first and foremost, for several years it meant mobilising a group of competent and highly motivated professionals who were aware of the stakes and convinced that success in this ambitious project was essential.



THE EARLY YEARS
(1974 - 1980)



Landmarks

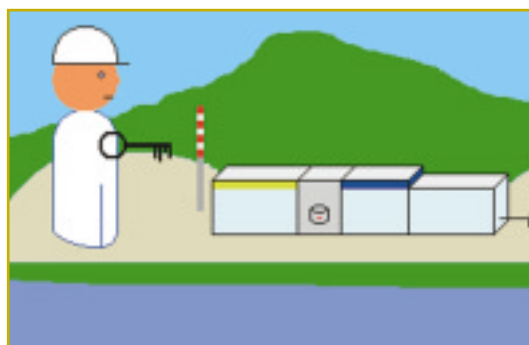
19 September 1974	First spent sub-assembly dismantled
11 December 1974	Tests with reactor at maximum power (597 MWt)
29 January 1975	Net cumulative electrical output: 1 billion kWh
3 June 1975	First leak from fuel pin cladding
18 August - 24 September 1975	Initial revision
13 October 1975	Net cumulative electrical output: 2 billion kWh
May 1976	High Activity Oxide (HAO) unit starts up at the La Hague facility
1 June 1976	Responsibility for the Marcoule site transferred from CEA to Cogéma
11 July 1976	Secondary sodium leak from intermediate heat exchanger E
3 October 1976	Secondary sodium leak from intermediate heat exchanger F
7 March 1977	First divergence of Fessenheim 1 (900 MWe PWR)
23 March 1977	Definitive shutdown of DFR (United Kingdom)
24 April 1977	First divergence of Joyo (Japan)
12 May 1977	Government issues decree authorising creation of Creys-Malville plant (Superphénix reactor)
31 August 1977	Secondary sodium leak from intermediate heat exchanger B
27 June 1978	High-activity waste vitrification facility opens at Marcoule
19 December 1978	General power cut throughout France
16 February 1979	Net cumulative electrical output: 5 billion kWh
13 January – 18 March 1979	Cycle running uninterrupted (66 EFPD in 64 days)
9 February 1979	Eurodif enrichment plant starts up
28 March 1979	Three Mile Island accident (Pennsylvania, USA)
1 May 1979	First cladding failure
1 February 1980	Definitive shutdown of G2 reactor at Marcoule
9 February 1980	First divergence of FFTF at Hanford (USA)
21 February 1980	First divergence of Tricastin 1 and Gravelines 1 (900 MWe series)
26 February 1980	First divergence of BN 600 (USSR)
28 March – 23 June 1980	First ten-yearly regulatory inspection and overhaul



Generally speaking, operating a prototype over a long period of time is not an easy task. The plant operation team was gradually formed towards the end of construction work. This team had to learn everything about this special reactor boasting no equivalent, except the English Prototype Fast Reactor built on the Dounreay site in the North of Scotland at the same time, but the atmosphere was more competitive than collaborative. Therefore, competent team members - assisted by researchers from Cadarache and engineers assigned to the Superphénix project - had to be drawn from its own ranks in an attempt to confirm the promising results obtained during the commissioning test period.

It was clear that the Phénix plant was built to demonstrate the facility's overall capacity of operating over time while meeting expected characteristics. Of course, various unforeseen obstacles were to complicate matters, forcing the team to find other alternatives to those retained during the project design phase. Being a demonstration reactor of what was meant to become a new reactor technology - the sodium-cooled fast neutron reactor - operation data was to be collected to serve teams working in parallel to the project and the construction of the following reactor, Superphénix.

After having reviewed the early operating years of the Phénix plant, this chapter will then be devoted to discussing the main results, as



well as unforeseen hazards that not only occurred in the core, the reactor - particularly intermediate heat exchanger leaks - the fuel sub-assemblies and components handling, the steam generators and the the electricity generating system. Discussion of the first ten-yearly regulatory inspection and overhaul will then conclude this chapter.

2.1. The early stages of operation

As early as 1970, the plant operation team was progressively formed. The total number of staff was set at approximately 220 employees, including about twenty section managers and engineers, with hiring taking place essentially between 1971 and 1972. Engineers with experience in reactor operation - such as G2, G3, Celestin 1 & 2 in Marcoule and Rapsodie at Cadarache - were particularly called upon for key positions. Besides an administrative group and a security cell (in connection with the radiation protection unit belonging to the Centre de Marcoule), activities were organised into four sections:

- reactor operation, involving six shift teams (7 members per team) working under a engineer on duty whose position is shared in turns by all engineers working in this section ; during a shift, each team is responsible for the facility (on behalf of the plant manager) therefore for all associated activities, including their own specific tasks,
- handling activities, involving all fuel sub-assembly movement, "special" handling of large reactor components and the operation of irradiated sub-assembly dismantling and examination cells; all handling personnel alternates between periods of normal working hours and shift working hours (during reactor refuelling in particular),



- maintenance, involving the management and co-ordination of activities concerning plant materials on the one hand and operations in fields such as mechanics, sheet metal work, electricity, electronics and reactor instrumentation and control on the other hand,
- physics, itself broken down into three groups: core physics, irradiation & chemical analysis and tests & statistics.



The hall of the reactor building

It is difficult to distinguish the construction and test phase from the actual operation phase. The shift from the first phase to the second was essentially felt when the project staff handed down responsibilities to the operating staff. The **first two years of operation** were characterised by an overall smooth running of the plant, which is remarkable for any prototype at the beginning of its life. Such results were obviously due to the conditions under which the plant had previously been built and tested.

More specifically, a load factor of 80% was reached during the first year, before being reduced to 70%, which was equivalent to a plant

availability of 75%, excepting the time strictly necessary for fuel reloading. November 1974 could have been the first month of continuous operation at rated power, however a strike by the shift personnel reduced the load factor to 60% for forty eight hours.

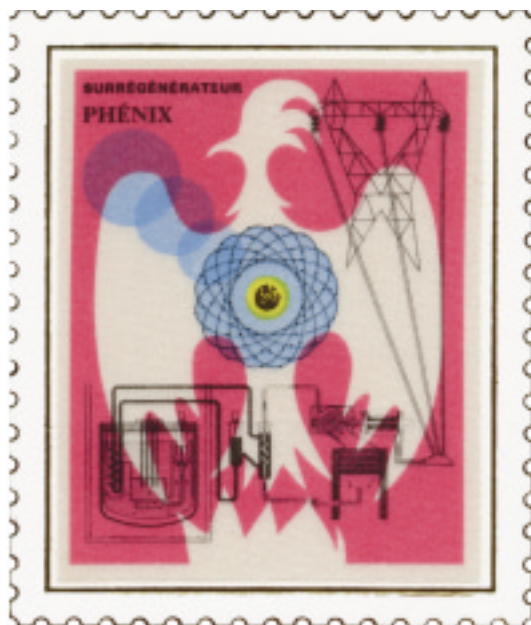
During this period, knowledge of the facility was developed and operating parameters were revised to increase outputs significantly in comparison to the initial project parameters. Thus, it became possible to generate a gross electric output of 265 MWe in December 1974 instead of the initially programmed 250 MWe. A 45% gross electric output was the best score held by any power plant of this generation, regardless of the plant's fuelling system (coal, petrol or nuclear).

Furthermore, plant operation teams were already able to refer to a complete set of instructions that had been tested during start-up trials, before being perfectly implemented during transients occurring during plant operations. For example, on March 13, 1975, a sodium leak was signalled by one of the detection beaded wires equipping all piping systems. Monitoring levels were stepped up and the following morning, a small whitish residue appeared at the joints of insulation sheets. The reactor was shut down and the system was drained. The insulation around the piping system was then dismantled for examination.

Qualification of pressure cookers

A well-known brand of easy-to-use, leakproof, stainless steel pressure cookers is used to store the shavings of solid sodium before they are destroyed in the washing pits. In order to validate their use, two tests, using an intelligently combined devices, were carried out in 1974. The tests simulated accidental flooding conditions or an introduction of water into the hermetically-sealed pressure cooker, even though such a situation is difficult to imagine. It was thus proved that the pressure cookers were both perfectly adapted and totally safe for their rather unusual function in the Phénix plant.





In the light of the delay due to repairs, the reactor was restarted with two secondary cooling circuits in operation, before being connected to the grid less than twenty four hours following the shut-down. Another leak was detected on July 16 the same year in the same area (cf. § 2.3). This time, the turbo-generator set was stopped even though the reactor remained at low power while the system was drained, with the loss in electric power generation lasting only twelve hours. Beyond the anecdote, these two examples are characteristic of the operational flexibility of this reactor.

During the programmed shutdown from mid-august to mid-september 1975 for complete inspection of the turbo-generator set after a year of operation, time was also taken to inspect other vital equipment and components in both the electricity generating system and the plant itself. All inspection results were satisfactory. Several improvements to plant operations were also made following analysis of test and operational feedback.

Leakage detected in the intermediate heat exchangers (cf. § 2.4) led to lengthy periods dur-

ing which the plant was operated at two-thirds of its rated power, which confirmed the possibility of operating the Phénix plant using only two of the three secondary cooling circuits without any restrictions whatsoever.

From April 1978 to March 1980, the plant once again experienced two years of **continuous operation at full-power** without any noteworthy incidents occurring. The load factor was increased dramatically, rising above 80% during this period (84.8% in 1979). In February 1978, nothing prevented the plant from operating normally, limited nevertheless to 164 MWe due to the unavailability of the secondary cooling circuit N. 1 (including the intermediate heat exchangers under repair). Despite the general power failure of the French electric grid on December 19, 1978, at the end of which the Phénix plant was the first to be reconnected to the grid, the plant still managed to set a record in net electrical output (175,277,000 kWh).

In February 1979, the plant operated at full power - 260 MWe - for an entire month non-stop. The 15th cycle was completed without any noteworthy difficulties from January 13 to March 18, 1979, equivalent to 64 days. However, the first cladding failure in the plant occurred on May 1, 1979 (cf. § 2.2). This incident rendered the plant unavailable for three days only. In terms of electrical output, the Phénix plant definitely gave its best performance in 1979, generating 1.7 billion kWh.

Before the plant was shut down on March 28, 1980 for its first ten-yearly statutory inspection and overhaul, after six and a half years of operation, the reactor had already generated 718,725 MWd (equivalent to 1,277 effective full nominal power days), not to mention more than 6.9 billion kWh^[1] provided to the electric grid (as well as 13 million kWh of steam used to power the

[1] Equivalent to the electricity consumption of a town the size of Montpellier (210,000 inhabitants) or a department such as Ardèche (290,000 inhabitants) over 7 years.



Marcoule site). The plant's rate of availability was equivalent to 63%. Its average yield reached 43% despite periods at intermediate power rates during which operation was not optimal.

2.2. Reactor core

The initial fuel composition was a combination of 50% mixed uranium-plutonium oxide and 50% enriched uranium oxide, as the stock of available plutonium at the time of building the first core was insufficient. During the first four years of operation, the proportion of mixed oxide was progressively increased with each refuelling, until it represented the totality of the core fuel. The plutonium content in the core was equivalent to 18% in the central region and 25% in periphery regions.

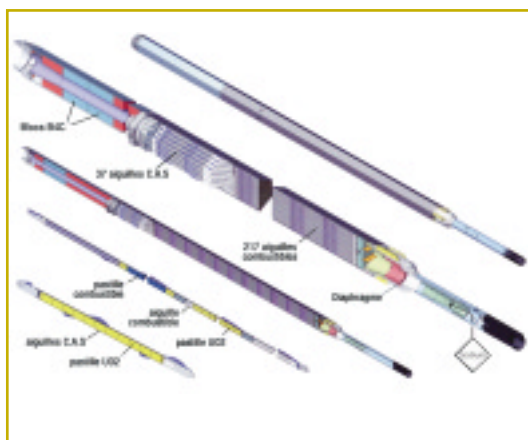
The good resistance behaviour of the fuel made it possible, even during the first few cycles, to notably increase its specific burn-up, which was originally set at 50,000 MWd/t^[2]. At the end of the second year, the specific burn-up authorised for standard fuel sub-

assemblies was increased to 75,000 MWd/t. The maximum admissible specific burn-up for experimental fuel sub-assemblies reached 72,200 MWd/t and 79,500 MWd/t respectively in March 1978 and March 1980. The enrichment of the core's inner zone was also increased by approximately one percent in order to benefit from a stock of reactivity sufficient enough to allow for experimental irradiations.

The sensitivity of detection systems was designed to indicate when fuel pins revealed even the slightest loss in leaktightness, releasing gaseous fission products over a period of several months. However, releases in such minute quantities renders detection of the faulty fuel sub-assembly impossible. Reactor operation is not disturbed by such minor and statistically permanent leaks. Due to the great number of fuel sub-assemblies having been implemented in the core, it seems reasonable to assume that the reactor's good results can be relied on when asserting that the fuel is capable of a higher performance.

Changes were made to several **precursor fuel sub-assemblies** to improve their reliability and life-span, as well as simplify fuel reprocessing activities. Therefore, graphite in the upper neutron shielding was removed, while retaining the same level of protection using steel and boron carbide. This alteration eliminated the risk of deformation to the upper neutron shield caused by the swelling of graphite under irradiation when in contact with sodium, which can occur in the event of a loss of leaktightness in the welded structure containing absorbent materials and partially composing each fuel sub-assembly head.

The 316 stainless steel used in the hexagonal wrapper was substituted for a titanium-stabilised stainless steel of the same grade with a much smaller swelling under irradiation.



Fuel sub-assembly

[2] Burn-up expressed in thermal energy (megawatt day) extracted from a tonne of mixed oxide. This parameter characterises both the use of fuel during irradiation and its depletion. In an EDF pressurised water reactor, the burn-up has been set at 45,000 MWd/t. The specific burn-up of 75,000 MWd/t corresponds to the burn-up of heavy atoms, such as uranium, plutonium and other actinides of about 9%.

The **first cladding failure** in a fuel pin occurred on May 1, 1979. Following the greatest release of fission gas (Xe^{135}) ever seen in the Phénix plant, the cladding failure neutron detection measurements rose considerably. It was possible to locate the faulty fuel sub-assembly before safety thresholds were reached as the progression of the event was sufficiently slow. The faulty sub-assembly was in fact one original experimental sub-assembly fitted with cold-worked 316 stainless steel cladding and a 316-Ti stainless steel hexagonal wrapper, which reached a burn-up of 73,100 MWd/t. The faulty sub-assembly was moved into the in-vessel fuel storage area on the core periphery.

The reactor was re-started on May 4 and analysis confirmed the fact that no other fuel sub-assembly revealed cladding failure. This incident illustrated the efficiency and sensitivity of the cladding failure detection and location systems. Gaseous fission product releases into the atmosphere were minimal and reactor unavailability remained under 3 days.

The six **control rods** and their associated mechanisms were also functioning properly. However, it was remarked that, thanks to in-sodium handling methods, hexagonal guide tube elongation due to irradiation was more pronounced than expected. Subsequent sheaths were shorted by 5 mm, before fabricating them using cold-worked 316 stainless steel, which proved to be more efficient in terms of swelling in comparison to the solution heat treated 316 stainless steel used in the initial sheaths. Following examination of the end-of-life control rods, adjustments to the initial dimensions and design of their spike were made to limit the possible consequences of swelling, such as the impossibility to reach the lowest position during a rod drop.

Following more than a year of tests on inactive sub-assembly models, the first irradiated

fuel sub-assembly was **dismantled in a hot cell** in September 1974. This fuel monitoring sub-assembly was irradiated during 56.2 EFPD and whose residual power was equivalent to 1.7 kW. Examinations (metrological, neutro-graphic, eddy current cladding inspections, spectrometric, etc.) were performed under good conditions in the irradiated elements cell. Results corresponded well to computer code calculations.

Following this period of operation, a rate of around 85 fuel sub-assemblies were dismantled each year, corresponding to 12,500 con-

Analysis, verification and feedback

During the tests, it was noticed that after a period in operation followed by a shutdown, the neutron control chambers indicated a level of neutrons that decreased for one or two days, with an apparent half life that corresponded to that of the sodium 24 (15 hours). The hypothesis of a reaction (γ, n) of the γ radiation of the sodium with a light element found in proximity to the neutron chambers was put forward. The only element liable to have produced a reaction of this type was the deuterium in the water of the emergency cooling system (0.02% of heavy water in natural water). To check this hypothesis, the emergency cooling system was drained, but without result (the tubes at the bottom cannot be drained). There was then no alternative but to add heavy water to the system. Two litres of heavy water were obtained from the operator of the Célestin reactors and the precious liquid was poured into the system. Immediately, the signal of the start-up chambers increased, only to decrease again in the usual manner. QED!

This phenomenon did not in any way disturb the operations of the Phénix plant. But it could have turned out to be a problem in the case of Superphénix. Its resolution at Phénix thus had a strong influence on the design of the neutron control system for Superphénix. In the end, the water pipes in the emergency cooling system were placed in such a way as to free the sectors of the three neutron guides. Similarly, the control chambers were protected with side protections made of enriched boron.





Spent sub-assembly dismantling cell

ditioned fuel pins. These fuel pins were transferred to the CEA pilot reprocessing unit at Marcoule and the reprocessing plant in La Hague after each handling activity. Several fuel pins were transferred to CEA examination laboratories at the Cadarache, Saclay, Fontenay-aux-Roses and Grenoble Research Centres. Sub-assembly structural waste (spikes, hexagonal wrappers, heads) were transferred to Marcoule site storage pits.

Washing facilities and sub-assembly dismantling equipment subjected to radiation equivalent to several tens of grays per hour successfully fulfilled their tasks. However, various adjustments were made using the remote control manipulator and other handling equipment, which involved lengthy and complicated manoeuvres. A short circuit in October 1974 for example meant the irradiated elements cell was unavailable for three months.

The most serious failures occurred while **dismantling the hexagonal wrapper**. Dismantling this wrapper involved sawing and tearing off in a coiling fashion a strip of metal from two sides opposite the hexagonal wrapper. Due to the decrease in ductility of stainless steel under irradiation, the coiling of cut-out metal strips on one side led to the rupture of the strip on more than one fuel sub-assembly. Several sub-assemblies were sent to laboratories at Cadarache to be dismantled by milling, before sending the fuel pins back to the Phénix plant for conditioning.

The decision was made that it was best to start the milling process on an angle of the hexagonal wrapper. Therefore, a cell was built next to the irradiated elements cell in which this new milling process was to take place. Furthermore, one fuel pin located in one angle of the bundle in new sub-assemblies was replaced with a fuel pin containing steel pellets only to avoid damaging fuel pins during this complex operation.

2.3. Reactor

The actual **reactor block** - referring to the vessels, roof and concrete slab - also functioned correctly and an extensive range of monitoring instrumentation allowed to verify it. The reactor block was fitted with over 550 sensors, including 450 thermocouples, with all information being processed by computers. These sensors were designed to measure and record temperatures of the sodium coolant and reactor structures, flow rates, cover gas pressure, sodium levels in free contact with gas, material stresses, structural movements and vibrations, etc. The relevance of results obtained during plant start-up tests actually transformed this monitoring equipment - an originally temporary situation - into a complex measuring system available to plant operators and CEA R&D units developing fast neutron reactors.



Space between roof and slab of the reactor



As the reactor had been shut down for five months due to problems with the intermediate heat exchangers (cf. § 2.4), an inspection of the vessel pit bottom and the space between the main vessel and the primary vessel was organised in March 1977. Not only was the insulation in perfect condition, but the possibility of accessing spaces close to the reactor taking in account the temperature and activity conditions was also demonstrated^[3].

The **primary and secondary coolant pumps** were in good working condition, which was all the more satisfying being immersed in sodium and thus representing a real technological challenge. An incident did however occur during the first semester of 1976: vibrations appeared in a primary coolant pump that slightly overstepped authorised vibration levels. The problem was solved by decreasing the pump speed from 800 to 790 revolutions per minute, until the pump could be extracted and analysed in July 1976. During this inspection, it was discovered that the hydrostatic bearing ring shrink fitted to the shaft expanded during severe thermal transients (automatic shutdown) and slipped down the pump shaft. The shaft was repaired in factory and the pump, after decontamination, was kept as a spare part^[4].

The simple effect monocylinder pump

The power plant's engineers and technicians were attentive to improving the operating of the facility. When, in 1975, problems arose in the re-armament of the braking hydraulic dash - pot of a rod mechanism, they studied and developed a " *device that makes it possible to boost, at a maximum pressure of two bars, the inferior capacity of the dash-pot. This device is composed of a simple effect manual monocylinder pump*". It is in fact a bicycle pump, and it is still used even today on this mechanism, along with its procedure.

[3] Such an operation took place in 1999 (cf. § 6.1).

[4] This incident occurred again in September 1981 following which all primary and secondary coolant pumps were modified. This involved tightening the ring by adding a mechanical fixation.



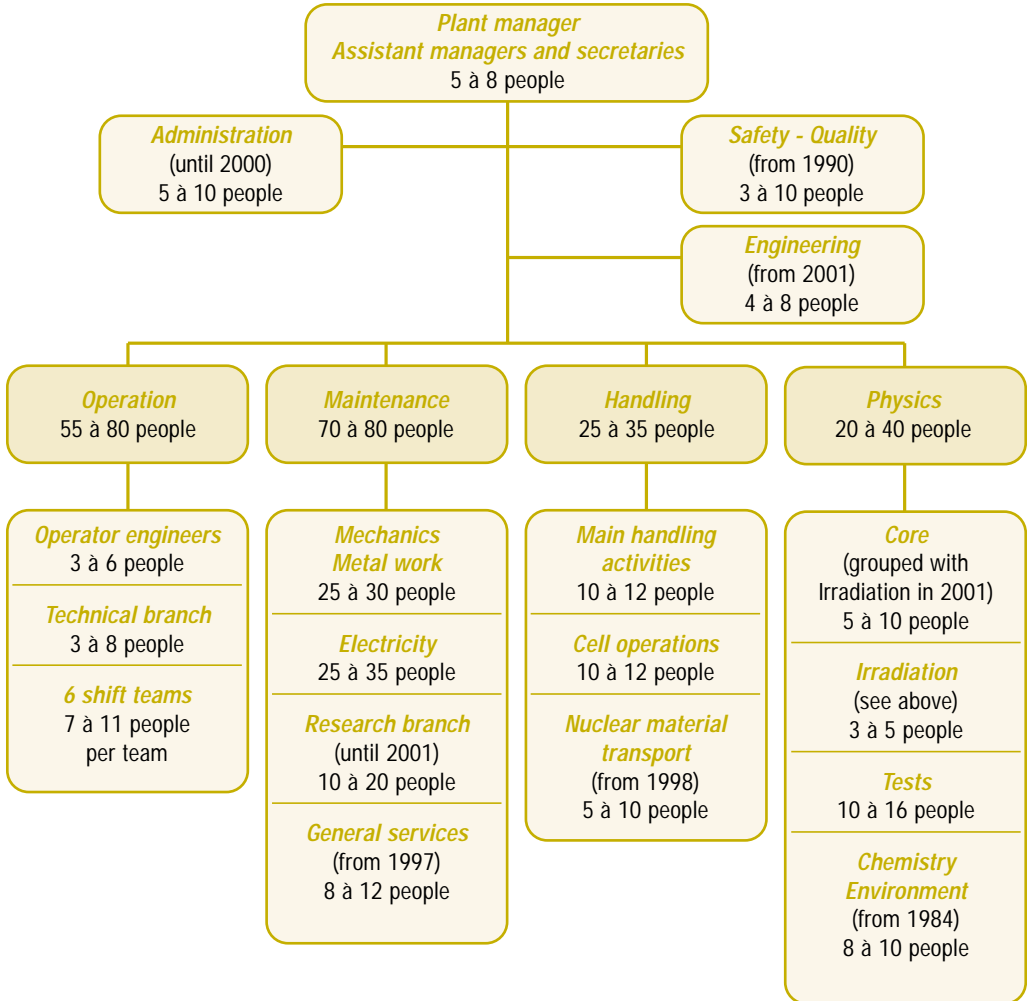
Primary pump

After several operations, it was remarked that the systematic replacement of mechanical seals recommended by the plant constructor was not necessary. Instead, grinding of the packing using a test bench and in-service follow-ups of vibrations, oil consumption and various other parameters were implemented. The actual life-span of such packing proved to be equivalent to several tens of thousands of hours.

The electrical equipment controlling the primary and secondary coolant pump speeds however required much attention. A variable speed drive was designed to continuously vary the speed of the induction motor of each pump from 150 to 1,000 revolutions per minute, as well as recover slip energy - usually lost through resistors in the motor's rotor. The principle behind this process



Phénix plant organisation chart and staff



Generally speaking, 80% of personnel works for the CEA while the remaining 20% works for EDF. Other than the plant manager (CEA) and the assistant managers (EDF), all other job positions were attributed regardless of the person's organisation. The number of staff mentioned in the organisation chart has varied over the years and in relation to internal reorganisational activities.

was interesting but the technology behind the equipment focused more on performance levels - sensitivity to control speed, recovery of energy, etc. - at the expense of reliability. Several trips provoking the automatic shutdown of the reactor are attributed to these pumps each year.

Minor **secondary coolant leakages** were detected three times, in September 1974, and then in March and July 1975. Leakage was traced to a joining weld on a large-diameter (450 mm) control butterfly valve located upstream from the steam generator N. 2. Approximately 20 litres of sodium was lost on the first two occasions and only about 1 litre on the last occasion. Leakage generally led to the slow spontaneous combustion of this sodium in the insulation, without triggering fires external to the insulation. As the secondary systems remained inactive and without pressure, there is no risk to safety, which makes it possible to empty the system and repair the leakage. However, repairs proved to be ineffective and the valves in all three systems were eventually replaced by diaphragms. Due to the experience acquired during operation, regulation possibilities could then be suppressed.

2.4. Leakage and intermediate heat exchangers

At 3 o'clock in the morning on July 11, 1976, while the plant was being restarted following a refuelling, the sodium coolant started leaking into the upper part of the intermediate heat exchanger E located in the secondary system N. 2. The fire was brought under control using two field operators who quickly arrived on location. The field operators only used one of the two Marcalina power fire extinguishers^[5] they had brought along. The secondary circuit was then drained. This was qualified as the first real sodium fire in the Phénix plant and revealed the professionalism of the shift team on duty at the time, highly praised by the CEA hierarchy.



Heat exchanger with insulation removed after a sodium leak

Having emptied the intermediate heat exchanger of its sodium and cleaned all around (approximately ten kilograms of sodium had caught fire outside the insulation), the plant was able to operate at two thirds of its rated power without using the secondary system N. 2, while waiting for analysis of what caused the leakage and decisions upon the repairs to be done. Two "mock heat exchangers" or devices designed to plug the heat exchanger penetration were also built so that the reactor could operate correctly without the faulty heat exchanger.

On October 3 the same year, the detection system identified sodium in the annular space of the intermediate heat exchanger F located in the secondary system N. 3. Provisional inspection was carried out in an attempt to precisely

[5] Marcalina powder was developed by the CEA and is the chemical agent used to extinguish sodium fires. This powder is composed of sodium and lithium carbonates and graphite.



locate the leakage and the reactor was shut down on the morning of October 5, 1976. At about 2:00 pm, the sodium overflowed from the annular space - kept in a nitrogen atmosphere - and ignited in contact with air. The secondary circuit N. 3 was immediately emptied and the sodium fire was brought under control by the shift team and security staff. Seeing that now two secondary systems were unavailable, the plant was shut down and the studies launched after the first leak were continued.

Investigation that followed was designed to a) locate the leak inside each of the two intermediate heat exchangers, b) understand what provoked the leak and c) define all necessary repairs and adjustments. The intermediate heat exchangers F and E were removed from the reactor using a component handling cask, cleaned (all sodium was removed) and decontaminated, which represented the first operation of this type with irradiated and contaminated material^[6]. Having decontaminated all material, repairs were a simple matter of sheet metal and mechanical work carried out under normal working conditions as no protection whatsoever was needed.

Examination, confirmed by metallographic analysis, revealed a crack in the joining weld on the metal plate closing off the sodium outlet in the secondary system above the heat exchanger. This rigid plate connects two long shells at different temperatures. The differential thermal dilatation of these two plates caused the weld to crack. This problem was described as a generic defect related to a faulty design (insufficient bending radius generating a concentrated stress region), a non-compliance with project specifications (excess thickness of shell) and



Repairing a washed and decontaminated intermediate heat exchanger

a poor understanding of hydraulic flows (mixtures of sodium jets at different temperatures exiting the heat exchanger tubes).

This analysis was corroborated by measurements recorded by instrumentation fitted to heat exchangers - including the heat exchanger F reinstalled after crack repairs - that operated at two thirds of the rated power from June 20 to August 22, 1977. However, on August 31, 1977, the intermediate heat exchanger B located in the secondary system N. 1 also began to leak. This minor leakage - several cubic centimetres - was detected during plant start-up by the spark plug leak detector fitted in the heat exchanger annular space. It was therefore decided to complete repairs on the heat exchangers before operating the reactor at full power again.

The idea behind the **adjustments made** consisted in replacing the faulty plate with a more flexible closure designed to reduce in-service stress (except in the case of the heat exchang-

[6] The intermediate heat exchangers are contaminated by irradiated atoms on the surface of fuel pin cladding (formation of Mn54 in particular) that are swept from the metallic surface by the flow of sodium, before being mainly deposited on the coldest parts of the reactor (regions under the heat exchangers).



er F whose plate was reworked to remove the cracked region and increase the weld joining radii). A thermal insulation was also fitted to reduce thermal differences between supporting shells, even if the latter were not responsible for the leaks. Leak detection systems were reinforced and all heat exchangers were equipped with metal leak-tight bellows. In parallel, a new intermediate heat exchanger was ordered based on a design having integrated all the adjustments made to the former exchangers.

A flow-mixing device fitted in the sodium header at the tube plate outlet was tested on one of the heat exchangers. This device was designed to reduce temperature differences between the shells and later came into general use. It was realised at a later date that cracking had been aggravated during operations the reactor operating with only two of the three secondary circuits, during which sodium levels were such that the sodium was able to infiltrate the heat exchangers during shut-down by rising over the top of the exchanger sodium shutting rings. The deficiency in the

primary system flow rate going through the heat exchangers during operation provoked an increase in temperature differences between the shells, thereby accentuating the cracking. Particular attention was thereupon paid to hot and cold sodium levels during non-symmetrical operation.

The plant was restarted with four modified intermediate heat exchangers and two "dummy exchangers" in December 1977, operating at two thirds of its rated power. Repairs to the remaining heat exchangers were completed in February 1978. After one last permutation, the reactor was once again able to operate using all six intermediate heat exchangers at rated power in April 1978.

During such incidents, the principle behind an integrated reactor such as the Phénix plant that excludes all possibilities of active primary system sodium leaks, proved its relevance. Leaked sodium is secondary inactive sodium. Nothing was contaminated during these incidents whose only repercussions were to decrease the availability of the plant. Sodium fires were quickly and easily brought under control. The generic defect cost a whole year during which the reactor was shut down and eight months of reactor operation at only two thirds of its rated power.

On a more positive note, the possibility of removing, repairing and re-operating essential components having operated in a sodium environment and under irradiation was proven. The possibility of replacing a primary coolant pump was also proven, which was carried out in 1976 (cf. § 2.3). Furthermore, feedback from such incidents was very valuable for commercial sodium-cooled fast neutron reactors, highlighting several design defects in intermediate heat exchangers.

Leaks in the intermediate heat exchangers

In 1976, the Atomic Energy Commission took great pleasure in announcing on the international stage all the success encountered with the Phénix plant. This success was proof that France was the world leader in the fast neutron reactor field. It was for this reason that, when the first leak in the intermediate heat exchanger occurred in July, news of the event reached the very highest level of the organisation instantaneously and measures were taken to assess the situation and start repair work. All the same, during this period of doubt about whether or not the problem would be resolved, particularly in terms of the possibility of repairing the exchanger, there were people who thought that all the boasting had been out of place, and the eternal sceptics dumbed down French technology, with the most pessimistic even predicting the demise of the French fast neutron reactor type.



2.5. Sub-assembly and component handling

Fuel sub-assembly handling does engender a few operational difficulties (equipment failures) that must be dealt with on a continuous basis. Theoretically speaking, the time needed to carry out a handling campaign, free of all complications, during which one sixth of the core is renewed - equivalent to about twenty fuel sub-assemblies (fissile, breeder, control rods) - was estimated at six and a half days. In reality however, the average time required does in fact exceed this objective by two days. The main incidents occurring during handling cycles concern fuel sub-assemblies that have been deformed during their time in the reactor, as well as the fuel transfer arm (gripper motor failure, friction due to sodium aerosols on the scraper) and the rotating plug (whose drive system power was increased and roller friction was improved by cooling them down).

Conversely, after various adjustment problems, the Visus^[7] device proved to work extremely well. This device was designed to monitor echoes on a cathode ray tube (CRT) display that are produced by core components during fuel sub-assembly handling. This device proved to be a very useful instrument in “seeing through” sodium, therefore compensating for one of the disadvantages of this very opaque fluid and avoiding all risky handling operations.



Unusual object in the drum

The space above the sodium in the storage drum is monitored using a periscope. In June 1976, during the loading of new fuel sub-assemblies, an unusual object resembling a metallic rod was discovered on top of a handling flask in which a breeder sub-assembly had been placed half an hour beforehand. After having visualised and photographed the object several times, the object disappeared. This unusual object was in fact a rod of solid sodium that had formed by freezing in the upper neutron shielding channel of the new fuel sub-assembly while being inserted into the storage drum and had been held upright under the effect of the hydrostatic pressure. The temperature of the argon atmosphere in the storage drum was high enough to melt the sodium rod.

In January 1977, a leak was detected in the inflatable seal of the rotating plug that forms a second leaktightness barrier between the cover gas and the reactor hall and guarantees the conditioning of the argon atmosphere protecting the fusible metal seal. The tin-bismuth eutectic alloy used for the seal gradually deteriorates by superficial oxidation during handling cycles. This seal is in a liquid state during handling cycles so that the rotating plug can rotate. The replacement of the inflatable seal with a double seal and the cleaning of the fusible metal seal were programmed for the first ten-yearly inspection in 1980.

In March 1977, as the reactor had been shut down for five months for intermediate heat exchanger repairs, two operators in protective suits inspected the upper part of A-framed fuel transfer system, maintained in an argon atmosphere. The operators noted the overall satisfactory state of equipment and material. The

[7] Visualisation in the sodium coolant by ultrasounds.

[8] The normal unloading of a control rod involves removing it from its guide tube using a handling arm. The guide tube itself is unloaded separately in an identical fashion. The simultaneous unloading of a control rod and its guide tube was not considered in the reactor design.

The Visus

The Visus (visualisation by ultrasound) is a sonar device that makes it possible to detect in a very precise manner objects inside the sodium. The ultrasounds are sent on to the target and the echoes are processed electronically. The main function of the Visus is to guarantee that there is no mechanical connection between the assembly network and the rotating plug before the plug starts to rotate. The Visus is also used to monitor the movements of the sub-assemblies. The results are interpreted by means of images and telemetries. Optimal operation takes place at 250 °C.

The Visus is composed of two wave guides filled with NaK (a sodium and potassium alloy) for emitting and receiving ultrasounds. NaK was chosen so that there would be only one liquid phase along the wave guide, thus attenuating the ultrasounds as little as possible. In the upper part, translators transform the vibrations into electric signals thanks to the piezoelectric effect. These electronically-processed signals are analysed by an oscilloscope and by an interpretation assistance system called TIITUS (computerised processing of images and ultrasounds telemetry).

The Visus sweeps an area of 180° in 2 mm bands on a total altitude of 200 mm. These movements are performed by hydraulic engines guided by a robot. Apart from during handling periods, the Visus is retracted 2 m so that the sodium jet at the exit of the area between the core and the core cover plug does not make the wave guides vibrate.

operators only absorbed a minimal dose as the radiation and contamination was almost non-existent in the lock with the operation procedures applied.

In July 1978, two **control rods** could not be removed normally from the reactor as their swelling prevented them from being extracted from their hexagonal guide tubes^[9]. This blocking remarked during handling occurred above the position in which control rods are found during reactor operation. Consequently, this swelling did not call into question the safety of the reactor, with the

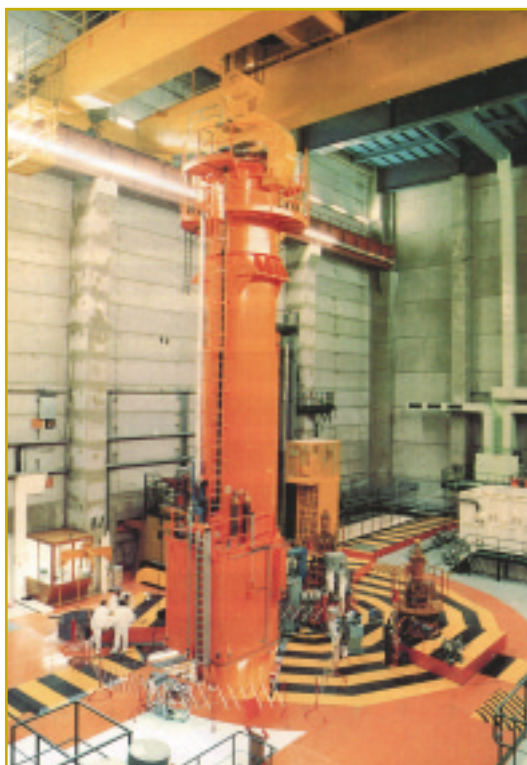
reactor being authorised by the Reactor Safety Commission to operate for one cycle on the condition that the good operation of the control rods is verified at mid-cycle.

This period of time was used to design, develop, build and test a special long tool as well as other adaptations made to existing tools designed to remove the control rod and guide tube unit using a component handling cask. This operation was successfully carried out in October 1978.

In May 1979, an empty bucket was found to be tilting in the storage drum under the pressure of sodium. While turning the rotary conveyor, the lower part of the bucket caught on the bottom of the transfer lock ramp and was distorted. Using a specially designed tool, it was possible to grip the bucket under the head rim and pull it through the opening of the manipulator door plug which had been previously taken off. This operation was carried out in less than 48 hours while preventing the sodium from coming into contact with air and protecting personnel from contamination and irradiation risks, thanks to tests performed on models prior to the actual operation.

After being cleaned into the irradiated elements cell, the experimental rigs were removed from their carrier sub-assembly that was then dried and then reinserted into the storage drum. During this final phase in November 1979, an ejection of sodium - most probably caused by remaining water - prevented the gripper from letting go of the sub-assembly head. The grapnel and sub-assembly were both brought back to the hot cell to be cleaned. A specially designed tool had to be made to unlock and remove the gripper. Following this incident, new carriers designed to avoid water retention were used to wash rigs.





Component handling flask

Specific handling activities were carried out to change the primary coolant pumps (in July 1973, October 1973, March 1974, and July 1976), the intermediate heat exchangers (from November 1976 to March 1978), control rod mechanisms and irradiation, measurement and monitoring devices. All such operations were carried out successfully. The main problems were caused by the accumulation of sodium oxides in various gaps and seal faces that require cleaning.

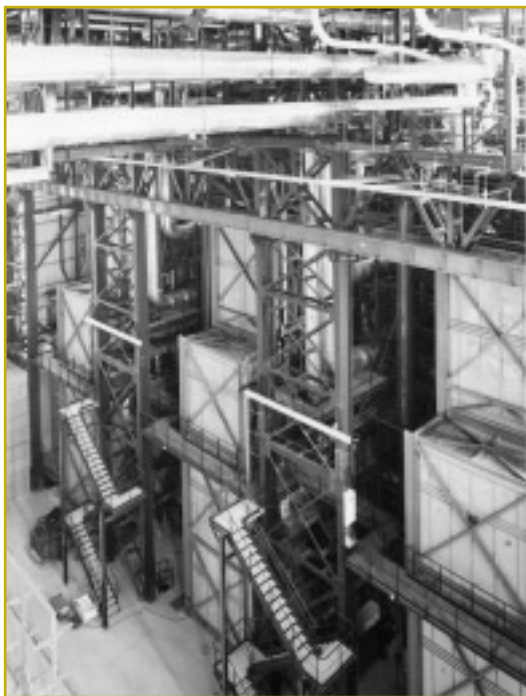
Cleaning and decontamination facilities were necessary to successfully complete all handling operations, but with numerous equipment overhauls and replacements (valves, pumps, seals, etc.) that proved to be poorly suited to the repetitive use of steam, acidic and basic washing, etc.

2.6. Steam generators and the electricity generating system

Steam generator failures at the plant are most feared of all equipment failures despite all the testing carried out during the design phase. The combination of non-radioactive sodium and water is dreaded in steam generators as the potential risk of provoking a violent chemical reaction is significant. Transitions to the steam phase are also apprehended, being a complex and demanding task for all materials. Nevertheless, the steam generators worked perfectly well and clocked up thousands of hours of operation under high temperature and pressure without the slightest incident. The only incidents worth mentioning occurred in equipment located within the steam generator environment.

In September 1975, two bursting discs located on one of the steam generator levels were replaced simultaneously with a new type of system. The deterioration of the pure argon atmosphere in the steam generator resulting from convection during such operations meant that the simultaneous opening of a sodium system was no longer possible in the future.

Four water leaks were found in the economiser - evaporator inlet of the steam generators between November 1975 and September 1976. The thickness of the sub-header underframe were measured using ultrasounds, which revealed the fact that they were wearing out rather quickly. Analysis demonstrated that this wear was provoked by perturbations generated by the high pressure water flow distribution orifice plates in the seven economiser - evaporator module tubes. Modifications made to improve the orifice plates geometry, including the addition of a protective cover and an anti-splash nozzle



Steam generator hall

(downstream), were qualified in a test loop. The 252 orifices plates were replaced between October 1976 and February 1977 while the reactor was shut down for intermediate heat exchanger repairs.

Plant operation feedback helped develop and implement a series of modifications designed to improve **performance**. A normal shutdown procedure was developed to maintain the turbo-generator set operating at very low power rates (20 MWe) in order to accelerate the turbo-generator cool-down and quickly end shaft line rolling and thereby reduce the period of time before operations can be carried out on components in the electricity generating system.

A new device designed to detect water or steam leaks in the steam generator sodium was also installed. This device detects hydrogen in the argon blanket of the expansion tank of each secondary cooling system and

complements the in the sodium detection device, particularly at low temperatures (250°C to 300°C) during facility start-up phases.

To reduce fire risks, the actuating medium of the isolating and decompression valve jacks in the steam generators was changed. The oil was replaced with air while maintaining the same pressure (160 bar), which made it possible to reuse the jacks and the pneumatic oil accumulators. This alteration was extended to include the actuating jacks of the sodium steam generator isolating valves.

The quality of water fed into the steam generators was improved through the systematic conditioning of the water systems when the shutdown period exceeds ten days. Thus, water is made to flow in these same systems 24 hours prior to feeding the steam generators and resin powders are used to treat con-

Staff from the CEA or EDF? No, Phénix

The staff of the Phénix plant come from a variety of places: the Marcoule site (G2, G3 and Célestin reactors), the Cadarache site (Rapsodie reactor, sodium loops and so on) and conventional EDF power plants, to name but three. There is a proportion of approximately 80% from the CEA and 20% from EDF. Recruitment started in 1970. Learning about the plant, as well as the numerous training courses (in sodium in Cadarache, in steam in the conventional EDF power plants and so on), soon made it possible to amalgamate the two organisations. The social climate in the Phénix plant is closest to that of the CEA's laboratories but events such as the St Eloi banquet (St Eloi is the patron saint of silversmiths and blacksmiths and, by extension, operators of thermal power plants) in early December has become a regular fixture. When a new employee arrives, he is presented to his future colleagues without mention of which employer pays their salaries. It is often purely by chance, or following a playful allusion to the advantages enjoyed by the employees of one or other organisation, that the new employee will discover that, of the colleagues with whom he shares his lunch in the canteen every day, Pierre and Jean are from the CEA and Michel from EDF.



densates during start-up phases. Hydrazine treatments were limited, without this having any influence on the water quality. But the diffusion rate of hydrogen through the steam generator tube walls and consequently the production of hydrides in the sodium coolant, not to mention the loading of cold traps were also notably limited.

Though the overall behaviour of the **electricity generating system** was generally satisfactory in relation to its complexity, various leaks appeared in small-diameter valves in the high pressure system. Two thirds of all incidents, more than often with no direct consequence on plant operations, occurred in the turbine room. On a slightly more humorous note, the operator found it "*a shame for nuclear power plants that conventional facilities always be dealt with as if it were the beginning of the century*".

A great number of unnecessary valves were eliminated. Information gathered from constructors revealed that these valves were a result of changes made to various different conventional EDF power plants that were integrated into Phénix's global plant design. It was also found that some steam generator equipment had been doubled up with this of

the electricity generating system, owing to the fact that the two facilities were designed separately.

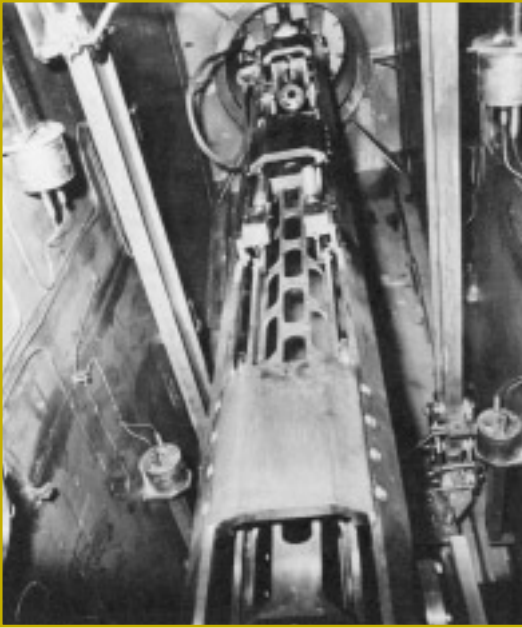
Nevertheless, the turbo-generator set responded very well to demands, such as in March 1974 when the Phénix plant operating at 150 MWe - carrying out power build-up tests - supplied the completely disrupted EDF 225 kV local network in electricity following a snow storm that seriously affected the Rhone region. The turbine endured disturbances from the electric grid yet the reactor remained stable thanks to the transient filtering function by the secondary cooling circuits.

2.7. First ten-yearly regulatory inspection and overhaul

The plant was shut down on March 28, 1980 for a programmed outage. This shutdown lasted almost three months. This shutdown had been prepared months in advance and numerous different tasks involving all facilities were to be carried out, from the reactor block to the turbo-generator set. Work on the turbine was first carried out, being the only real pressing matter during this shutdown^[9]. The general schedule was governed by safety requirements,



Operator sketches nuclear and conventional parts of the plant



Transfer lock tipper

particularly during the drainage of the secondary cooling circuits (a loop always remains operational in order to evacuate residual power in the reactor), by the prior analysis of the risk of extending deadlines, by the use of heavy handling apparatus, by logistic management (work and storage sites, etc.) and by the work load of each team.

In addition to the regular plant staff, more than 250 extra people - equivalent to almost 100,000 working hours - from about forty different companies took part in outage work, without the slightest problem in terms of irradiation and contamination. The accumulation of doses absorbed by personnel was equivalent to 0.019 man Sv. The worksite emitting the highest doses was the transfer lock, reaching a maximum of 0.004 man Sv^[10]. The plant was once again connected to the grid on June 29, 1980, a little more than a week ahead of schedule.

[9] The ten-yearly regulatory inspection and overhaul was carried out two years in advance.

[10] Doses in force at the time were expressed in rem (1 rem = 0.01 Sv = 10 mSv).

With regards to the reactor, the deteriorated inflatable seal of the rotating plug was replaced with a double seal and the drip pan of the fusible metal seal was cleaned from the powder produced by oxidation of the tin-bismuth eutectic alloy. Calculations had estimated a certain amount of wear in the screw-bolt system of the A-framed fuel transfer system rocking device, and even though no excessive wear was remarked after having dismantled the part, it was nevertheless replaced by a system with an improved lubrication device.

The A-framed transfer system preheating resistors were also replaced. The two valves located on the primary side of the A-framed transfer system were cleaned as approximately 60 kg of sodium had accumulated here during sub-assembly handling activities between the reactor and the storage drum. The Visus device was removed from the reactor using a component handling flask, before being cleaned, decontaminated and dismantled for inspection purposes following several problems encountered during operation. Only disposable parts (seals, etc.) needed replacing.

The intermediate heat exchanger F that had been repaired in a different manner to the five other exchangers was replaced with a new exchanger. This exchanger was then inspected and kept as a spare part. The plug designed to

The thermal balance

At the end of the 1970s, you had to be strong to do the power plant's thermal balance. It all started in the lab where the technicians calibrated the sensors themselves and where they gathered together their material so that they could spend a few hours on the facility. Once on site to measure the static pressure of the water and steam systems, they connected their "dead weight balances", the precursor of the pressure sensors that we use today. Certain weights on this balance could be as heavy as 40 kg...



locate gas leaks due to cladding failure, which had only been used for a few hours owing to the fact that the purification flasks, was clogged with sodium aerosols, was replaced with a new plug. This plug was developed and tested by the sodium technology section based at Cadarache, in which regeneration is possible. The plug of the central core channel was also removed and replaced with a rod mechanism used by the complementary shutdown system (cf. § 3.5).

In conformity with regulations, **steam generators** are normally expected to resist a pressure of 1.17 times the operating pressure. However, during examination of the technical file in March 1980, the French government Mining Service requested this be raised to 1.5 times the operating pressure, as is the case for new machinery. This request was based on the argument that most of the machine could not be inspected (tubes inside the modules), despite the fact that leaks can be detected by measuring the hydrogen content in the secondary cooling system.



Cross section of steam generator module

All three levels of each of the three steam generators were not only tested in conformity with government authority regulations but were also in line with associated in-depth non-destructive tests. These non-destructive tests highlighted some deformations in reheater module tubes. These deformations were occurring when the tubes became blocked by the supporting grids and preventing their expansion during the steam preheating phases. It was therefore decided to modify the steam generator preheating procedure before sodium filling in order to avoid excessive temperature differences between the different parts. Furthermore, new reheater modules were ordered with newly designed grids.

Two important operations were carried out on the **turbo-generator set**. The first operation involved a complete overhaul that is usually performed after having operated for several years. A loss of power remarked in June 1979 was partially explained when three metal rods were found in the inlet flow nozzle of the high pressure body. These rods were most probably left there during the initial assembly of the turbo-generator set yet caused no damage. Various other inspections and examinations revealed only normal wear and tear.

The second operation was carried out after having remarked the gradual misalignment of the shaft between the different turbine bodies. This phenomenon was explained by swelling due to the oxidation of expansive cement sealing the turbo-generator bearing and bodies^[11]. In order to remove this cement and replace it with another type of cement, all the turbine bearings were dismantled, the medium-pressure body was set aside and the high-pressure body was lifted off. This operation was also performed on the generator and the feed water pumps.

[11] This phenomenon also occurs in other 250MWe EDF turbines and requires the same solution.

2.8. Very encouraging results

Throughout the reactor's long periods in operation, during which the project characteristics were met and sometimes even significantly surpassed, the Phénix plant proved its role as a demonstration plant and justified the value of the technological choices that were implemented:

- the reactor operated at a power rate close to 600 MWt (for a project rated power of 563 MWt) and the turbo-generator set constantly provided the electric grid with 260 MWe,
- a gross power output of 45% was reached, thus representing the best power output for a conventional power plant of this generation, regardless of the reactor type (coal, fuel or nuclear),
- a specific burn-up of 75,000 MWd/t for most of the fuel sub-assemblies, sometimes even nearing 80,000 MWd/t for several precursory sub-assemblies,
- the core was entirely fuelled with pins containing a mixed uranium-plutonium oxide,
- the plutonium produced in the core was extracted from the fissile or breeder irradiated fuel sub-assemblies in fuel reprocessing plants.
- irradiation from experimental sub-assemblies (cladding material, etc.) was used to greatly improve fuel characteristics,
- operational results validated the technological choices made within the Superphénix 1200 MWe project framework whose construction began in 1977 at the Creys-Malville site.

On the contrary, analysis of incidents also highlighted the relevance of building a prototype. All technologies need to be experimentally validated, as sometimes failures prove to be more enriching than having a perfect operating record. The minor failures that occurred every now and then are used to constantly improve the performance of various different facility components, from the reactor, the turbo-generator set and its auxiliary equipment to the reactor core and the dismantling and evacuation of irradiated fuel sub-assemblies.

On a more fundamental level, the intermediate heat exchanger leaks revealed some design defects that were rectified in both Phénix and Superphénix. The defined and implemented corrective action clearly demonstrated the feasibility of carrying out repairs on equipment having operated for several thousand hours in a sodium environment under irradiation. On this occasion, the plant's capacity to operate using only two secondary cooling systems and at two thirds of its residual power was definitely taken advantage of in an attempt to limit the unavailability times of the Phénix plant.



Fernand CONTE

Director of the Phénix plant from January 1970 to 1984

The power plant's first ten years

Genesis of a genius. That is how one might describe the birth of the Phénix power plant and its first years in operation. "Genius" because Phénix has always been a highly intelligent plant and fully deserves that name. Producing electricity quietly and safely with an innovative system that promises centuries of trouble-free power, dependent on nobody and with a uniquely high efficiency. For having witnessed the preparation, construction and first ten years of this power plant I can confidently make that assertion. And say to all who worked to achieve it that their satisfaction is fully justified.



After a eulogy like that I must try to sum up those years of work with enough explanations to give a full picture. The Phénix plant is a prototype fast breeder reactor. Thanks to the research done by the CEA and the tests already run with the Rapsodie reactor, Phénix could be launched in full knowledge of the facts. And the predictions were fulfilled. This book tells, in detail, the story of those years. In so much detail that readers may be impressed by the number of little incidents and their consequences from day to day. But had they been left out, it would have been an endless tale of "everything went smoothly". And indeed, for all those who worked to that end, that is how it sums up.

All the results of the first ten years in operation are positive. The two incidents I describe below were also positive in terms of their consequences for the future of fast breeder reactors.

So: ten years with just two incidents involving sodium - the substance regarded as dangerous. There are so many other dangerous substances you could almost call that a joke. Here, then, are the two incidents – inevitable, in a prototype intended to bring problems to light.

A sodium leak in an intermediate heat exchanger above the reactor vessel. A slight leak, due to damaged metal, causing smoke as the sodium exposed to air slowly burned. The incident was easily stopped and the heat exchangers were repaired and strengthened. Any other source of fire would have done much more damage. After the repair work, the worksite was cleaned and renovated – except for the heat exchanger loft concerned. Rather dirty, but no trace of fire around the heat exchanger. Visitors were easily convinced.



Later, a leak was discovered between the secondary sodium and the water of the steam generators. Here too, slight damage to the metal casing had caused a slow, calm sodium-water reaction. Repair work revealed the damage: some tubes were filled with a pasty sodium salt that was easy to scoop out.

These two incidents provided the opportunity to revise certain components and further improve the equipment maintenance of the plant and of the future plants. Safety is achieved through experience with a prototype. That is what prototypes are for. To be brief and realistic, when it is properly managed, sodium is not very dangerous, especially in small quantities. Peoples' lives are at greater risk in an oil refinery than in a sodium-cooled power station.

As to nuclear incidents, there simply weren't any! And the other incidents in no way endangered safety. It is enough to note the doses received by the personnel involved in the tests and in operating the plant. The details are given elsewhere. Brittany's granite is more radioactive than the air at Phénix.

This outline leads to an indubitably positive conclusion, which is confirmed by the figures for production and efficiency. Results that match the enthusiasm and drive shown by the staff who worked there during those first ten years.



PERFORMANCES
TIME
(1980 - 1986)



Landmarks

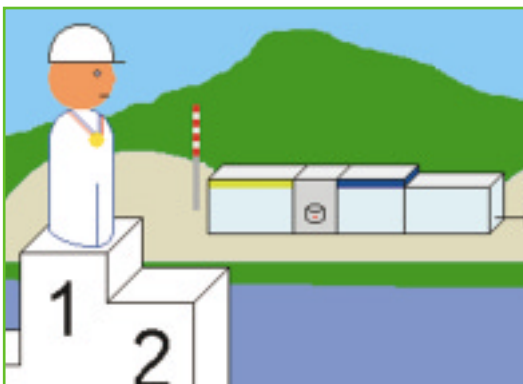
- 29 April 1982Sodium-water reaction in steam generator N. 2
- June 1982Eurodif enrichment plant at Pierrelatte starts up
- 10 August 1982Net cumulative electrical output: 10 billion kWh
- 16 December 1982Sodium-water reaction in steam generator N. 1
- 15 February 1983Sodium-water reaction in steam generator N. 3
- 20 March 1983Sodium-water reaction in steam generator N. 1
- 15 April 1983Definitive shutdown of Rapsodie
- 25 August – 13 November 1983Uninterrupted operation at rated capacity (81 days)
- 15 May 1984First divergence of Paluel 1 (1300 MWe series)
- 19 June 1984Definitive shutdown of G3 reactor at Marcoule
- 20 September 1984Phénix celebrates its 10th anniversary
- 7 September 1985First divergence of Superphénix (Creys-Malville)
- 26 April 1986Chernobyl accident (Ukraine, USSR)
- 17 September 1986Net cumulative electrical output: 15 billion kWh



After the success encountered during the early years, the challenge remained to operate the Phénix plant as stably and as continuously as possible. The operations team thus focused on reducing the losses of availability of the plant, on the one hand, on preventive maintenance and constant improvements to the equipment and functions, and on the other hand, on reducing reactivation time in the event of a shutdown. This was a constant task that is hard to grasp other than through the assessment of the victories obtained and presentation of the main problems encountered and often solved by dint of sheer hard work.

Several stimulating objectives motivated the workforce at the Phénix plant throughout the 1980's. The staff was constantly called upon to demonstrate the reactor's operating capacity, to improve reactor functioning and especially fuel performance while remaining within the safety rules, to develop or upgrade equipment for various purposes, to use the neutron flux to irradiate increasing numbers of materials within the scope of practical experiments for the CEA.

After several years of operation, the ownership and operation of the Phénix plant was to be transferred from the CEA to EDF, to serve as a means of electricity production within the nuclear reactors series. However, in the early 1980's, the CEA sought to retain its leadership



in the fuel cycle field by controlling the irradiations which could be conducted in the reactor, and EDF, despite its good operating results, was not keen on recovering this unusual reactor, just when its totally standardised fleet of nuclear power plants was experiencing fast growth. Accordingly, no steps were taken to carry out the transfer which had been planned for at the outset, and the structure of the power plant remained unchanged.

In 1982 and 1983, sodium-water reactions successively affected the reheater stages of three steam generators. Once these last "youthful incidents" were solved, the power plant set off at its true cruising speed and enjoyed a period free from any significant incidents during which it was operated under excellent conditions. Herein we provide some general evaluations of these operations, beyond the period under consideration, and statistics for electricity production, fuel, experimental irradiations, core instrumentation, control rods and other equipment. This chapter closes with the description of the Phénix plant in its physical surroundings.

3.1. The sodium - water reactions in the steam generators

After the first ten-yearly regulatory inspection and overhaul in spring of 1980, the power plant operated continuously at full power with no outstanding incidents for two full years. The load factor stood at 80 % during that time, with the cumulated total exceeding 60 %. Then, on 29 April 1982, a sodium - water reaction took place in module 12 of the reheater on steam generator N. 2.

When the corresponding alarms went off in the control room, the reactor was rapidly shut down and the steam generator N. 2 dried out. However, the failure in the opening of a check valve in the steam generator nitrogen filling



system^[1] enabled the sodium to slowly enter the water - steam part through the leak opening, and to reach the reheater isolation valve^[2]. As a result, the tubes of the twelve reheater modules and the related steam piping were polluted by the soda-sodium mixture. Instructions did not call for systematic drainage of the secondary circuit involved in a sodium - water reaction, however due to the pressure drop in the steam generator, the operators opened the drain valves to stop the sodium flow.



Steam generator module during handling

The sodium-water reaction

"The sodium-water reaction is an inevitable incident in our sodium reactor power plants. So far, Phénix has not suffered this type of event. Bearing in mind that a sodium-water reaction usually takes place two or three times in the normal life time of a power plant of this type, we prepare ourselves for it every day. We have become accustomed to the idea."

*Note from the director of the Phénix plant.
18 January 1981*

The sodium in the N. 2 secondary circuit, contained in the drainage tank, was purified by circulation through its cold trap. The defective module and the position of the leak were determined from outside of the casing, by looking for the strongest vibrations with a broom handle found near the zone^[3]. They were then confirmed by vibratory analyses. The radiographies showed that there were holes in two tubes, which was confirmed once

[1] Injecting nitrogen into a steam generator after a sodium – water reaction eliminates the water and stops the reaction. At the same time, the counter pressure of the gas in the water and steam pipes avoids sodium entering them, where it could once again react with the ambient moisture.

[2] The isolation valve was designed to function in water or in steam, but not in sodium. A slight sodium leak outwards appeared the next day and caused a sodium fire to start, which took the plant operators on duty by surprise (the small orange flames characteristic of a sodium fire on a steam valve !). The fire was rapidly extinguished.

[3] Broom handles and welding rods are often used as "stethoscopes" by the operators.

the module was taken apart. The total surface area of the holes came to approximately 2 cm². The combustion flame created by the water pouring into the sodium at the time of the initial leak created an erosion-corrosion effect (wastage), burning a hole in a second tube and damaging the module's shell.

Approximately thirty kilos of water leaked into the sodium. Analysis of the occurrence of the event proved the proper functioning and good sensitivity of the hydrogen detection system which monitors the leaktightness of the steam generators tubes. Nevertheless, the warning levels were made stricter so as to significantly reduce the system response time (from three minutes to one minute, approximately) and to provide earlier warning to the operators.

The reheater modules in steam generator N. 2 were removed after being polluted by the sodium. The plant started back up on 21 June and operated at two-thirds its power rating, using steam generators N. 1 and N. 3. During this period, the sodium-polluted steam piping was removed and cleaned, scraping off all the solidified sodium then eliminating any remaining traces by spraying the pipes clean with water and afterwards totally immersing them. These cleaning operations removed approximately 1200 kilos of sodium. Construction began on spare reheater modules following the conclusions reached during the first ten-



yearly outage inspections. The first modules were installed in steam generator N. 2 between September and November 1982. The reheater was subjected to hydraulic testing and the steam generator connected to its sodium circuit during refuelling shutdown in late November. The reactor started up on December 13th with the three steam generators on line.

Three days later, after a few problems with the generator, an early sodium - water reaction was detected in steam generator N. 1. The rapid shutdown and the steam generator dry-out occurred normally, and plant operations benefited greatly from the feedback experience after the previous leak. The very next day, the defective tube was located in the reheater module 12 and was removed, while the N. 1 sodium circuit was purified. The plant started back up on 29 December, and after a few adjustments reached rated power on 7 January 1983.

On 15 February 1983 the third sodium - water reaction occurred, again on reheater module 12, this time in steam generator N. 3. Operations were controlled as effectively as two months earlier, and the plant was reconnected to the grid on 28 February. However, the Interdepartmental Industrial Board ("Direction Interdépartementale de l'Industrie") restricted operating power to two-thirds rated capacity, after analysing the appraisals of the cracked tubes, which were indicating a fatigue mechanism.

Then, a fourth sodium - water reaction took place on 20 March 1983. This reaction involved module 11 on the reheater in steam generator N. 1. Like for the two previous reactions, the amount of water which reacted with the sodium was only a few kilograms. The decision was made to accelerate the replacement of the reheater modules on steam generator N. 3, originally scheduled for one month

later, in order to start the plant back up with two steam generators with new reheaters. The reactor operated thusly in June and July 1983, during which time steam generator N. 1 was also outfitted with new reheater modules. On 14 August 1983, Phénix once again operated at rated power. These four separate sodium - water incidents cost a total of six months of outage for the reactor, and nine months of restricted operations at two-thirds rated power.

At any rate, the defect which caused the sodium - water reaction was identified on the edge of a tube butt weld. The reaction's wastage effect twice damaged another tube (during the first incident, when operator reaction took a longer time, and to a lesser extent, during the fourth incident, when the tube was located just opposite the leak). The module shell was only slightly affected each time. The steam generators' safety bursting discs were never affected, which proves that the increased pressure had not reached the failure threshold (10 bar).

The expert investigations which were conducted on the modules involved in the four incidents and on the other modules showed that the tube defects were always on the butt weld beads. The defects were worse on the higher number modules (those which are closer to the steam intake) and on the first welds



Steam generator tube with leak



on the tubes (in the direction of the circulation of the steam). Additional records led to the conclusion that the origin of these leaks came from water mixed with steam passing through the reheater module tubes at certain transient conditions at plant start-up. When this water abruptly turns to steam in the tubes, this causes thermal shocks which affect their resistance.

The diagnosis resulted in a slightly modified design of the units which were replaced. The nitrogen injection circuit was modified to improve reliability (doubled injection lines, more accurate measurement of the low pressure values ...), as were some automatic and measurement sequences (creation of a safety channel). An automatic plant trip system was designed and installed actuated by a new hydrogen detection signal processing, using two dedicated computers. The decompression organs on the steam generator economisers were doubled. This was also when sodium aerosol detection was installed inside the steam generator casings. These replacement operations were made much easier by the fact that the steam generators at the Phénix plant were highly modular. A superheater module and a evaporator module were removed for expert evaluation, so as to compare them to the reheater modules, and they were found to be in perfect condition.

Such incidents were both dreaded and expected by the operator. Dreaded due to the violence of the phenomena involved - exothermic sodium-water reaction producing hydrogen and increasing pressure, yet expected because the probability of their occurring is not negligible. It can even be said that the system designers perhaps even secretly hoped such incidents would occur, if only to justify what they had designed to contain them. Experience has proven that the hydrogen detection is effective and immediate. This detection warns of leaks

as soon as they appear, rapidly decompresses the water-steam side of the installation and safely stops it. Since the secondary circuit carries clean, inactive sodium, no radioactivity or contamination is dispersed.

3.2. Electricity production

After these four sodium-water reactions, the plant continued to operate at full power until the second ten-yearly regulatory inspection and overhaul in spring 1989. January to September 1985 was the only time that production was reduced to two-thirds rated reactor power, due to a sodium leak on the F intermediate heat exchanger in November 1984 (cf. § 3.6). Plant staff increased to 250 persons, due to the increased size of the shift teams (10 agents per team), the shorter 39-hour work week and the creation of the Environment laboratory (cf. § 3.7).

The 10th anniversary of the power operation was properly marked by several days of festivities in September 1984, in combination with the 50th anniversary of the discovery of artificial radioactivity. The first day of the anniversary celebrations brought some five hundred national and regional scientific and political leaders together. Retired and current staff and their spouses were treated to a commemorative meal on the second day, and the "Open-House" weekend which followed hosted nearly one thousand visitors.

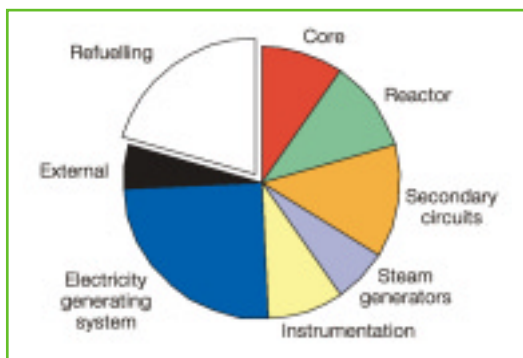


During this time, the plant improved on its own **performances** several times. 176,827,000 kWh were supplied to the electricity grid in January 1982. Between 25 August and 13 November 1983, the plant operated at full rated capacity for 81 days in a row^[4]. The plant topped the 10 billion kWh mark on 10 August 1982, then went on to exceed 15 billion kWh supplied to the grid on 17 September 1986.

Several interesting statistics can be provided by closely examining plant operation:

- The uninterrupted operating periods (without disconnection) are highly variable. Only in 10% of the cases did they exceed one month. On the other hand, half of the operating periods lasted less than 10 days.
- Half of the shutdown periods lasted less than one day^[5]. These were production losses (2 % of the total) caused by small incidents which were rapidly analysed and repaired. 10 % of stoppages lasted more than one week,
- 70 % of the irradiation cycles had an availability factor better than 70 % (not including scheduled shutdowns for refuelling), and 40% achieved operating availability better than 85 %,
- the total availability factor is slightly over 60 % for the period between the industrial start-up (1974) and the second ten-yearly outage (1989)

There have been approximately fifteen **plant shutdowns** each year. One-fifth of these shutdowns were due to refueling, and one-fourth to failures in the electricity generating system, that is the "conventional" part of the plant. One-fourth of the shutdowns were due to secondary sodium circuits and to the steam generators (in a non-nuclear zone), and one-fourth to the reactor itself, the balance due to causes outside of the



Origin of the shutdowns of the power plant

plant (lightning, electricity grid disconnection, ...). Not including the scheduled shutdowns, the corresponding power losses were due for the most part to intermediate heat exchangers (45 %) and to the steam generators (20 %), as well as, to a lesser extent, to the electricity generating system (10 %). The reactor core (sub-assemblies and control rods) was solely responsible for 5 % of the production losses.

Everything possible has been done to reduce the time involved in a shutdown. For example, in September 1980, a primary pump displayed an abnormal vibration level. It was shut down and the plant operated several days with only two primary pumps, while the spare pump was being prepared. On 9 October, the reactor was shut down and cooling was piloted so as to reach the stability conditions required to fit-up the handling flask as quickly as possible. Removal of the motors, replacement of the primary pump and final reassembly took place in only 5 days, and the reactor diverged on 17 October, then reached rated power the next day. There are very few, if any, operating

[4] The record for grid-connected operation was 99 days, set between 11 May and 18 August 1990, when reactor power was limited to 500 MWth during the analyses by the Safety Authority about decay heat removal capacity of the plant (cf. § 5.3)

[5] In the core of a fast breeder reactor, no xenon or samarium poisoning occurs, which provides for constraint-free start-up after a shutdown.



reactors on which an element as important as a primary pump can be replaced in such a short time.

However things were not always this easy. For example, in November 1983, the plant completed its 29th irradiation cycle under normal circumstances. After refuelling, divergence took place on the 26th and connection on the 28th at 9 a.m. One half-hour later, the operators noticed a drop in the steam pressure and set off the turbine without shutting down the reactor, for the power (130 MWth) was still sufficiently low. At 10 a.m., the plant was connected again, but 30 minutes later, the complementary shutdown system, which had just been installed for tests (cf. § 3.5) spuriously shut down the reactor. After going through all the operations for the entire approach to criticality, the reactor diverged at 4:30 p.m. and the turbo-generator set was connected to the grid at 4:00 in the morning.

During the power build-up, the vacuum in the condenser deteriorated, and the turbine tripped automatically, causing rapid shut-down of the reactor. By now it was 8:30 a.m. One hundred minutes later, the operators diverged the reactor, but had problems water

filling the steam generators. So they stopped the reactor in the early evening to work on a valve which controlled the feed water flow entering one of the steam generators. After repairs which took the day on 30 November, the fourth divergence in as many days took place at 4 p.m. The 1st of December was devoted to work on turbine regulation electronics. The plant was reconnected the following night, then disconnected to check the proper functioning of the regulation which had just been worked on, then reconnected, definitively, once again at 4 o'clock in the morning. By 10:30 a.m., the reactor reached its rated power level.

Unfortunately, less than 36 hours later, the complementary shutdown system spuriously dropped again, shutting the reactor down. The entire approach to criticality had to be done again to diverge the reactor at night, then reconnect the turbo-generator set early in the morning and power up. However, during this time a valve on a high pressure superheater had started to leak, and it took until 12 December to place it back in service and reach the supplementary 25 MWe which enabled the reactor to operate at rated power.



The turbo-generator set



Between 1974 and 1989, the plant experienced **power operations** (connected to the grid) approximately 70 % of the time, one-fourth of which was at reduced power, in particular during the periods it operated with two out of three primary pumps or two out of three secondary circuits. With an average of 12 unscheduled shutdowns a year, which is 2 shutdowns per thousand hours connected to the grid, its performance is far from the levels reached by the 900 and 1300 MWe pressurised water reactors in the EDF series. But the Phénix plant is a unique, prototype installation with a two-fold difference. On the one hand, it tests, on a life-size scale, new, often highly sensitive equipment, and on the other hand, it does not benefit from any return on experience or improvements from the operation of similar reactors.

Beware of dropping rods!

Many of the technicians at the power plant are confronted, through their professional activities, with the fear of dropping rods because of an error of judgement or poor handling on their part. Thus it was that, one day in 1981, an electronician was making adjustments to a regulation line on the turbine. He was in the middle of a telephone conversation with the machine operator when he heard the deafening explosion produced by opening the connector just above him. At the same time, he heard a voice shouting down the phone, "S***! The turbine! Crashed!", after which the connection was abruptly cut short.

He went to the control room wondering what error he could possibly have made that would have caused the automatic shutdown of the turbine and thus of the reactor. As he walked, feeling very small, he went over what he had done, from the preparations he had made to his final adjustments to the regulation line. Where and when had he made the error? In the control room, everyone was busy with the task in hand, as you can imagine in this kind of situation. Nobody took any notice of him. Sheepishly, he finally went over to the machine operator and asked him. The answer came quickly, "Don't be silly! I didn't say that you had made the turbine crash! I said that the turbine had crashed" ... You can imagine the relief felt by the technician!

The plant's thermal efficiency reached 45.3 % during the periods of stabilised operations. On the average, this is equal to 40 % due to the long periods of operations at two-thirds the rated power (following the anomalies encountered on the intermediate heat exchangers and the steam generators) during which the operating point cannot always be so well adjusted. This confirms the vital benefit of this type of power plant, which has a reduced environmental impact due to less heating of the cooling water from the river per unit of energy produced.

3.3. Fuel

The operating characteristics have practically not been modified. The maximum authorised linear power density - that is the power supplied per unit of length of each fuel pin) increased slightly (from 430 to 450 W/cm) due to the good fuel behaviour.

The limit to the nominal clad temperature for the fuel pins^[6] (650 °C) was replaced by a stochastic limit (no more than 5 % of fuel pins whose clad temperature is over 650 °C and no more than 0.1 % of the pins above 670 °C).

The end-of-life criteria for the sub-assemblies, related to the **improvements in the materials used** (cf. § 4.7), have however evolved in a spectacular fashion. This lifetime, which primarily depends on structure resistance under neutron flux, is generally expressed in burn-up of fissile elements. Initially fixed at 50,000 MWd/t, the specific burn-up of the Phénix sub-assemblies reached 90,000 MWd/t in the inner of the core and 115,000 MWd/t on the periphery, which corresponds to approximately 13.5% fission of the heavy atoms contained in the fuel, due to changes in the grades of steel in the cladding

[6] This temperature is constantly calculated based on mathematical models using measurements of the sodium temperature at the outlet of each assembly.



and hexagonal wrapper. This was equivalent to practically doubling the assembly residence time in the core, where they stayed for five to seven irradiation cycles.

A few of the experimental sub-assemblies reached specific burn-up values of around 120,000 MWd/t. The record was won by the VIGGEN 6 sub-assembly (15.15 Ti steel hexagonal wrapper and 15.15 Ti cold worked steel fuel pin cladding) which accrued 136,100 MWd/t between December 1984 and March 1990 in 15 irradiation cycles^[7]. Studies were done to reach specific burn-up of 150,000 MWd/t for all the sub-assemblies overall, after increasing the enrichment in plutonium (in the inner zone), using ferritic steel (EM 10) for the hexagonal wrapper, and modifying core management.

The limits are imposed by damage to the fuel sub-assembly structures. This is why the search is on for steels with the best behaviour under irradiation. It has been shown that, in the center of the Phénix core, specific burn-up of 90,000 MWd/t corresponds to damage to the cladding and to the hexagonal wrapper which is equal to 90 displacements per atom (dpa NRT), which means that with the neutron bombardment and the resulting atomic rearrangements, each iron atom moved an average of 90 times during its time in the reactor.

The gradual decrease in the available reactivity in the core, combined with the increase in the driver fuel burn-up rate and the consideration of the experimental program all required significant specific developments. The reactivity was adjusted by increasing the enrichment in the plutonium fuel in the inner zone (by 6 % in 1980, then by an additional



Transport cask for irradiated pins

4 % in 1984) and by changes in the size of the fissile core (use of locations which could be occupied by either fissile or breeder sub-assemblies). These changes provide for sufficient reactivity reserve at the beginning of the cycle to compensate for the fuel wear and stop the reactor in any circumstances^[8]. In correlation, the average length of the irradiation cycles has increased between two refuellings, gradually increasing from 56 EFPD (at the outset) to 90 EFPD (as of 1983).

Two to four refuelling campaigns take place each year. The time required to pass from an operating state to a handling state (FON - MANU) is approximately thirty hours. The average length of a sub-assembly movement (one period, which is loading, displacement or unloading) is approximately one hour. During each campaign, one to two hundred periods are performed. The return to the operating state (MANU - FON) takes approximately ten hours. Since the bowing of the steel sub-assemblies in the first ring (due to their irradiation) could disturb the unloading operations, these sub-assemblies underwent a 180 ° turn in 1986. Computer assistance for operator use of Visus was developed and implemented in 1988. Fuel dismantling takes place regularly, on a par with the handling campaigns. Dismantling

[7] The BOITIX 9 experimental sub-assembly (EM 10 hexagonal wrapper and 15.15 Ti cold worked cladding) reached 144,174 MWd/t in April 1995. The 7 CZAR 1 experimental pins, also clad in 15.15 Ti cold worked, reached 151,600 MWd/t in August 1990.

[8] The safety criteria require that at all times there be an anti-reactivity margin of 10 \$ (which is 3600 pcm) at 250 °C. This corresponds to increasing the control rods' efficiency by approximately 75 %.



occurs in the new annex cell, for the spent elements cell has been modified to work on the experimental sub-assemblies.

After preliminary operations on **fuel reprocessing** in 1975 and 1976 in facility 1 in La Hague on a few sub-assemblies, the first campaigns began in December 1976 in the Marcoule Pilot Plant, for direct reprocessing. This facility, operated by the CEA, is an spent fuel reprocessing installation which tests new instruments and processes on a representative scale and applying the same constraints as a reprocessing plant. Between 1976 and 1983, the facility reprocessed 9 tons of spent fuel from the Phénix plant whose fuel irradiation level reached 80,000 MWd/t^[9]. The Cogéma plant in La Hague (UP2 - 400) reprocessed approximately 10 tons between 1979 and 1984 (diluted with the fuel from the GCR reactors).



Marcoule Reprocessing Pilot Plant

This operation, a fundamental part of the fuel cycle, has taken full advantage of the CEA's experience in the field. Research and development also focuses on the very high level activity waste resulting from the reprocessing operations. The PIVER facility at Marcoule vitrifies the fission product solutions which result

[9] The Marcoule Pilot Plant then underwent a major renovation program, with the construction of the TOR facility which includes a specific FBR fuel reprocessing line and reprocessed 7 tons of fuel between 1988 and 1991 (maximum burn-up: 106,000 MWd/t).

from the reprocessing of 2 tons of Phénix fuel at the Marcoule Pilot Plant.

In all, the equivalent of four and a half cores from the Phénix plant have been reprocessed, which accounts for 25 tons of fuel. The plutonium which is recovered is reused to make new fuel, and sub-assemblies containing plutonium from the Phénix plant started being used in the core as of 1980. Some of these elements have even been re-reprocessed at the Marcoule Pilot Plant. Thus, the Phénix plant has performed the fuel cycle loop several times, clearly proving the value of the breeder reactor system.

The **breeding rate** planned for Phénix was 1.13. Measurements made at the time of the dissolution of the fuel evacuated from the plant provide an actual value which is closer to 1.16. This means that not only are as many fissile atoms created in the core as there are destroyed to produce energy, but 16% more fissile (plutonium 239) atoms are also created.

This entire experience, involving reprocessing high specific burn-up fuels, waste confinement and closed fuel cycle, is unique the world over, and proves that the fast breeder reactor fuel cycle is an industrial reality.

From the start, the reactor core was reloaded the equivalent of 7 times. This means loading of more than 700 fissile sub-assemblies, of which nearly 200 were experimental, or 140,000 fuel pins. To this must be added the several hundred breeder sub-assemblies where the plutonium forms. Starting in 1985, the new sub-assemblies were entirely assembled in the Cadarache plant. Up until then, insertion of the pins in the hexagonal wrapper, and welding of the spike were performed in the handling building.

Out of these thousands and thousands of pins, leaktightness failed on fifteen during opera-



tions, placing mixed oxide in contact with the sodium. This is referred to as **clad failure**. This low number clearly demonstrates the strength of the fuel, especially since half of the clad failures occurred on experimental pins irradiated beyond the normal limits, with the approval of the Safety Authority. The residence time of these failures in the reactor, starting from the beginning of the emission of the delayed neutrons which were able to be detected, varied from a few minutes to several months. For most of the clad failure cases, the reactor was stopped before the automatic trip threshold was reached.

The detection and location instrumentation beautifully fulfilled its function. It is both sensitive and reliable. Improvements to the location function enable it to complete a thorough prospection cycle in 6 minutes, instead of the original 47 minutes. Moreover, as of 1980, quantitative and qualitative measure-

ments were added, for the detection and location using chromatography and spectrometry of cover gas, in order to identify the isotopic ratios which are representative of the irradiation rate of the defective sub-assembly. This makes the search for failed elements faster and more reliable. Small leaktightness defects, seen only in the release of gaseous fission products, can thus be detected and monitored before they evolve into a clad failure which requires removing the sub-assembly from the core.

A clad failure always causes the plant to shut down for a maximum of 3 days, the length of time required to replace the defective sub-assembly. Such failures also cause gaseous discharge in the stack, which, due to the radioactive decay in the primary argon circuit tanks and the related purification, is limited to a release of approximately 0.1 TBq per pin of krypton 85, which should be compared to the prescribed annual limit of 400 TBq.

Another clad failure, a one-of-a-kind event, took place during a handling campaign in November 1983, when a core sub-assembly was being transferred to internal storage. The sub-assembly's temperature had strikingly risen during the previous cycle, near the limit of abnormal behaviour which could require early removal. During the handling in the reactor, to go from the core to internal storage, the sub-assembly is solely cooled by conduction with the sodium surrounding it. The temporary temperature increase caused pressure in the pin which evacuates the fission gases it contained until the sub-assembly is once again positioned and supplied with sodium through forced convection. The connection to the external storage drum was rapidly closed to avoid transferring the released activity which is trapped in the cover gas argon circuit without outside contamination.

The role of physicists

The engineers and technicians of the Core-Irradiations group in the Physics department draw up the orders for the standard objects of the core that need to be replaced (fuel and breeder sub-assemblies, control rods and so on). From the point of view of the safety of the reactor, they analyse the design of all the objects that will have to be introduced into the core (particularly the rigs and experimental sub-assemblies). They are also responsible for the quality control of the manufacturing process. They have even been asked to take part directly in the manufacture of certain experimental objects, in a special workshop in the power plant. When the object has been produced, they draw up the irradiation file that will be presented to the Safety Committee. In addition, they define the plans for loading the core, as well as the corresponding programs for handling the sub-assemblies, plus the operating parameters for the core for the Operations department. In operation, the readings for the core (temperatures, neutrons) are monitored on a daily basis to check that the technical requirements are being respected. Finally, for the back end of the cycle, the Core-Irradiations group characterises the irradiated sub-assemblies before they are evacuated by the Handling department.



3.4. Experimental irradiations

The first experimental irradiations were set up simultaneously with the reactor start-up, in 1973. The program was fairly small at first, and primarily involved samples of steel for certain sub-assemblies. Though it grew, reaching some twenty experiments between 1976 and 1980, it remained limited so as not to hinder the reactor's primary mission, which was to demonstrate the reliability of a fast breeder reactor. The experimental irradiation program took on greater significance starting in 1980, and was further enhanced by the definitive shutdown of Rapsodie in 1982.

There are approximately forty experimental sub-assemblies or rigs loaded in the core at all times, which change depending on the requirements requested by the CEA or its clients. Highly specific management of the number and location of the various experiments in the core is required to avoid any damage to the operating characteristics. The irradiated element cell is used to conduct the last assemblies. The operators frequently rebuild sub-assemblies or rigs based on experimental or fuel pins which have previously been irradiated in the reactor.

In all, **more than 200 experiments** have been conducted. Slightly over half of the cases involve experimental sub-assemblies where the type of materials or geometry of their structure differ from the driver sub-assemblies. The other cases involve irradiation rigs placed either in the central channel fuel sub-assemblies (which generally have only 180 fuel pins and do not have upper axial breeder blanket), or in central channel steel sub-assemblies. These experiments have achieved more than 1500 cycles of irradiation, which is equivalent to 3 centuries of reactor operation at full power rating. They have primarily aimed at

developing the fast breeder reactor system, focusing on the following points:

- acquisition of knowledge on fuel and structure materials,
- core element technology,
- qualification of the behaviour models and the design codes,
- improved understanding of the basic nuclear data and qualification of the design codes for fast breeder reactor cores.

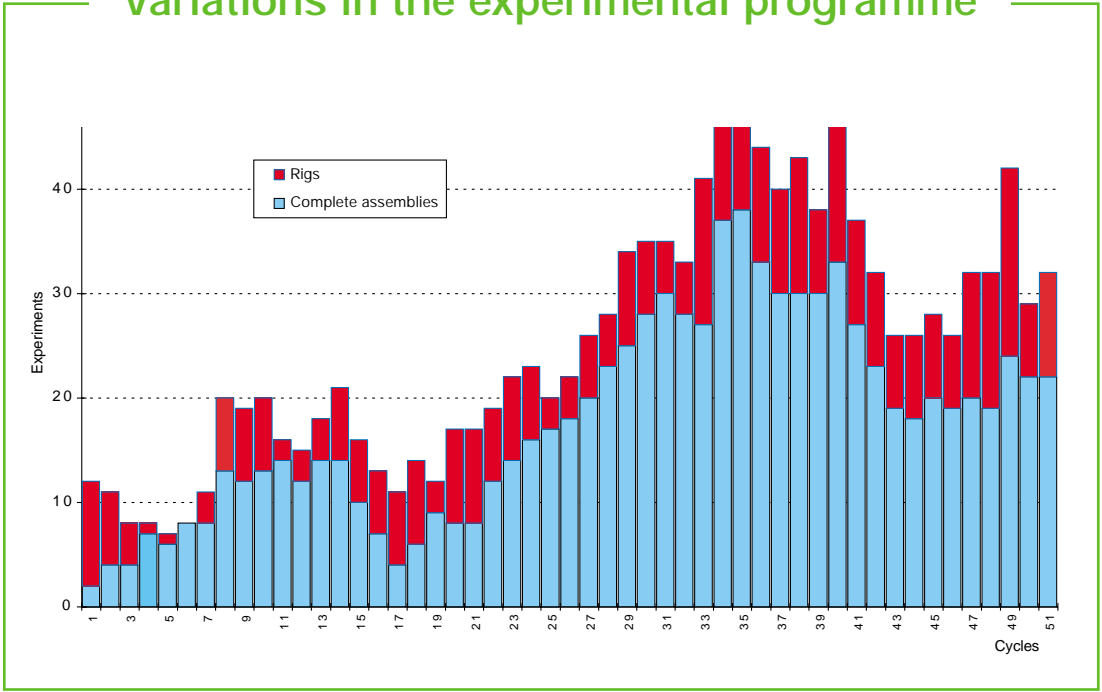
Different types of **fuels** have also been tested: annular (for Superphénix), carbide fuel, nitride, axial heterogeneous fuel, ... and fuel pellets for use in Superphénix or the SNR 300

A name for each experiment

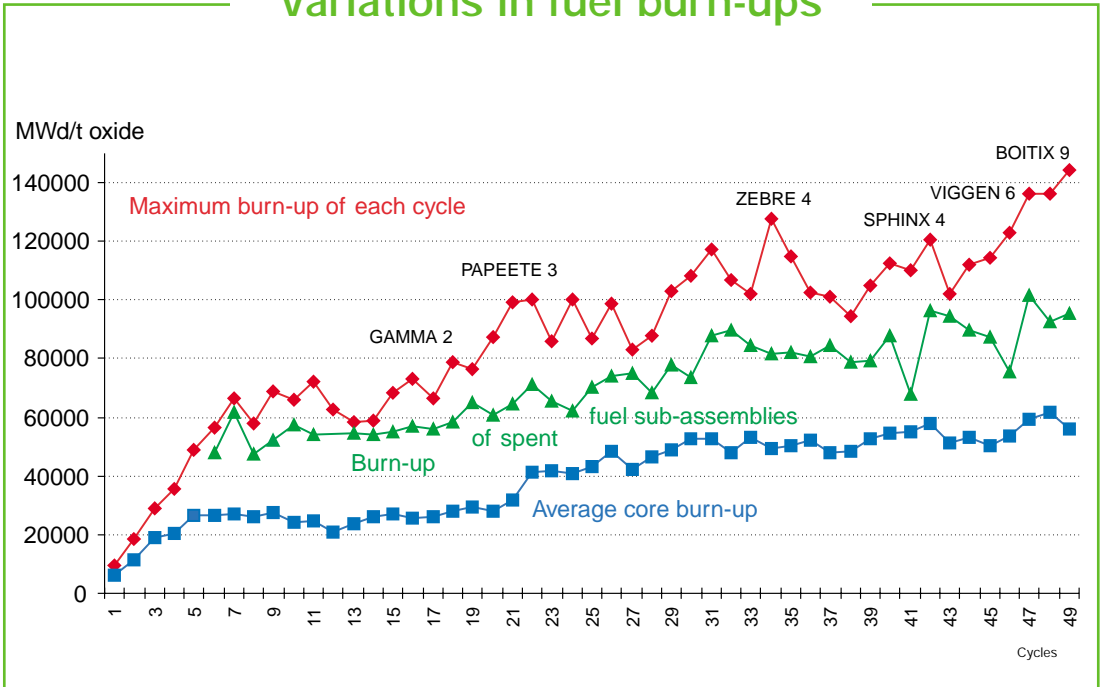
Each experiment has a name that is more or less associated with its subject, sometimes just by the juxtaposition of initials, but often the result of a fertile imagination combined with a certain amount of liberty-taking as far as spelling is concerned. The core of the Phénix plant thus has three Dinosaur, three Zèbre, five Sphinx, and so on. It is also sufficiently big to be able to contain, amongst others, Papeete, Floride, Durance, Bosnie and Memphis. The female sex (Irma, Ophélie, Flora, Vénus, Ariane, Félicie, Pretrouchka, Hildegard and Émilie) is represented, alongside a number of male dignitaries (Diogène, Charlemagne, Athos, Mathusalem, Roland – with his Oliphant -, Fracasse, ...). Space is covered with Supernova and Quasar, and Bacchus and fine wine are also present (Sauternes, Gigondas, Hermitage, Tavel, Santenay), as well as a Carape (carafe) and a Carafon (small carafe). Precious stones such as Onyx, Tourmaline, Jade, Saphir, Topaze, (as well as an Ecrin (jewellery box)) have not been forgotten, and nor have flowers Myosotis (forget-me-not), Passiflore (passionflower), Giroflée (wallflower), ... Under the joint influence of a certain comic book hero (Astérix), the ending of the name of the power plant, and the nickname of a French "grande école" ("X", for Polytechnique), Hadix, Boitix, Specix, Optimix, Poussix, Soudix, Idéfix, Scarabix have also spent a few months in the core. Finally, it is funny to note the presence of an Intrus (intruder), a Siffleur (whistler), a Postillon (spit), a Prophète, a Fétiche (fetish), a Monocle, and a Ferrofeu (iron in fire). An inventory that the poet Jacques Prévert would have been proud of!



Variations in the experimental programme



Variations in fuel burn-ups



German reactor. New structural materials for cladding and hexagonal wrappers, among others, have been tested and some have then been used at the Phénix plant, while others are planned for use in future cores for Superphénix and the FBR 1500 or the European Fast Reactor that engineers are in the process of designing (cf. § 4.2). A series of experiments also focused on the Japanese reactors, using the technologies implemented by their PNC designer. A vast program has also focused on absorbing elements in order to increase knowledge on materials behaviour, improve production processes, extend lifetime and benefit from the studies performed on Superphénix.

In order to conduct safety experiments on breeder reactor fuel, a test loop was designed by the engineers at the Phénix plant, working with engineers from the CEA Physics and Reactor Experiment department. This loop, named Bauphix, consists of a long leaktight tube inserted into the centre of the core from the core cover plug.

Inside the tube lies a sodium circuit in which new or spent fuel pins can be inserted and withdrawn depending on the irradiation needs. The unit is cooled by the primary sodium which circulates throughout the system. A handling flask for the device and a dismantling cell are also planned. A mock-up was built and tested at Cadarache at the end of the design studies^[10].

Thanks to the **neutron flux** ($7 \cdot 10^{15}$ n/cm².s at core centre) which is much higher than in a moderated reactor, and its high energy spectrum (150 keV on average), the Phénix plant is also used to gain knowledge on structural materials for pressurised water reactors and even for the future fusion reactors. Phénix can rapidly provide irradiation doses on test pieces which correspond to



Electrical equipment cabinets

the effect of several years of pressurised water reactor operation.

Preliminary experiments have been conducted to test the possibility of industrial production of cobalt 60 using the neutron leakage along the breeder zone. Here, the neutrons do not participate in the chain reaction (they are, in a sense, “free for the taking”) and the neutron flux spectrum is favourable to capture by the cobalt 59, if the neutrons are slowed down by the interposition of a calcium hydride matrix. A rig containing topaz stones is also placed in the core to study the production of blue topazes from clear topazes under irradiation. The experiment was successful, however it has not been pursued due to the regulatory problems involved in marketing precious gems produced by a nuclear reactor.

With respect to core physics, experimental irradiation of separated isotopes has been conducted. Studies underway in this field have resulted in the proposal to irradiate certain actinides or fission products to **transmute** them in the reactor. The original idea was to transform them by neutron bombarding them in the core, into either recyclable radio-elements (for example, create ²³⁸Pu, used for spatial generators, from

[10] Tests on the model were stopped in 1992 on orders from the Safety Authority to place a complementary shutdown system in the core centre (cf. § 5.4). The Bauphix loop thus never saw the core of the Phénix plant.



^{237}Np), or into stable elements (destruction of ^{99}Tc into ^{100}Ru). The Superfact experiment, conducted between October 1986 and January 1988 with the Karlsruhe Institute for Transuranian Elements, ran five cycles of irradiation. Four pins containing high content Americium and Neptunium actinides (heterogeneous concept) and four others containing mixed fuel with a low content of the same elements (homogeneous concept) were placed in a rig. The transmutation rate reached was 6.8 atoms per hundred, which is highly satisfactory. Metallurgical and radio-chemical analyses show good materials behaviour and demonstrate the feasibility of transmutation for these elements. This early research will be extremely valuable when the decision is made to undertake in-depth research on transmutation of minor actinides and long lived fission products (cf. § 5.5).

3.5. The core instrumentation and the control rods

The **core instrumentation**, in particular the temperature measurements providing the temperature of the sodium at the outlet from each fissile sub-assembly (T.R.T.C. system) has displayed outstanding reliability and accuracy. Core instrumentation has enabled monitoring temperature changes throughout the irradiation cycles, analysis of which shows evolutions in the flow which can be caused either by clad swelling or by warping of the spacer wires, well before such defects become a problem. The various core management design codes and the different irradiation parameters are highly inter-related. They are installed on the increasingly widespread office computers, which facilitates access by the Physics section technicians and engineers. Actual recordings of measurements made on the reactor have confirmed the values predicted by the codes.

On the subject of the **control rods**, sodium aerosols deposit on their mechanisms, hindering rod movement. Torquemeters and strain gauges have been installed to monitor this occurrence. The rods are raised and lowered in a 15-cm space every two weeks to monitor proper functioning. During scheduled shut-downs, the drop time of each control rod is precisely measured in order to monitor any change and detect the slightest anomaly as early as possible.

Design modifications were applied to the absorbers (clearance, geometry, materials ...) to limit any deformation and swelling during irradiation. Their residence time in the reactor has been increased from 220 equivalent full power days (EFPD) to 350 EFPD (rods) and 500 EFPD (for the hexagonal tube). There was a single occasion, on 4 December 1981, when a control rod did not fall gravitationally during an auto-





The core cover plug

matic shutdown at a very low power level, however its powered fall was automatically launched. After analysis of this incident, the automatic trip control logic was modified to make it less probably that such an incident should occur again (at the risk of increasing the frequency of unscheduled trips, which however has not occurred).

In order to validate the concept of **the complementary shutdown system (SAC)** planned for the Superphénix reactor, a prototype was built and tested in Phénix between March 1979 and June 1984. There are three principal phases to the experiment. First, a control rod secured in the up position in the assembly is tested to validate the hydraulic, thermal and neutron aspects (flux deformation in operation, heating of the boron carbide ...). Then, a dummy rod, with no boron carbide and a reduced stroke, is combined with a rod mech-

anism to test the overall functional aspect but with no direct neutron effect. Lastly, the full-scale test of a rod and its control mechanism takes place, to validate the overall concept.

The rod was installed in the core central channel and carried out three of the four initially planned operating cycles. After the two cycles between November 1981 and August 1982, the mechanism was unloaded and underwent new modifications. Repositioned in July 1983 with a new, instrumented electromagnet, it was removed anew 18 days later, for the sheathing on the cables supplying the electromagnet had been corroded by the acid used to decontaminate the translation tube, which had been poorly rinsed. A fourth electromagnet was installed, achieving a cycle between December 1983 and February 1984. The rod was withdrawn from the reactor in March 1984, and a last cycle took place between March and June 1984 using the dummy rod.

An intervention by divers

Providing an accurate account of the activity of the power plant, particularly through the monthly report, does not mean that there are no practical jokes. Thus it was that in February 1982 the photograph

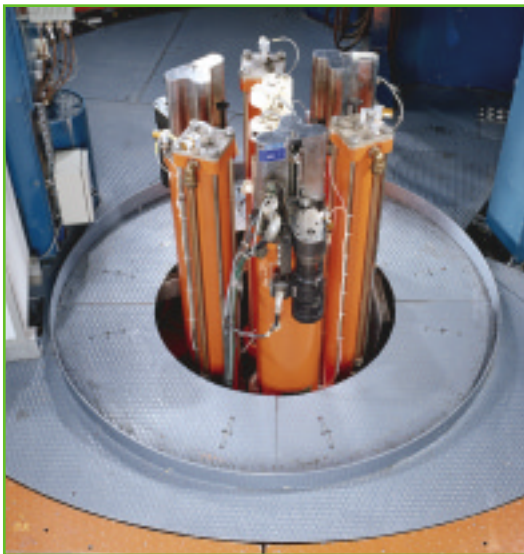


illustrating the underwater renovation of the pliers of a rod mechanism (a highly radioactive element because of its presence in the core) was a montage showing a diver holding the actual object of the operation in his hands. The following month, the first director to react to this surprising information was exceptionally awarded the "Grand Prix for readers of the monthly report".



These tests caused major interruptions in reactor operations. Among other disturbances, they caused five shutdowns during the power build-up phases. Nevertheless, the experience was rich in findings for the SuperPhénix complementary shutdown system, then for the new SAC installed in the Phénix plant in 1996, at which time there were no production losses. The following are examples of the knowledge gained due to this work:

- Elimination of the holes in the upper part, and increased sodium temperature exiting from the SAC (to avoid the thermal cracking of the electromagnet),
- Questioning the design of the electromagnets, in several steps
- Changes in the connection between the magnetic head and the rod,
- Change in the materials and the design of the rod and the mechanism (to limit friction between the rod and the guide tube, and between the electromagnet and the rod dash-pot cylinder).



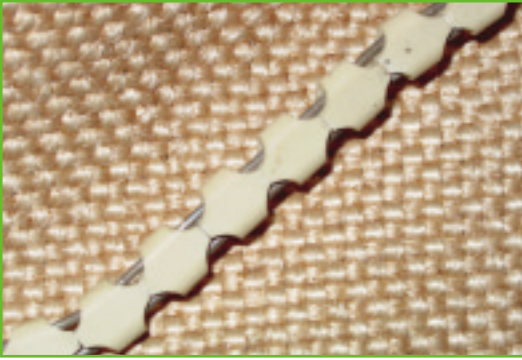
Control rod mechanisms

3.6. Behaviour of other materials

A secondary sodium leak occurred in the annular space on the **intermediate heat exchanger C** on 25 March 1984. The reactor maintained power operations, and the leak was monitored and did not evolve. At the next refuelling stop, the intermediate heat exchanger C was replaced by the F, on hold since 1980. During decontamination of the C exchanger in October 1984, the anti-vibration belts on the tube bundle were destroyed by the sulfamic acid bath which was used for the first time to replace the sulfo-phosphoric acid bath which was found to be insufficient to decontaminate the equipment. After expert appraisal, the exchanger was dismantled.

The F intermediate heat exchanger also suffered a sodium leak in November 1984. At first, the plant was authorised to operate at full power despite the presence of the leak (the sodium poured into the annular space kept under argon atmosphere. Sodium level was monitored by argon bubbling and it was transferred into the vessel when it reached a preset level). Indeed, under these conditions it was acceptable to wait for the delivery of the new intermediate exchanger expected in early 1986. However, the leak flow increased and the plant was shut down in late December. The G and F exchanger, which was paired on the N. 3 secondary circuit, and whose shutting ring drive rods were plugged by accumulations of sodium oxides (“mesos”), were removed from the reactor and replaced by two “dummy exchangers”.

The two exchangers were washed, decontaminated and thoroughly overhauled. The leak on intermediate exchanger F was located at the level of the weld bead made during the previous repairs. After repair (which included elimination of the previous repair welds and



Beaded wires for detecting sodium leaks

modification of the shutting ring drive system), the two exchangers were replaced in the reactor in October 1985. The new shutting ring drive system was applied to the three new intermediate heat exchangers (H, I and J), which were delivered in 1985 and 1986. A last sodium leak occurred on intermediate heat exchanger B, in September 1988. The system was replaced by exchanger H, which was specially instrumented to study the temperature gradients which appear during the operation transients and thus verify the fatigue codes.

Highly delicate complex **electrical equipment** ensures the variation and the regulation of the primary and secondary pump speeds. Despite strict maintenance and improved procedures and inspection tests, the equipment is not exempt from spurious, fleeting defects. Troubleshooting is difficult if not impossible, which results in repeated trips. The new materials that have been ordered operate on the same principles but use more recent technology, dual tracks and fault finding and memorisation. One such piece of equipment was replaced on the primary pump N. 1 in July 1984. Its successful implementation led to installing the same equipment for the five other pumps in June 1986.

Overall, the pipe electrical trace heating and the leak detection beader wires are reliable. The difficulty lies with the need to de-insulate

several meters of piping to find the defect when an alarm goes off. Attention must also be paid to distinguishing between a spurious alarm (wire defect) and a true alarm (sodium leak). Particular care must be taken in reassembling them on the piping, to avoid their tension or the presence of metallic particles which then could damage them or cause short circuits.

After identifying slight, gradual damage to the heat exchange on the evaporators on the **steam generators**, research got underway with the CEA and EDF specialised laboratories to determine the origin of the damage (mag-

The mechanics

A team of around ten people are responsible for the mechanical maintenance of the:

- turbine and its associated systems (cooling, lubrication),
- tapping apparatus (valves in the water and sodium circuits fixed with flanges, taps, valves...),
- classic pumps in the water circuits and those specific to the sodium circuits,
- compressors responsible for supplying the compressed air to actuate the valves,
- diesel engines that provide the back-up electricity supply for important safety equipment,
- ventilation circuits for the buildings in the power plant,
- filters in the condenser.

Preventive maintenance consists of regularly maintaining this apparatus by using procedures of systematic maintenance, sort of checklists that the operator follows. The associated operations are lubrication, changing injectors, cleaning filters, drainage, tightening packing and so on. Curative maintenance is dictated by material failures: changing pump packing, revising the blocks of the feed pumps damaged by rust particles, revising the valve seats, changing measurement apparatus, changing fan belts etc.



netite deposits) and to develop a cleaning procedure. Tests were conducted between 1981 and 1983 with two types of acids and various different processes. The procedure was tested on three tubes taken from steam generator N. 1, whose evaporator was successfully cleaned in July 1984. A cleaned module was sampled after one thousand hours of power operations, which validated the suitability and the harmlessness of the intervention. The operation was then repeated for the two other steam generators in October 1986 (N. 3, same steel grade), and in April 1990 (N. 2, for which a module was cleaned in advance to test the procedure on the steel in this steam generator, which was different from the two others).

Close attention is paid to the protection bursting discs on the steam generators which protect from a sodium - water reaction. Every 10,000 hours of power operations, one of these discs, chosen from among the hottest, is replaced and subjected to a bursting test. This tests the stability over time of the pressure at which these discs fail and thus ensures that the steam generators remain well protected in the event of a violent sodium - water reaction.

There have been no problems with the primary cold trap, which has accumulated the chemical and radioactive pollution carried by the sodium over time. The spare trap has not had to be used^[11]. However, secondary cold traps receive the permanent flow of sodium hydrides due to the diffusion of the hydrogen in the steam generators and the sodium oxides during the occasional pollutions caused by air ingress during the interventions on the drained circuits.

These traps are changed regularly. A report established in May 1983 showed that 13 tons of secondary sodium were consumed, essen-



Secondary cold trap

tially due to these changes. As a result, with a view to future operations, 43 tons of sodium were supplied between February and August 1984. This sodium comes from the core flood tank from the Rapsodie reactor, definitively shut down since April 1983, after sampling measures verified that it complied with acceptance conditions.

All the other equipment which is not described here operates fairly trouble-free, which does not mean that it is taken for granted. The frequency of the unscheduled shutdowns clearly shows that keeping a nuclear power plant operating does not come easy. In addition to the materials described above, there are many other types of equipment which display trouble spots, often recurrent, sometimes transitory, which makes troubleshooting all the harder. The following

[11] In 2003, the primary cold trap filling level measurement confirmed that no replacement would be required before the final shutdown of the reactor.



equipment is among that which presents the most problems yet which do not necessarily entail a shutdown:

- The 220 V distribution network, supplied by inverters,
- Leaks on the water and steam valves,
- Alternator auxiliaries (coupling devices, excitation, cooling),
- The turbine thrust-bearing oil supply circuit,
- The fuel transfer arm gripper actuator,
- The diaphragm control valves on the liquid effluent tanks.

3.7 Local environment

Located slightly north of the Marcoule site, the Phénix plant is separated from Marcoule by a fence and a guarded entrance through which Phénix employees come and go, and which authorises admission to a few accredited vehicles. Traffic is practically non-existent, though, for apart from the daily to and fro between the plant and the canteen, the power plant stands alone, virtually self-sufficient. The northern entrance, with its own parking lot, is the usual path for all those who do not take advantage of the buses which pick up and drop off the staff daily. The administrative offices, located in the Control and offices Building, then in the "chalet", do all the liaison work with the CEA Centre de la Vallée du Rhône.

A certain sense of isolationism can be felt, in particular with respect to the rest of the Marcoule centre, which is primarily occupied by Cogéma teams. This sense of segregation is further reinforced by the specific nature of the power plant, the only electricity-generating operating reactor run by the CEA, with a staff belonging to both CEA and EDF, whose hierarchical and technical



Aerial view of the Marcoule site

relations turn primarily to the centres Cadarache and Saclay.

The perception from outside the site is also different. The surrounding population can easily identify the power plant which produces electricity, for it stands out on the edge of the site, with its red and white striped stack, whereas the rest of the facility seems less easy to identify, clearly discreet and purposefully unexceptional, as befits national defense-related activities. Many people even refer to the "Marcoule power plant" as if the entire site was assimilated to the production of electricity, familiar to and appreciated by any modern-day household.

However, the power plant conducts no industrial communications with the local communities. Communication is managed by the CEA Centre de la Vallée du Rhône management. And the agents working at the Phénix plant are involved in local activities to the same extent as any other citizen, no more no less than elsewhere. Some are town councillors active in their town politics, and a former engineer even became mayor, then member of parliament from Bagnols-sur-Cèze in the nineties.

A nuclear power plant also means releases to the environment. And Phénix stands out as being particularly clean in this respect. The liq-



uid effluents include the washing water resulting from washing the fuel sub-assemblies and the components extracted from the reactor for maintenance and repair. Contamination primarily comes from the metallic structure activation products (^{54}Mn) and the average annual volume is approximately 300 m^3 , for radioactivity which varies between 0 and 5 TBq per year depending on the number of parts that have been washed^[12]. This water is carried by special tank trucks to the treatment station which handles the liquid effluents from Marcoule where the liquid waste is decontaminated prior to being discharged into the Rhône. The contamination which is extracted from the effluents is concentrated in storage facilities where the radioactivity can gradually decay at no risk.

Phénix also takes in and discharges river water from the Rhône, using it to cool the turbine condenser (approximately $43\,000\text{ m}^3/\text{h}$), just like any other standard power plant. However, as the plant's efficiency is as high as 45%, much less heat is discharged into the Rhône river (the Rhône river water heats up by approximately 0.1°C during low water periods).

Solid waste consists of structural waste from the fuel elements, the irradiated control rods and their guide tubes, and, to a lesser extent, dismantled components. This waste is transported to excavations prepared on the site, where it is stored for several decades until its radioactivity has decayed to a level where it can be sent to its definitive repository. The amount of solid waste is variable depending on the fuel sub-assemblies' removal to the reprocessing plants, and accounts for an annual average β and γ activity of approximately 40,000 TBq.

Gas effluents are the only waste released directly into the environment. This waste is produced from the ventilation of the plant

[12] The liquid effluents in 1977 and 1978 reached 9 TBq and 6 TBq respectively due to the first washings of the intermediate heat exchangers.



buildings, which is collected, filtered, monitored then released through the stack. This air contains low quantities of the argon cover gas which corresponds to the gas volumes which vary depending on the reactor power and temperature.

Radioactive decay tanks and a purification equipment considerably decrease the radioactivity contained in the argon, particularly in the event of a clad failure in a fuel sub-assembly. The radioactivity released through the stack is generally too low to be quantifiable, and the background noise of measure instruments is counted which comes to approximately 5 TBq per year. Annual accumulated gas releases vary between 5 and 7 TBq, which must be compared with the discharge permit, which authorises 400 TBq, and corresponds to radioactivity which would have no impact on the health of a person standing permanently in the direction of the stack releases.

All these discharges are under the **strict control** of the Central Service for Protection from Ionising Radiation (S.C.P.R.I.) which is part of the Health Ministry. The directives issued by this ministry often seem out of proportion given the good results reported on above. In the early 1980's, the S.C.P.R.I. decreed that the environmental monitoring conducted at Phénix had to comply with the same rules as the other EDF nuclear plants. This meant that instead of the Cogéma laboratories continuing to monitor the



entire site, the Phénix plant had to take on the equipment specifically selected by the S.C.P.R.I. and accredit a new team to implement the new equipment.

The measurements demanded must be taken within immediate proximity of the facility, however they also include samples taken in the outlying environment (plant life, milk ...), several kilometres from the site, equivalent to but distinct from the measures already taken by Cogéma. After several months of insistence and unsuccessful attempts to convince Cogéma to take such measures, plant management had to bow to the administrative ruling, and the Environment - Chemistry laboratory was opened within the Physics section in July 1984.

In early May 1986, the measurements taken revealed the passage of radioelements in the atmosphere from the Chernobyl accident, which confirmed the S.C.P.R.I. press releases which announced, as of 30 April, *"a slight increase in atmospheric radioactivity at some of the [metering] stations in south-east France, insignificant for public health"*. Thus, the dose rate due to the ambient gamma activity reached a peak of $19 \cdot 10^{-2} \mu\text{Gy/h}$ on the 5th of May. This must be compared with the average value of $4 \cdot 10^{-2} \mu\text{Gy/h}$ during the previous months, before then gradually decreasing to reach the normal values. These levels correspond to a dose rate which is *"two times lower than the rate to which you are exposed in any airplane flight. In fact, this exposure, and its variations, are only capable of being detected due to the extreme sensitivity of the radioactivity measuring instruments. These rates have no impact whatsoever on public health"*^[13] ».

In the field of **radiation protection**, the fast breeder reactors display outstanding performances, due to several elements which have been designed into the reactor from the start:

- The biological shield provided by the massive volume of sodium in the primary circuit,
- The absence of circuits carrying radioactive or contaminated fluid into the areas where operations and maintenance staff work,
- Rapid detection and repair of any sodium leak,
- The internal reactor structures are confined at all times in the main vessel.

This is verified by the dosimetry measured on the persons working at the Phénix plant,

The radioprotection team

In the Phénix plant, as in the other CEA facilities, responsibility for the radioprotection of workers, be they CEA, EDF or sub-contractors, is in the hands of a team of independent radioprotectionists. In the present case, this team has, since 1976, been seconded to the RadioProtection department of the Cogéma management of the Marcoule site.

The team is composed of around fifteen members in normal working hours plus shift workers. This is to be able to have permanently on hand, and in relation to the operations programmed, the radioprotection staff necessary for guaranteeing that the rules are respected at all times. The activities of this team particularly concern the radioprotection of all those who work in the power plant (dosimetric films and pens, control portals at the exit of buildings, etc.), radiological monitoring of the premises and operation of the fixed and mobile radioprotection materials, monitoring the movement of radioactive materials (fuel, sources, waste, etc.) and controlling liquid and gas emissions.

All its activities are very close to the operator as they make necessary frequent contact with the operations team of the reactor, the operators of fuel and components handling facilities, those who do maintenance on the equipment, the chemists in the environment laboratory and so on. In short, only the turbine hall would be unfamiliar to them if they did not monitor the gammagraphies that are periodically performed there.

[13] Excerpted from an S.C.P.R.I. press release on 6 May 1986.



whether plant workers, CEA or EDF staff, or contractors who perform timely interventions, particularly during the shutdown periods for maintenance.

The accumulated dose received by people working in the plant restricted area (between 250 and 450 persons annually, depending on the work to do) is, on the average, 0.065 h.Sv per year, which is approximately ten times less than that received in a pressurised water reactor plant. This dose varies depending on the amount and the type of the work done in the plant, which explains how dose rates reached 0.16 h.Sv in 1977 due to the work on the intermediate heat exchangers, and 0.12 h.Sv in 1985, due to new repairs on the intermediate exchangers and assembly of a hydrogen meter on the primary argon circuit near the piping where the dose rate is high. The annual dose taken in by the person with the most exposure is equal to 12.5 mSv. This value is exceptionally high and very few agents exceed 5 mSv per year.

With respect to **work accidents**, the average stands at five accidents per year among plant staff, which for a population of 250 is a frequency rate of 10 accidents per million hours worked. The Phénix plant is on an average with the other CEA installations and EDF plants. Many of these accidents are everyday occurrences (falls due to tripping or on the stairs, backache due to lifting of acceptable loads ...) and have no connection to the facility itself. One special feature is that due to the presence of two employers - CEA and EDF - there are two occupational health and safety committees (C.H.S.C.T.) at the Phénix plant - CEA centre de la Vallée du Rhône committee, and the committee for the EDF Phénix plant personnel sub-unit.

Like many other civil nuclear facilities, the Phénix plant hosts a great number of **visits**.

During its very first years, several famous visitors came to celebrate the successful start up of the plant. Among the illustrious guests were the French Minister for Industry and Research, the French Minister of the Interior, the Shah of Iran, the Vice Prime Minister of China, and other top energy policy makers from many countries throughout the world. Then as the Creys-Malville plant took up the challenge with the most advanced technology in the field of fast breeder reactors, the Phénix plant attracted fewer VIPs.

During a normal year, outside of restrictive measures for security reasons, the plant hosts approximately 1500 visitors per year. Visits feature a presentation on plant operations and a panoramic view of the reactor building, the steam generator building and the turbine room. The radiation-free environment means that visitors can view the main reactor equipment from a glassed-in room directly installed inside the reactor building itself. Annual visitors include an average of five to six hundred academics, politicians, scientists, and representatives from French and foreign business. All seek information on the technological and scientific results obtained by the Phénix plant. Visitors come from all around the globe - Russian, Japanese, Chinese, Koreans, British, American, Germans, Italians, Indians and many more, further proof of the widespread interest in fast breeder reactors.

3.8. A telling summary

This period confirmed the sustainability of the good results obtained by the plant. This was the era of records, both for electricity production and fuel performances. Plant staff regularly shared the progress made during speeches at the big international conferences on the nuclear industry and nuclear research. The most significant advances were



those in specific burn-up, for though the project had called for average specific burn-up values in the neighbourhood of 50,000 MWd/t, the research done on the results obtained was looking at values three time higher. In addition, the smooth operations and unproblematic maintenance of the plant were further expressed in the excellent results on both the human and the environmental levels, in terms of dosimetry and environmental discharges.

Indeed, the steam generators suffered feared incidents, and the intermediate heat exchangers required new repairs, yet as these disturbances were met and overcome, new knowledge was gained, in particular on the behaviour of steel in the presence of sodium. Production losses were limited, due to a large extent to the plant's ability to operate with only two secondary circuits in service. This chapter has gone into great detail on incidents which did occur and occasioned setbacks, but make no mistake about it - this period, like the previous period, was above all a time of excellent overall performance by the Phénix plant.

The reactor gradually incorporated an increasing amount of experiments. Due to its high neutron flux, to the relative ease of loading and unloading isolated sub-assemblies, and to the targets' low impact on the fast neutrons which maintained the chain reaction, the plant became a highly demanded tool for experimental irradiations. Reprocessing of the reactor's spent fuel was also a success, and akin to the mythical bird, the Phénix plant re-used the fuel that it had produced itself. Thanks to the highly specific core geometry, it can even produce more fuel than it consumes, thus recycling the stocks of depleted uranium, by-products of enrichment for the pressurised water reactors.



The control room

The Phénix plant is representative of the main options in sodium-cooled fast breeder reactor series and is living proof of the technological choices which have been implemented. These choices have generally been confirmed during the studies performed for the Superphénix reactor at the Creys-Malville plant, whose start-up has been followed, if not actually assisted in certain files by the engineers and technicians from the Phénix plant.



Bernard GIRAUD

*Deputy director (EDF) of the Phénix plant
from July 1979 to July 1983*

The utility's viewpoint

The Phénix power plant is the first demonstration of industrial-scale electricity generation by a fast neutron reactor. It should also be remembered that in 1974, EDF had not yet brought its latest conventional 250 MWe generating units on line.



And what an adventure it was for all the teams working on design and construction, all the young operators from such different backgrounds: CEA, EDF, research centres, conventional power stations, GCR and heavy water reactors etc.

The operator's mission was clear: while keeping to strict safety rules, "operate long-term at maximum possible power". This has been amply fulfilled in all respects: electricity output, materials endurance, operating and maintenance methods, fuel performance, safety, radiation protection, environment (effluent) etc. And, incidentally, providing a neutron flux as high and permanent as possible to enable experimenters to test materials and fuels for the future.

And the plant, attentively monitored, proved to be easy to run.

The reactor is controlled by a single parameter^[1]: the level of the bottom line of the six control rods. This is simply readjusted by a few millimetres each day to keep the core critical and so maintain the desired power level.

The heat inertia in the primary cooling system is important^[1]; it significantly absorbs any transients coming from the secondary cooling systems, steam generators, turbo-generator set or the electricity grid. This isolation makes it easier to control power variations, especially during hot start-up when the Phénix may rebuild power at a speed similar to that of conventional power plants.

Reactor control is simple because there is no significant effect on reactivity from xenon or samarium, however long the shutdown or transient lasts^[1]. At most, if the shutdown lasts several days the neutron physicist must calculate the transitory increase in reactivity due to the decay of neptunium 239 (period 2.4 days, precursor of plutonium 239).



The coefficient of reactivity associated with temperatures does not significantly vary during the irradiation cycle of the fuel^[1]. It always remains strictly negative^[1]. This means that nuclear heat-up can be allowed at any temperature^[1].

Two very significant examples illustrate some of Phénix's capabilities regarding operation and maintenance:

- passage from rated-capacity operation to operation with two secondary coolant circuits^[1] requires isolation from the grid, emptying of the defective secondary circuit, re-build up in power, re-connection to the grid and stabilisation at 160 MWe. In July 1975, all these operations were carried out without having to shut down the reactor completely; its power was gradually reduced to a level compatible with keeping the two available steam generators and their associated steam-water circuits running at minimum. The plant was isolated from the grid for less than nine hours.
- in September 1981, complete replacement of a damaged primary pump required only a nine-day halt in production^[1].

As regards radiation protection and effluent, the Phénix plant's outstanding results, much better than those of other reactor types, are due to the choice of an integrated primary cooling circuit^[1] and the fact that the primary coolant is not consumed during operation^[1].

How lucky we were to take part in the Phénix adventure during the tests, the commissioning and then during power operation! It was a time of abundant results, both for electricity output and for the irradiation tests. The period also included the first ten-yearly regulatory inspection and overhaul, some incidents, and major work on the intermediate heat exchangers and steam generators.

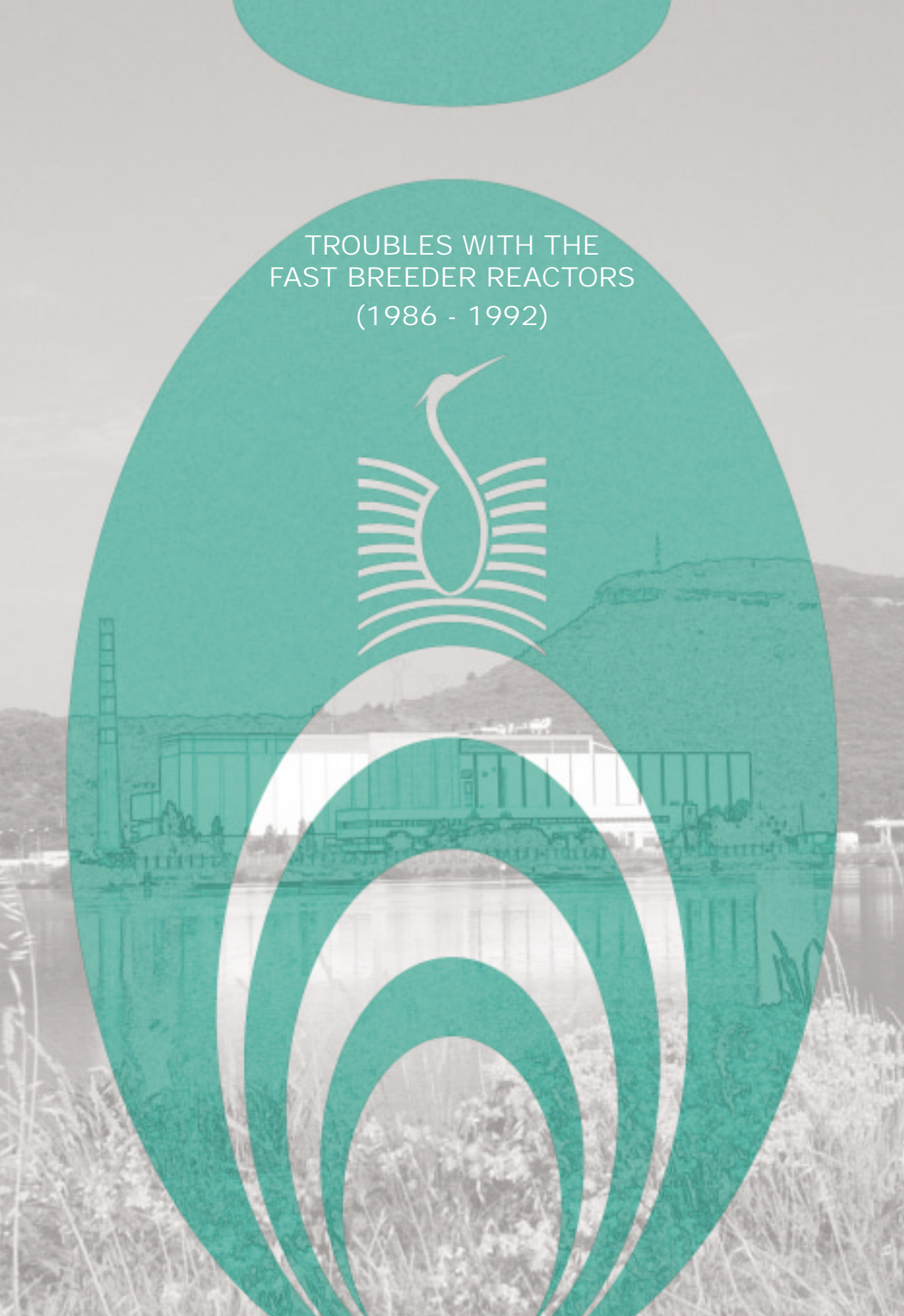
Operators and engineers alike, we all worked together enthusiastically and amassed a great deal of new knowledge.

We will not forget it!

^[1] All the points marked [1] are significant differences compared to even the most modern pressurized water reactors.



TROUBLES WITH THE
FAST BREEDER REACTORS
(1986 - 1992)



Landmarks

5 May 1986	..Sodium leak on a T-piece of secondary cooling circuit N. 3
18 August 1986	..Accident at Almeria solar power plant (Spain)
9 December 1986	..Superphénix reactor running at rated capacity
26 May 1987	..Creys-Malville shuts down following leak from storage drum
30 October 1988 - 8 January 1989	..Cycle running uninterrupted (72 EFPD in 71 days)
16 January - 28 March 1989	..Cycle running uninterrupted (72 EFPD in 72 days)
28 March - 2 August 1989	..Second ten-yearly regulatory inspection and overhaul
6 August 1989	..Shutdown due to negative reactivity transient
24 August 1989	..Shutdown due to negative reactivity transient
14 September 1989	..Shutdown due to negative reactivity transient
11 May - 18 August 1990	..Reactor running uninterrupted at 500 MWt (99 days)
3 July 1990	..Creys-Malville plant shut down owing to pollution of primary cooling circuit
22 July 1990	..Net cumulative electrical output: 20 billion kWh
9 September 1990	..Shutdown due to negative reactivity transient
14 January 1991	..Inauguration of Atalante laboratories at Marcoule
30 October 1991	..Definitive shutdown of Chooz A reactor (300 MWe PWR)
October 1991 - March 1992	..Tests at very low capacity in order to analyse shutdowns caused by negative reactivity transients
30 December 1991	..French Parliament passes law on research into management of radioactive waste ("Bataille" law)



The objective shared by the operating teams at the Phénix plant remained to operate the plant as best as possible to produce electricity (which earned substantial income for the plant's budget), to conduct experiments at the highest neutron flux (which minimised experiment time and obtained results as early as possible) and also to continue to learn about material behaviour and the equipment specific to sodium-cooled fast breeder reactors.

However, in the late 1980's and early 1990's, the course of events decided otherwise, for new incidents occurred, disturbing reactor operations. Elsewhere, storm clouds gathered over other power plants using liquid sodium as coolant, and cast their shadows over the future of the Phénix plant.

Now is the time to briefly describe the avatars of the big younger brother, the Creys-Malville plant, otherwise known as Superphénix, and to explain the evolution in other fast breeder reactors operated in several regions throughout the world. This will then bring us back to the events which occurred at the Phénix plant, particularly on the subject of sodium leaks, the second ten-yearly regulatory inspection and overhaul, and lastly, the four automatic trips which occurred in 1989 and 1990, caused by swift and negative variation in the reactivity measured in the core. These

four events, whose exact origin has remained elusive, were extensively analysed. The expert evaluation will be discussed at the end of the chapter, prior to reporting on the state of the knowledge gained on core materials.

4.1. Superphénix and anti-nuclear opposition

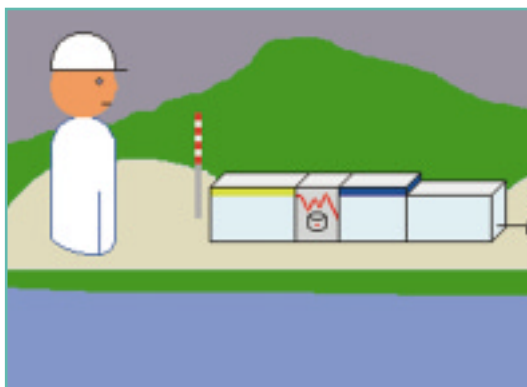
We will not go back over the history of the Creys-Malville plant which has been developed in many different communications, several parliamentary reports and at least one book^[1]. Nevertheless, for the purposes of our story, the main events must be retraced to see their direct and indirect impact on the Phénix facility.

As of 1970, the CEA had proposed a preliminary project for a 1000-MWe reactor named MIRA, in pursuance of the Phénix project. At the time, EDF had sought to raise the capacity to 1200 MWe, on a par with the pressurised water reactors which were taking over after the first 900-MWe series. The design of the Superphénix reactor was prepared by the same teams from the CEA and the GAAA that had successfully led the Phénix project, joined, in 1973, by the Nira Italian engineering teams, a subsidiary of Ansaldo and Fiat, who worked closely with EDF.

In July 1974, three European electricity producers (EDF, ENEL and RWE) formed the NERSA company to build and operate a nuclear power plant using the fast breeder reactor system. Licensed by the CEA, the Superphénix reactor was built by the Novatome firm, subsidiary of Framatome and CEA, created in 1976, and by Nira. The reactor

[1] Some of the best documented French studies include the following:

- "Bilan scientifique et technologique des recherches sur les réacteurs à neutrons rapides", a study by the Conseil Economique et Social, 1993
- "Superphénix pourquoi ?" by Georges VENDRYES, published by Nucléon, 1997
- "Superphénix et la filière des réacteurs à neutrons rapides", report N. 1018 by the Assemblée Nationale inquiry committee, 1998



was extrapolated from the Phénix reactor, whose start-up operations had confirmed the soundness of the technological choices, with a selection of slightly lower basic parameters (linear power rating of the sub-assemblies, temperatures, ...) in order to take advantage of the experience gained. Construction of the plant (3000 MWt, 2 x 600 MWe) began in late 1976 on the Creys-Malville site in the French department of Isère.

From the very start, construction of the reactor was heatedly opposed by the **ecologists' movements**, in particular from across the border (Swiss, German, ...). Several aspects were intensely criticised, fanned by often groundless rumours, some of which concerned the use of highly toxic plutonium which the public associated with the atomic bomb, the presence of immense quantities of sodium (5000 tons), which can react with the air and with water, the risk of a nuclear accident in the event of a runaway nuclear reaction, or core compaction ... Generally speaking, the opposition reflected overall current societal phenomena, as seen in the rejection of many areas of progress, radical refutation of industrial society and fear of strong military presence and a police state due to the required surveillance of the plutonium.

The basis issue disputed by the anti-nuclear protesters was the seemingly endless use of nuclear power, able to last for centuries through the tool provided by the combination of the breeder reactor and the spent fuel reprocessing facility in La Hague. The fast breeder became the scapegoat, the symbol of the quintessence of civil and military nuclear danger. On 30 July 1977, a major demonstration took place near the site, gathering several thousand opponents from many European countries, primarily Germany and Switzerland. The death of one demonstrator that day became the symbol of the struggle against "nucleocratic" power.



The Creys-Malville plant

Despite that demonstration and many other often spectacular, sometimes trouble-seeking, sometimes violent actions (rocket fire launched against the reactor building!), construction continued on the plant. The facility was completed in mid-1983, followed by acceptance tests, and the circuits were filled with sodium. The first criticality took place on 7 September 1985, followed by gradual power build-up until rated power was reached on 9 December 1986.

In May 1987, the reactor was shutdown after finding a sodium leak on the main storage drum vessel. The sodium leaks out in the nitrogen-filled safety vessel surrounding it. Eighteen months later this equipment was replaced by a gas-filled containment vessel, which was used to transfer the new and spent fuel sub-assemblies, yet which no longer served as storage, like the former vessel, thus extending the shutdown times for refuelling, and negatively impacting facility operations.

The Safety Authority granted authorisation to start up again on 12 January 1989. **Power build-up** still occurred in steps, and full power was reached on 16 June. One year later, on 3 July 1990, the reactor was shutdown once again, due to pollution in the primary sodium from the argon cover gas, in which air had entered. The sodium purification operations



took approximately seven months. At this point, the plant had already produced 4.5 billion kWh since first being connected in January 1986.

In 1991, just as the Creys-Malville facility was ready to go back into operation, two separate actions kept it shut down. On the one hand, the Safety Authority required several additional documents on the plant's ability to withstand very large sodium fires, and on the possibilities of the occurrence of events similar to the negative reactivity trip which had occurred at the Phénix plant (cf. § 4.5). On the other hand, several legal proceedings had been filed against the plant by groups opposed to the Creys-Malville plant, and one article in the January 1989 authorisation decree was rescinded for a flaw in the drafting.

At the end of the first half of 1992, after several experts had analysed every detail in all the safety files issued by the Creys-Malville plant operator, the Safety Authority decided to authorise the plant to start up once again. But the facility had to be modified to reinforce performances in the event of a major sodium fire based on extreme worst-case scenarios. These modifications were scheduled to be defined and implemented at the first refuelling shutdown the following year. At the same time the Academy of Science Applications Committee and the Parliamentary Office for the Evaluation of Scientific and Technological Choices issued separate reports whose findings clearly stated the importance of Superphénix, in particular for the present and future management of plutonium. Yet the Prime Minister decided that start-up would only occur once the sodium fire related work was done. He also requested that a public enquiry be held and that the Research Ministry draft a report on the incineration of radioactive waste.

The personnel at the Phénix plant closely followed the events at Creys-Malville. Start-up of the Superphénix reactor meant that some of the engineers and technicians would change locations, though remaining in the same technical environment. This however involved only a very few employees. Much more important were the tests being conducted at the Phénix plant on concepts developed for SuperPhénix. The flexibility provided by the configuration of the experimental reactor was put to use to test systems and irradiate material (cf. § 3.5). The incidents which arose at Creys-Malville during operation were investigated to ensure that similar cases did not disrupt operations at the Phénix plant. The two facilities, based on the same concept, had some simultaneous operations times and shared normal relations, though they were different in size.

The **nuclear power protesters' attitude** toward Superphénix was puzzling. The hostile demonstrations against the Creys-Malville plant reached a frenzy, yet the Phénix plant was left overall unchallenged. What was the explanation behind such divergent treatment? Three interdependent reasons can be advanced. The first relates to the construction period, which, for Creys-Malville, coincided with the wave of protes-

Place your bets for the divergence of Superphénix!

At the time of the divergence of the Phénix plant, the key actors in this success decided to bet on the date of the planned divergence for Superphénix. Given how successful the construction and commissioning tests had been for the Phénix plant, and given that the project studies for Superphénix were already well under way, most of those involved in the bet imagined the date somewhere around 1980 or 1981. There were even some, carried away in the enthusiasm of the operations that they had just lived, who bet on 1979. Only one pessimist imagined that the divergence of Superphénix would not take place till November 1982. But the future had some surprises in store!





tations in French society as the prosperous thirty year post-war boom period came to an end. The second reason relates to the size of the Superphénix reactor, which was a source of conflict even among nuclear energy supporters, and to the fact that it was presented as the prototype for an entire generation of fast breeder reactors, which would double the size of the EDF nuclear program, whereas the Phénix reactor always came across as an experimental reactor. The third reason related to the societal images which set EDF, “the big bad” industrial producer against the CEA, “the good” researcher. EDF, along with its fellow Italian and German “villains”, was perceived to operate the Creys-Malville electricity-producing nuclear power plant at all costs and regardless of the danger, whereas the CEA was perceived as a nice researcher making progress in experimental investigations in the small Phénix reactor which never made the news. Although it is a detail, another rea-

son for the difference in the attitudes toward the two is the proximity of cities with a reputation for protestation, such as Geneva, and even Grenoble, which facilitated the opposition to the power plant in Creys-Malville.

4.2. The other fast breeder reactors^[2]

The operators of fast breeder reactors throughout the world remain in fairly close contact, despite the competition among them at the outset. They hold international meetings, some solely on the subject of fast breeder reactors. They work together in bilateral meetings under various auspices, during which specific problems and issues are discussed in detail. Gradually, a veritable cooperation built up between the different sites, propelled by most of the nuclear power plant operators who were determined to share their operating experience and learn from others'. The slowdown, and halting of the fast neutron reactor programs has also served to strengthen the relations between plant operators, to the extent that some meetings took on the feel of a “veterans” reunion.

Operators actively collaborated in sending teams of experienced engineers from one reactor to another. This procedure deepened the understanding of the other systems and the way they are operated, even if the language and culture barriers were sometimes be hard to overcome. Such exchanges have taken place with the United Kingdom and Japan.

The **United States** were the first to embark on the fast neutron reactor path, and they continue to invest heavily in this system. However, in October 1966, the Fermi I reactor scrambled and had its core partially melt-

[2] See the map in the appendix, page 224. The point here is to provide a general view from the standpoint of the Phénix plant, and not to provide in-depth detail on these reactors' operations.



ed down due to a metallic plate which had come unfastened and obstructed the circulation of the sodium. The reactor was started back up in 1970, then definitively shut down in 1972, primarily due to economic reasons. However, the 400 MWt FFTF reactor commissioned in 1980 in Hanford, provided significant information on fuel development due to the many irradiation devices it had. It was definitively shut down in 1992. The CRBR 300 MWe reactor was abandoned in 1983 after work on the Clinch River site was halted in 1977, even though components continued to be built in the factory. Today, significant R&D work remains underway, though there are no plant construction projects on the short-term horizon.

In **Great Britain**, in Dounreay, the DFR demonstration reactor was definitively halted in 1977. The prototype reactor (PFR) whose design was similar to the Phénix reactor, reached rated power in February 1977. From the very start it experienced persistent problems with the steam generators which ended up requiring sleeving at all the tube-to-tube plate joints. The plant worked at a good load factor. It provided valuable information on operating flexibility and sodium technology. Environmental impact was negligible, personnel exposure was extremely low, and spent

fuel reprocessing was possible, with the reprocessing plant on the same site as the reactor. This experience fully confirmed many points shared by the Phénix plant. The Dounreay reactor was definitively shut down in March 1994.

Based on its successful experimental reactor program, the **U.S.S.R.** built the BN 350 on the banks of the Caspian sea. This 1000-MWt loop reactor feeds three 50 MWe turbine generators and a seawater desalination plant (5000 t/h). It diverged in November 1972 and power gradually built up. In October 1973, a major sodium - water reaction occurred in a steam generator. The Western world only learned about the incident after an American satellite observation of a cloud of sodium aerosols escaping from the building. On inspection at BN 350 three months later, representatives from Phénix did not see a single trace of the event, despite close inspection of the premises. The Russian operators had already thoroughly cleaned the plant. The steam generators were replaced and the plant started back up in 1976. After the collapse of the Soviet Union, the plant became Kazakh. Funds were lacking for plant maintenance and the decision was made to shut it down definitively in 1999.

The BN 600 nuclear power plant was then built in Beloyarsk, near Ekaterinburg. Equipped with an integrated 1470 MWt reactor and three 200 MWe turbine generators, it diverged in February 1980 and reached rated power in December 1981. It is still on operation and is an essential part of the electricity supply to the entire region, where it contributes 3.5 TWh yearly. It has won many awards for its availability and regular operations. The BN 600 has an annual load factor which often exceeds 80% and very few unscheduled shutdowns, due in particular to the redundant equipment in many sensitive



The British PFR and DFR reactors





The Russian BN 600 reactor

areas. As a result, the dozen or so sodium - water reactions in the steam generators have caused very few production losses.

Japan developed its interest in fast breeder reactors neutrons somewhat later, but quickly undertook a strong program. An experimental 50 MWt reactor called Joyo diverged in April 1977. Power was increased to 100 MWt in 1982, then to 140 MWt in 2003. Joyo has been used as an irradiation test bed. Based on this experience, the Japanese then built an electricity-producing reactor on the Sea of Japan, near Tsuruga. This 714 MWt and 280MWe reactor, a loop reactor, was named

A guaranteed change of scene

All the employees arrive, not in a bus, but in their own private cars just before the start of their working day. They go into the changing rooms and put on their uniforms before going to their work stations. A melody is on the air warning that work has started. The head of the department gives a short briefing and each team manager makes a short speech in front of all the staff, who are standing at their stations. At the end of the speech, a proverb on quality or safety is read out and then repeated in chorus. A rallying cry, "Yush!" is then given by all the staff, arms outstretched and index fingers pointing forwards, signifying the end of the briefing.

This is how every work day starts in Joyo in Japan. This practice was noted by the engineers who spent a few months seconded there, but was not used as a source of inspiration at the Phénix plant.

Monju, after a divinity symbolising wisdom and intelligence. It diverged in April 1994 and started power build-up operations marked by many tests and verifications, in the presence of engineers from the Phénix plant who followed one another on a yearly basis on the Monju site.

On 8 December 1995, a sodium leak occurred in an area where one of the secondary sodium circuits circulates. The leak occurred through a fatigue crack on a thermocouple thimble. The operators did not immediately drain the circuit (640 kilograms of sodium leaked out of the piping) nor stop the ventilation. The fire involved a relatively restricted volume (a few cubic meters which were then rapidly cleaned up), but the ventilation carried sodium aerosols to other building zones, setting off many alarms. After repairing the installations, the operator provided several documents explaining the analysis of the incident, and the modifications he intended to make to the facility and procedures. A significant research and development program was created to support the changes. However, the regulatory procedures, particularly lengthy in Japan and prolonged by legal recourse, have not yet issued any decisions on the subject.

In other countries, development of fast neutron reactors is not, or not yet, significant. In Germany, the experimental sodium-cooled KNK reactor (58 MWt; 20 MWe) first worked in the thermal spectrum, then in the rapid spectrum, from October 1972 to October 1991. One consequence of the strength of the German anti-nuclear movement were the delays in the construction of the SNR 300 prototype reactor. Start-up tests then took place, and the sodium was added, but the testing was interrupted prior to loading the fuel. Despite the administrative and legal decisions in favour of pursuing operations,



the operator gave up in 1991. Italy undertook the construction of an experimental reactor referred to as the PEC, but never finished it. India has built and continues to operate one experimental reactor, designed at the start in collaboration with France. However, the fact that India has refused to sign the Treaty of Non-Proliferation has halted all contacts. In the late 1990's, China launched a development program and started the construction of an experimental reactor.

In **France**, thermal asymmetry was found on the Rapsodie reactor vessel, and intermittent traces of radioactive sodium aerosols were found in the double envelope hermetically surrounding the entire primary circuit. The operating power was decreased to 22 MWt in June 1980 to minimise the thermal stresses which could be the source of the cracks and cause the cracking to continue near the vessel loop nozzles. The reactor was definitively shut down in April 1983, after a series of tests that studied its behaviour in scenarios of accidental situations.

Pursuing the studies on the Superphénix reactor, and during the time that the reactor was being built, EDF ordered a series of design studies from Novatome for a reactor which would become the leader in a series of identical fast neutron reactors. Thus, in the early 1980's, the RNR 1500 (also called Superphénix 2), was designed, the result of simplifications of the Superphénix concept and of the feedback from its construction and its start-up tests. There was talk of setting up four of this kind of reactor, associated with a fuel production and reprocessing centre in the town of Saint Etienne des Sorts, a few kilometres north of Marcoule. Then, however, the projects were cut back. Nevertheless, between 1988 and 1998, a consortium of European electricity producers funded studies on an advanced reactor known as the

European Fast Reactor (EFR). Building upon the accumulated knowledge shared by the different European developers, designers, industrialists and operators, this new project incorporated the very latest sodium-cooled fast neutron reactor technology.

4.3. Sodium leaks

After this brief overview, another incident occurred which had serious repercussions on the French fast neutron reactors. The incident took place a thousand kilometres away from the Phénix plant, in the **Almeria solar plant**, in southern Spain. 70 tons of sodium are used here, to transport the solar energy which the mirrors concentrate on a boiler to the steam generator which turns a 500 KWe generator. In order to work on the sodium circuit, the operators cool the piping to freeze the sodium at ambient temperature, then they cut the circuit. But, on 18 August 1986, a problem occurred with the formation of the solid sodium plug and pressurised sodium spewed through the cut made, splashing off the nearby steel structures and causing a fire in the hall. The metallic beams distorted and a hole was blown in the roof. The temperatures reached an estimated 1200°C during approximately fifteen minutes. Fortunately there were no fatalities and the workers sustained only slight injuries.



Almeria solar power plant



This use of the solid sodium plug has always been prohibited at the CEA and EDF plants, where operators routinely drain the entire circuit prior to opening it for works. Nevertheless, for the French Safety Authority and its technical support, this accident clearly showed that under certain pressure conditions and depending on the shape of the opening, a sodium jet can be formed of drops, which has more intense combustion than if the sodium poured out and burned spread out in a pool on the floor. Consequently, the operators of sodium-cooled fast neutron reactors were required to study the maximal occurrences of breaks in the biggest pipes accompanied by spray sodium fires (cf. § 5.3).

Yet the Phénix plant, like the other reactors which used sodium, already possessed **significant experience** with sodium leaks. Since the facility had started up, there had been some twenty leaks, approximately one per year of power operation^[3]. The most damaging incidents were those involving the intermediate heat exchangers (cf. § 2.4 and 3.6). The other incidents caused only very limited production losses.

All sodium leaks are notified, either by ad hoc systems (beaded wires, spark plug type leak detectors, sodium aerosol detectors, smoke detectors), or by the field operators whose practiced eye detects whitish traces which are a sure sign of a small sodium leak on secondary equipment which does not have leak detectors. In most cases, the leak is detected before the sodium has the time to ignite in contact with air. The largest leaks are those which occurred on the butterfly valve at the inlet to the reheater on steam generator N. 2 in 1974 and 1975 (cf. § 2.3). Two different times, approximately twenty litres of sodium drained into the insulation, with no sodium fire outside.

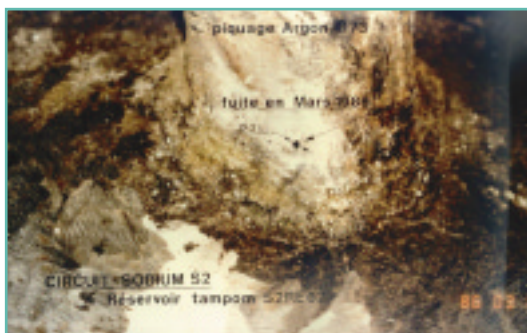
[3] This frequency of occurrence has remained constant.

One unusual case is the leak which occurred on the N. 2 secondary circuit buffer tank. In March 1986, argon leaked from the top of the tank, at the point of the level measurement nozzle. This part of the tank does not have sodium detectors. The operators were alerted after observing the abnormal increase in the flow of the argon feeding the tank. Investigations of the circuits showed traces of white powder on the insulation, which turned out to be sodium (as confirmed when they turned red with the application of phenolphthaleine). The gas flow carried the sodium aerosols which accumulated in the insulation. It was seen that the sodium aerosols had corroded the tank wall, which was then repaired in a difficult operation which required welding new metal on the spherical tank wall.

More serious consequences arose from a **leak on a tee-piece** at the inlet to the reheater on steam generator N. 3 on 5 May 1986, when all eyes were focused on the serious nuclear accident which had just taken place on reactor N. 3 at the Chernobyl nuclear power plant in Ukraine. The beaded wires on that piping portion showed an electric contact. However, since there was no sign of any leak outside of the insulation, the plant was maintained in operation until the scheduled shutdown on 19 May, and surveillance was stepped up with frequent inspections of the piping. On 21 May, the piping insulation was partially removed and solidified sodium was found inside. The pipe was drained, and investigations got underway.

A few dozen kilograms of sodium plus the insulation had created an extremely hard mixture that had to be chiselled off. The largest lump had formed on the upper tee crown. An approximately 12-centimetre crack had formed on the weld welding the tee to the steam generator inlet header. The tee was then cut away for later expert appraisal. The





Sodium leak on a T-piece

technicians then inspected similar equipment on the other circuits and found no defects. The plant started back up on 19 June with N. 1 and 2 secondary circuits in service. The tee-piece was replaced in early August with a new, specially made tee and the plant once again reached full power output on 23 August.

The investigation of the tee showed that the initial weld displayed manufacturing defects which evolved under the operating stresses. The weld then leaked several thousand hours, soaking the insulation. The mixture corroded the stainless steel pipe, particularly along the periphery of the wet sodium area, to a depth up to 7 mm (the tee is 19-mm thick). The corrosion mechanism was analysed in depth, and resulted in new developments at the time of the reactor safety upgrading (cf. § 5.3).

Leak surveillance was stepped up using permanent monitoring of the trace heaters. For, during a sodium leak, the heating element is generally destroyed by the corrosion products. In addition, as insulation was removed, each circular weld was wrapped with leak detection beaded wires. And, the instructions to follow in the event of alarm were intensified. It was prohibited to remain more than three days' time in the presence of an assumed unexplained defect. Within this imposed three-day period, investigations must be carried out to confirm or invalidate

the alarm (partial removing of insulation, portable sniffer (Na 101: flame spectrometer) to detect the presence of sodium aerosols, electrical tests ...). The suspected circuit must be drained as soon as the alarm is confirmed or if it is not invalidated during the three-day period.

An **operating error** also caused a sodium leak in October 1988. This is a good example of an incident which can be caused by human error on valves and fittings. During the early filling operations on the N. 2 secondary circuit, with the plant shut down, two field operators in charge of aligning the circuits operated a valve which was supposed to be in the closed position and which seemed to them to be in the open position, as compared with the position of another valve. They immediately realised that their action had actually opened the valve and enabled sodium to flow into the piping, and they immediately closed the valve. However, they did not report it, and were unaware that the valve,

A sodium leak

In the night of Friday 5 to Saturday 6 October 1984, a little after midnight, the systems for detecting the presence of sodium aerosols in the steam generator building alerted the operators. A small fire had been detected. The reactor was shutdown and the secondary cooling system concerned drained 4 minutes after the alarm. At the same time, the operating agents put out the sodium fire with portable dry chemical-type extinguishers. When the fire brigade arrived on site, less than a quarter of an hour after the alarm, everything had already been dealt with. The cooling process was started for the reactor. The engineers and technicians necessary (insulator, electrician, sheet metal worker, welder, operator and gammagraph analyst, ...) were called out and appeared at the power plant as the interventions took place. Insulation of the pipes was taken off and a crack was found on a knee. The part was changed during the night of Saturday to Sunday. At midday, the system was once again filled with sodium and the start-up procedures launched.



which had been installed approximately one year, was in fact a sealing disc that their action had torn. The sodium therefore continued to flow toward the sodium-water reaction products relief tank throughout the entire filling operations of the secondary circuit. The operators on the following shift observed that the filling was not taking place at the usual rate, and reported this finding at the next shift change.

The new shift operators had no time to lose, for twenty minutes later, several alarms went off, signalling the presence of sodium upstream of the N. 2 steam generator evaporator bursting discs, setting off the automatic actions in the event of a sodium-water reaction (which were not necessary, since the steam generator was empty). Hastening to the site, the operators witnessed sodium burning at the bottom of the discharge stack and used the Marcalina powder extinguishers to put it out, rapidly backed up by the fireman on the Marcoule site. Analysis of the situation revealed that the sodium had filled the dump tank, then started up the connection toward the bursting discs (which set off the alarms), and the stack, where it escaped through a packing gland. Approximately 200 litres of sodium had escaped and were recovered in a metallic leak pan at the base of the stack. 400 kilograms of Marcalina powder were used. A drain valve on the secondary circuit was opened, which stopped the arrival of the sodium and enabled the fire to be totally extinguished in less than one hour.

The repair operations took ten days. The portions of the piping which had been in contact with the sodium had to be cleaned. Above all, the approximately one hundred tons of sodium in the dump tank had to be transferred to the storage tank using mobile equipment. Approximately 500 litres of solid sodium remained at the bottom of the separator. The

incident was extensively analysed by the various shift teams to stress the importance of following instructions and reporting every action. Start-up activities then got underway and the plant reached rated power on 31 October.

4.4. The second ten-yearly regulatory inspection and overhaul

Operations at the Phénix plant then went on to be totally outstanding. The plant produced 177,779,000 kWh in December 1988, an absolute record high. The 45th and 46th irradiation cycles (70 days each) took place with no production stoppage other than the 8 day scheduled refuelling in January 1989. The satisfaction of a job well done therefore set the stage for the second ten-yearly inspection.

The date for the ten-yearly regulatory inspection, scheduled for spring 1989, was a compromise between the date for the complete turbine overhaul which should have occurred in 1987 or 1988 and respect for the pressure equipment regulations which required retesting in April 1990 at the latest. The power plant was shut down on 28 March 1989, after the 46th irradiation cycle. The initial calendar, based on the period of time required for maintenance of the turbo - generator set, called for a shutdown of thirteen weeks. This was extended by six weeks due to the defects discovered during the non-destructive testing on the steam generators and on the turbine. The low-pressure rotors on the turbine and on the generator had to be replaced, and repairs were required on the welds on the steam generator reheaters. After spending several days solving unexpected problems with the new generator exciter, the plant returned to rated power on 3 August.

During the regulatory inspection period, a seventh team was created in the operations section, working normal hours and made up



of staff assigned from the shift teams. The new team prepared the blocking, based on a logic of very large blocks, and managed the timetable and operations sequences. It organised the requalifications and power resumption tests. During the regulatory inspection, the three secondary circuits were simultaneously drained of sodium for the first time, during a nine-day period. The analysis of this situation and a prior test showed that the emergency cooling circuit and the primary sodium auxiliary circuit cold trap were sufficient to remove residual power from the core, and that in the event of failure of one or the other, there was ample time available to fill a secondary circuit. Throughout this period, the reactor diverged two separate times in the framework of preparations for the next cycle and the experiments which would be conducted during it.

Over 400 service providers from some fifty sub-contractor firms worked on the power plant. On the busiest days, there were up to 250 outside workers working simultaneously on the site. Almost all the doses absorbed by the workers came from two work sites: inspection of the inner space between the reactor block vessels (0.018 h.Sv) and the work on the primary sodium purification circuit cells (0.040 h.Sv).

The work on the **reactor block** essentially involved verification of the condition of certain sensitive materials after fifteen years of operations. One measurement involved equipping the 21 upper hangers in the main vessel with an instrument to measure the support pad movements. This equipment provided measurements during the various temperature regimes in the reactor block, thus enabling estimation of the potential effects of ageing on the pad material.

The inspection of the inner space between the main vessel and the safety vessel was



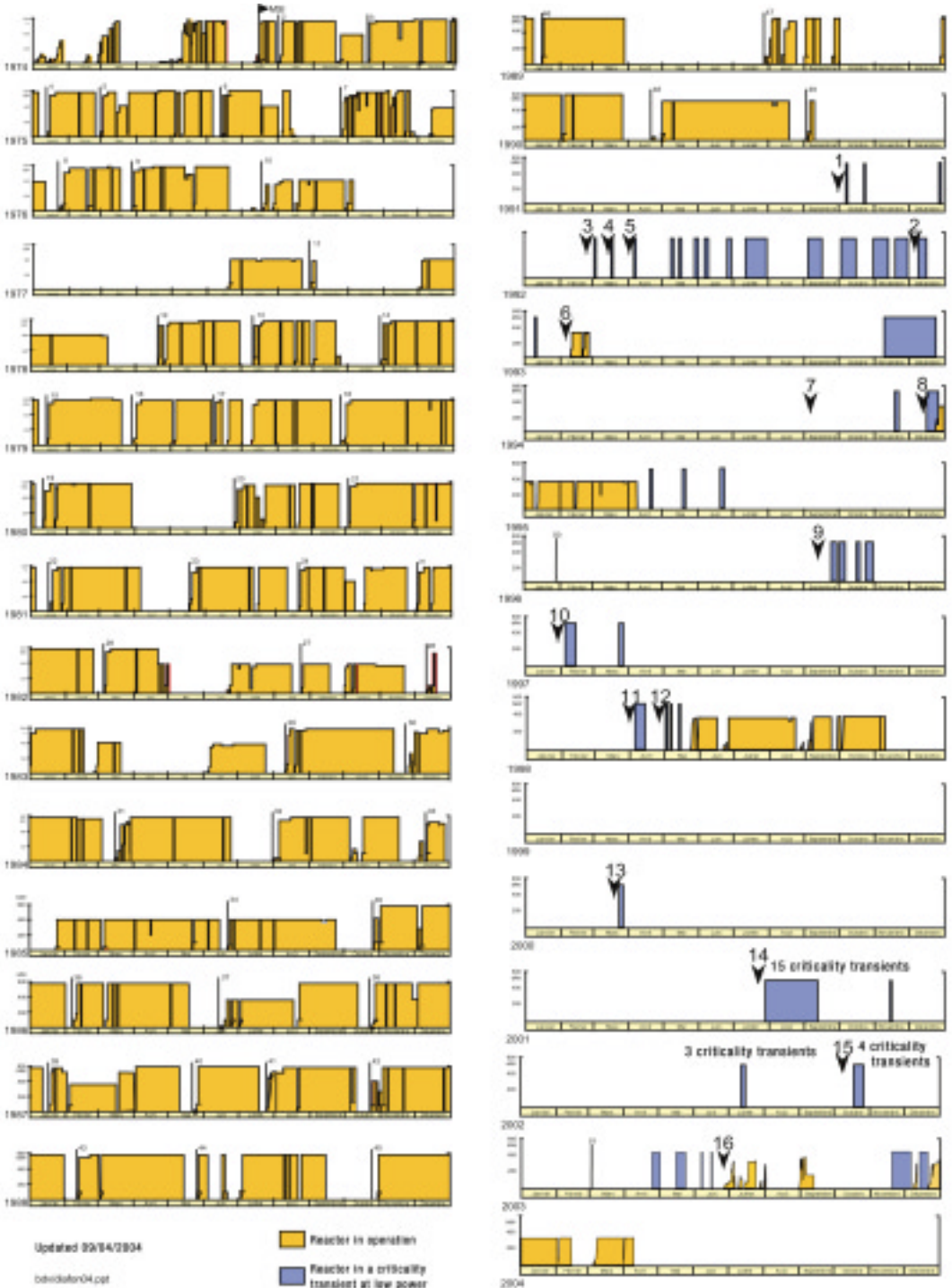
Space between main vessel and containment vessel

organised once this zone was aired and the reactor temperature lowered to 150 °C. The insulation was in excellent condition. A feasibility test was conducted for ultrasound inspection of the connecting welds between the roof and the main vessel. Due to the trouble-free access, the low ambient radiation and the effortless removal of the heating panels covering the welds, the test was extended, and more than half of the welds were inspected. In order to select which welds were to be inspected, the manufacturing reports were examined and those that raised any “doubts” were tested. However, the tests applied did not reveal any indications that pointed to defects generated during the years of operation. Simultaneously, visual and ultrasound inspections were conducted on one-tenth of the storage drum vessel welds^[4].

[4] These tests were completed in March 1991.



Phenix power plant - operational chart



Over one hundred welds were checked on the **secondary sodium circuits**, bringing to nearly half the number of welds which have been examined since the power plant started operation. Five welds presented defects. Accordingly, for three of the welds, the corresponding piping portions were cut and replaced by new elements. In the two other cases, analysis showed that there was no risk in leaving the weld as it is.

One experimental decay heat removal circuit (CREX) was installed to improve the reactor's capacity to remove decay heat and avoid any extended or definitive unavailability in the case of loss of the three secondary sodium circuits. This experimental circuit consisted of a sodium-air exchanger installed next to the steam generator building, and of piping connecting it, via an electromagnetic pump, to an intermediate exchanger of the N. 2 secondary sodium circuit. The Safety Authority did not grant the connection authorisation and the tests were postponed^[5].

The **steam generators** were subjected to the regulatory hydraulic tests in the same conditions as in 1980, after first verifying that these repeated conditions (pressure equal to 1.5 times the working pressure) would not harm the equipment. In May, the tests and checks were satisfactory, except for the reheater on steam generator N. 2. Here, a water leak was detected on the bimetallic connection weld at the inlet to a module of the steam header. After removing and redoing the weld, a new hydraulic test was conducted, and this time, leaks appeared in the same area on five other modules.

All the equivalent welds throughout the three steam generators (144 welds) were subjected to non-destructive tests, which confirmed the results of the previous trial, detecting defects on the welds in the five modules. The expert evaluations which were then conducted attributed these defects to a caustic corrosion mechanism resulting from accidental sodium intake in the zone during the sodium - water reaction in April 1982. The welds on steam generator N. 2 were replaced by a sleeve which contained the bimetallic weld done in the workshop. This sleeve was then welded in place on site using two homogenous welds. The reheater on steam generator N. 2 then successfully passed the new hydraulic test in mid-July.

Just as during the preceding inspection in 1980, the **turbo generator** was entirely taken apart and evaluated. The high and medium pressure cylinders were in good condition and there were no traces of erosion on the blades. Many cracks on the low pressure rotors were however detected. It was first thought that the corrosion which caused them was also due to the sodium entering the water - steam circuit in April 1982. However, later analyses showed that soda had entered the water circuit at the same time as the hydrazine, for the hydrazine had indeed been delivered to the plant in tanks which had also been used to transport soda.



Turbine rotor blades

[5] The tests took place in 1992, when the reactor was shut down. Then the CREX was abandoned, for there was little motivation to continue the operations. First, it did not add much to demonstrating the safety, since it was connected to an intermediate heat exchanger. Then, the sodium - air exchanger did not operate satisfactorily after the initial tests and was finally dismantled in 1999.



The two rotors were replaced by identical equipment taken from the EDF power plant in Ambès, where only two units continued to operate. The generator rotor was also replaced by a rotor from the same plant, because the Phénix rotor had oxidised between the rings and the winding which were impossible to take apart on the site.

The old, obsolete voltage regulator was replaced by a model used on the 900 MWe generators in the EDF nuclear reactor series. This required modifications to the exciter (work done with new parts and an inductor

taken from the EDF Nantes-Chevire plant). New tubing was also installed on the condenser, which meant that the 22,000 tubes through which the Rhône river water circulated were changed. The steam transformer was eliminated. From this point on, the Phénix plant directly used the steam supplied by the Marcoule boilers for its preheating needs. Lastly, over 150 pressurised tanks were subjected to regulatory tests.

4.5. Negative reactivity trips (A.U.R.N.)

After the second ten-yearly regulatory inspection, the Phénix reactor experienced 4 extremely fast and high amplitude oscillations in the signal from the power range neutron chamber, which triggered automatic trip when the negative reactivity transient threshold was reached. These four events, which took place in August and September 1989, then in September 1990, were labelled A.U.R.N. They occurred on the 6th of August, the 24th of August and the 14th of September 1989 and on the 9th of September 1990. The first three occurred at a rating of 580 MWt, and the last one at 500 MWt, after operating periods between 4 and 15 days.

The recordings from the power range neutron chambers, located beneath the vessel, show a signal in the form of a double oscillation. The following are distinguished:

- A sharp, practically linear drop, triggering the automatic trip due to overshooting the negative reactivity transient threshold, to a minima reached after 50 milliseconds,
- Symmetrical rebuild up to a maximum below the initial level,
- A new drop, but with less amplitude,

Instrumentation for the sodium pipes

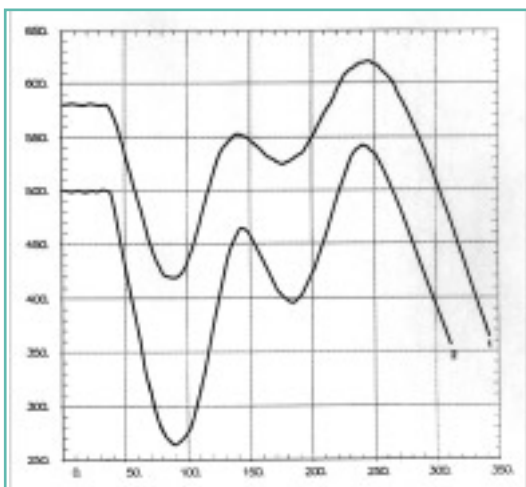
The electricity preparators define the instrumentation for the sodium pipes (that is, the leak detection wires of steatite beads, the thermocouples and, above all, the heating cables, as they require very precise sizing calculations and lead to very detailed specifications – their length in particular must be specified to within a few centimetres). These electricians are responsible for supplies, having to take into account delays of several months for the heating wires as they are all made to measure. The delays for the other materials are shorter, which in particular makes it possible to have the leak detectors manufactured by a centre for handicapped people in the Avignon region. Finally, they supervise the installation work, as well as the checks and commissioning tests.

This means a considerable amount of interfaces that need managing (mechanics research departments, sheet metal companies, scaffolding, lagging, electricity companies, company controllers, the controllers of the plant, operators and so on). There are always a few incidents along the way that have to be dealt with (last minute changes in the definition of the pipes, manufacturing behind schedule with a supplier, work site problems, the most common of which are thermocouples or a leak detection wire totally covered with lagging, making it impossible to connect it to the outside). As this activity is always the last stage on the schedule, there is often a lot of pressure, putting relations under great strain and resulting in shouting matches that are sometimes quite violent! But in the end, the instrumentation is installed, on time, and is fully operational.



- A second positive peak, slightly exceeding the initial state, approximately 200 milliseconds after the beginning of the occurrence,
- decrease of power toward zero following the rod drop triggered by the trip order.

The oscillation shape was duplicated from one event to the next. The amplitude varied fairly sharply. The power drop was 28 % on 14 September 1989 and 45 % on 9 September 1990. There were no precise recordings from the two first A.U.R.N. However, it appeared that the amplitude of the first A.U.R.N. was similar to the 9 September 1990 event, and the second to the 14 September 1989 event. On the assumption that the signal represents actual power variation, the maximal occurrence show a 320 pcm loss of reactivity and a rebuild to 37 pcm above the initial level^[6].



Reactivity curve of negative reactivity trips

The signals from the neutron measuring channels were accompanied by several other signals with variable informational content: cover gas pressure, geophone, presence of methane in the cover gas argon, pressure at

[6] In the Phénix reactor, the proportion of delayed neutrons is 0.325 %, which is $\beta = 325 \text{ pcm}$ (1 \$).

the primary pump discharge, position of one of the six control rods, ... None of the signals, except for the geophone signal, was clearly observed during the four A.U.R.N. Yet, their chronological coincidence with the reactivity signal, insofar as the acquisition frequencies of the signals can be used to judge, tends to show that they indeed have a common origin.

The **early explanations** submitted after the first two A.U.R.N. refer to interference on the measurement channels, which had been modified during the ten-yearly regulatory inspection in 1989, though no particular sensitivity could be shown for any of the channels.

Investigation of the negative reactivity trips which had occurred since the reactor's first divergence showed that two shutdowns, on 6 April 1976 and 6 June 1978, were similar to the 1989 and 1990 events. It was also established that the previously accepted explanation about a control rod slipping was wrong. Thus the possibility that A.U.R.N. had occurred earlier in the existence of the power plant cannot be excluded.

The elements gathered after the third event in summer 1989 led to attributing the variation in power to the passage of a volume of gas in the core. This explanation seems consistent with both the observation of an indication of excess pressure in the cover gas and with the possible plugging of the special venting sub-assemblies in the diagrid. Without waiting for a possible injunction from the Safety Authority, the operator then decided to stop operations at the plant during the time it took to gather the additional information. After analysing this scenario, and its consequences, and taking preventive measures, the reactor was authorised to start up again in December 1989. Two irradiation cycles took place, during which the Phénix plant passed the point of 20 billion kWh produced since its start-up.



The explanation considered above was invalidated by the occurrence of the fourth event in September 1990. The CEA then set up a very large **investigation** program, with the creation of a committee of 15 French and foreign experts, who were assigned to:

- Examine every possible cause for the reactor anomalies,
- Provide elements of response for these anomalies,
- Examine every possible consequence in the event that these abnormal conditions should occur in different conditions,
- Make proposals for preventive measures.

This committee coordinated the overall investigations which marshalled the forces of the Phénix plant itself, and several different CEA

research sectors in many different fields, such as neutronics, hydraulics, mechanics, chemistry and others. During this time, the plant did not operate. Tests were carried out in the framework of the re-evaluation of the total loss of decay heat removal system. Non-destructive testing was also conducted, and repairs were made to the piping and secondary circuit equipment.

The expert evaluation came to an end in 1991, and presented the following two basic findings:

- The phenomenon which triggered the anomalies has not been clearly identified,
- However, there is a strong presumption that the variation in reactivity was due to a radial expansion of the sub-assemblies followed by a return to the centre (outward movement phenomenon).

One, two, three negative reactivity trips. ...

Sunday August 6, 1989, 9 h 05: emergency scram of the reactor due to negative reactivity. Two of the three neutron chambers from which this signal is computed had been replaced during the 10-yearly inspection that finished on July 29. A time test for the drop of control rods was performed and their efficiency measured: everything was as it should be. A physical phenomenon associated with operating the reactor is hard to envisage given the rapidity of the variation in the reactivity. It was almost certainly an electrical or electronic disturbance that led to the unscheduled shutdown. The reactor was diverged on the Monday morning and reached full power on the Tuesday afternoon. The Safety Authority, which esteemed that the analysis was insufficient, then asked for the reactor to be shutdown. After a few additional explanations, the power plant received authorisation to start-up again on Friday August 11.

Thursday August 24, 1989, 23 h 37: emergency scram of the reactor due to negative reactivity. The neutron chambers were still the main suspects. The splice cases were modified (addition of an earth connection) and then tests on immunity to parasites were conducted, which validated the advantage gained by the modification. The Safety Authority authorised the start-up of the reactor on Thursday August 31 and the plant reached full power the next day.

Thursday September 14, 1989, 14 h 48: emergency scram of the reactor due to negative reactivity. This time, the signal from the neutron chambers was recorded by a rapid acquisition system installed especially at the start of the month. Low power tests (between 5 and 40 MWth) were undertaken to try to reproduce the signal, but without success. The search for the physical phenomenon that could be behind these shutdowns continued. At the same time, a problem on the turbine axis ("bowing") mobilised considerable effort. The power plant started up again and reached its full power on Wednesday September 27.

On Friday September 29, many specialists from the power plant and the centre in Cadarache met to analyse the physical phenomena that could be behind these automatic shutdowns. It appeared that a bubble of gas going through the core, in its peripheral area, could provoke a negative disturbance in reactivity. In addition, certain operations that took place during the 10-yearly inspection (variations in the speed of the primary pumps) could have led to an accumulation of gas under the diagrid. In the face of such uncertainty, it was decided to shutdown the reactor and prepare a check test of the special diagrid venting sub-assemblies, whose role is to prevent gas from accumulating in such a way. The reactor was shutdown on Sunday October 1 at 11 h 50.



Furthermore, the safety analysis conducted on the basis of the various initiator phenomena showed that none of the phenomena would lead to a damaging situation for the reactor. The following approach was then proposed:

- First, substantially reinforce the measurement and recording means for reactor surveillance so as to obtain the information required to identify the cause of anomalies, if a new event should occur,
- In order to obtain new elements with respect to certain scenarios, to test the instrumentation, and to verify reactor and core behaviour, conduct tests at very low power at first, then a 10-day phase at power operation.
- Then, proceed to start up the reactor to continue the 49th cycle.

Since 1991 monitoring of the reactor has increased:

- Additional instrumentation in several fields: neutronic measurements, the SONAR device which follows the movement of the core sub-assemblies, acoustic detection in the core, measurement of the magnetic field in the vessel, structure displacement, ...
- Specially designed fast measurement acquisition means were installed,
- Special organisation was set up.

The objective was to provide permanent surveillance of the reactor in order to detect any and every precursor anomaly or event occurring before an A.U.R.N. And, in the event of a new occurrence, to have access to a set of data which traces back to the origin of the phenomenon.



The **tests** at very low power took place between October 1991 and March 1992. Their goal was to:

- Verify the condition of the core, from the neutronic standpoint,
- Confirm the reactivity insertion sequence during the rod drop in a normal automatic trip, to get a more accurate interpretation of the A.U.R.N. signal,
- Gather elements about the scenarios involving a passage of gas or a primary hydraulic transient,
- Describe the dynamic behaviour of the reactor instrumentation and validate the additional instrumentation which was developed, including the SONAR device, installed above the core and designed to perform continuous measurement of any lateral displacement of two peripheral sub-assemblies in order to detect any outward movement phenomenon.

These tests also led to dismissing several assumptions which had been made to interpret the anomalies. They confirmed the proper behaviour of the core and of the primary hydraulics.



4.6. Causes for A.U.R.N.

The expert evaluation work conducted within the scope of the A.U.R.N. analyses identified the various causes which could be at the origin of these occurrences.

The first assumption made was the **existence of interferences** in the reactor measurements. However, the neutronic measurements recorded during the A.U.R.N. display a high degree of consistency: up to seven measurement channels delivered the same signal using different technologies (current and pulse measurements) and are made up of different chambers, electronics and acquisition systems.

Regardless, the briefness of the phenomenon and the reproducibility of the signals raises questions about the reality of the reactivity variations. In-depth investigations were carried out on any phenomena likely to disturb the channels.



Neutron chambers' electronic cabinets

- Electromagnetic disturbance likely to be generated by the electrical supply, the earthing systems or electromagnetic radiations. It was found that the equipment was well designed with respect to the risk of interference excitation, yet it was difficult to extend such assurances to all the measurement material used occasionally,
- Magnetohydrodynamic (MHD) phenomena. Although it is difficult to issue a definitive conclusion due to their complexity, no elements were identified that could lead to think that a magnetohydrodynamic effect could be at the origin of a neutronic chamber disturbance, or of forces on the core sub-assemblies, causing an outward movement.
- Changes in the neutron propagation conditions. None of the various phenomena likely to modify the transmission of the neutron flux between the core and the chambers under the vessel (an obstacle between the core and the chambers, ...) could account for the amplitude nor, in most of the cases, for the kinetics of the A.U.R.N. signal.

In the end, even though some people continue to support this assumption, no physical phenomenon has been shown that was likely to induce variation in the electric signal from the channels that was not related to a real variation in power. The primary question which remains relates to the amplitude of the reactivity variation, reconstituted from measurement. Furthermore, the information gathered from the staff present during the A.U.R.N. (the third A.U.R.N. was the only to occur during daytime working hours), has produced no useful leads. Although the measuring scale changes made in the reactivity meter at the time of the bimonthly tests on the control rod movement have been suspected, no firm conclusions have been reached.

In the assumption of a **real variation in reactivity**, three generic types of phenomena could be at the origin, either alone or in combination: the effects of sodium void, the movements of the control rods, the core movements. Fuel burn-up and the effects of temperature were eliminated because they were not compatible with the kinetics of the A.U.R.N. It was also proven that effects related to absorbing, moderating or reflecting parts were not realistic.

Analysis of the mechanisms behind the void effect, which included gas passage (condensable or not) in the core, cavitation in the sub-assemblies, boiling sodium, hydrocarbon vaporisation, do not provide a scenario which is capable of explaining the A.U.R.N. signals.

Investigation of the causes involving the control rod mechanisms, the absorber pin bundle, the rotating plug which supports the rods has led to eliminating the hypothesis of a rod movement as the sole cause of the reactivity signal. The reactivity effects are low with respect to the amplitude of the A.U.R.N. signals, and the required accelerations would have been very high. Of the various plausible origins for the reactivity variations, it fairly quickly appeared that solely the group of **mechanical core movements** could cause the reactivity transients at the speed and amplitude which were observed. In particular, investigative focus concentrated on the movements causing radial expansion of the sub-assemblies (outward movement) followed by inward return.

These investigations took a double approach:

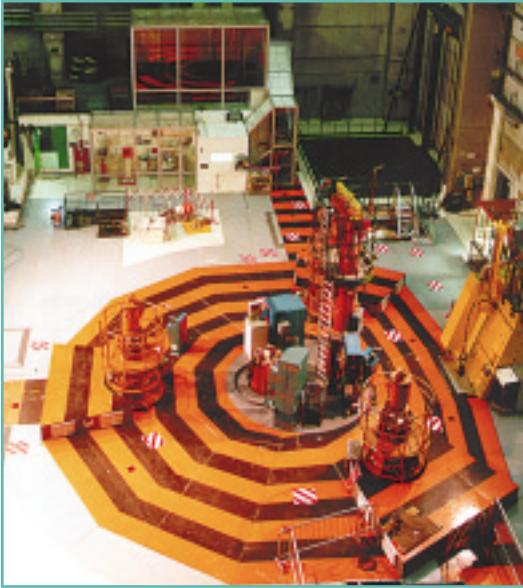
- Extensive modelling of the outward movement coupled with tests on models, independently of any causes. It was shown that under stresses from a pulsed source inside the core, with the right parameters (form and period of the force field, source alti-

tude, etc...) it is possible to reproduce a reactivity line similar to the one observed. Calculations also show that any movement of the sub-assemblies started up by transversal excitation from the diagrid would cause an increase in the reactivity. Consequently, this type of scenario was rejected.

- Search for a mechanical or hydraulic mechanism capable of inducing core movement. The main mechanisms investigated were: those based on abnormal behaviour of core block structures; spontaneous reconfiguration of the core due to an unstable condition, due for example to obstructed bending; a pressure hammer from gas passing in a pump; gas expansion occurring above the core, under the core cover plug, or a loss of absorber pin tightness; oil passing in the core causing a mechanical effect by vaporisation and cracking. None of these mechanisms were shown to be a plausible initiator of the incidents.

A more general analysis of the reactor block behaviour to impulse regardless of the origin, has shown that an axisymmetrical vertical impulse applied to the upper part of the reactor, through a combination of a vertical movement of the core and a outward movement, could theoretically induce a signal in the A.U.R.N. shape. However, this type of interpretation requires the implementation of much too high of energy to be realistic. Nevertheless, the hypothesis of a sharp movement by one of the reactor structures remains the most subscribed to theory. Certain researchers, some of whom are now retired, continue to work out scenarios to explain the phenomenon. Their theories are tested and if need be, additional measurements are taken on the reactor. Unfortunately, so far, no new decisive elements have been identified.





The hall of the reactor building

With respect to reactor **safety**, the A.U.R.N. have raised three main issues.

When the control rods dropped, did the incidental reactivity related to the anomaly remain sufficiently low so as not to threaten reactor safety ? It was shown that the core regained practically normal reactivity after 200 milliseconds.

Could the reactivity anomalies be the symptoms of damage of mechanical origin affecting the reactor structures, in particular the core support structures, due, for example to damage accumulated since the reactor was commissioned ? No potential initiator has been identified. And surveillance of the reactor structures has not revealed any deviation from normal. This finding was backed up by the inspections of the internal structures conducted between 1999 and 2001 for the reactor lifetime extension program.

Under modified conditions, could the mechanisms liable to be at the origin of the A.U.R.N. lead to reactivity insertion which

could affect core integrity and cooling of the fuel ? Taking into account all the uncertainties affecting the calculations, it was shown that the maximum reactivity insertion that can be taken into account, able to be caused by sodium void effect or sub-assembly movement during springback after outward movement, does not present any risk for the fuel.

All in all, substantial research and testing resources were devoted to seeking the origin of the A.U.R.N., including a total of 200 men per year and the publication of nearly 500 documents. The investigations have shown that the events were not the sign of internal reactor structure damage, in particular the core support structures, and that regardless of the mechanism initiating the incidents, there was no threat to reactor safety.

The final explanation involves outward movement of the sub-assembly lattice under the effect of an impulse located in the core, followed by a centripetal return. However, no single complete scenario in line with the observations, has been able to be established. Lingering doubt remains about the representativeness of the measurement with respect to the amplitude of the core reactivity variation. Doubt even remains as to its physical reality. However no element has been identified, and continued reactor surveillance since the 49th cycle has provided no new information. It is highly probable that the only way to make any new progress in the identification of the cause of the A.U.R.N. would be to have a new A.U.R.N. occurrence. On the other hand, it is also probable that the A.U.R.N. occurrences were related to the simultaneous presence of objects in the core (experimental sub-assemblies ...) that have since been removed.



Mechanic or electric?

When you talk with an electrician who tried to find the cause of the negative reactivity trips, he will tell you that the power supplies to the sensors which recorded the signal are so different in nature that it is impossible that an electric or electronic failure could have produced all the signals recorded during the trips of 1989 and 1990. When you talk with a mechanic who tried to find the cause of the negative reactivity trips, he will tell you that the energy needed to create a significant movement in the reactor would be so great that it is impossible that a structural defect could have produced a movement in the core that generated the decrease in reactivity recorded. The truth is almost certainly somewhere between the two.

4.7. Core materials

The means available at the Phénix plant have been used to conduct major research and development work on the core materials used in the fast neutron reactors. Research work reached a peak in the 1990's, with the expert evaluations of the driver and experimental sub-assemblies that had been irradiated during the entire previous decade.

The objective was to obtain ever-higher specific burn-up in order to lower the fuel cycle cost, and for that to increase the doses that the pin clad and sub-assembly structure materials could withstand. These doses reached extremely high values of 100 to 200 dpa. Under irradiation, the materials underwent deformations associated with swelling and creep, and they experienced structural evolution, primarily seen in the deterioration of the mechanical strength and the resulting embrittlement. It was crucial to find materials with the best performances.

It quickly became apparent that the sub-assembly **residence time** was restricted by the excessive strain (elongation, deformation, bowing at the top of the peripheral sub-assemblies, ...) affecting the essential structures such as the cladding and the hexagonal wrapper. Simulation methods implementing high-energy electronic irradiation were used for preliminary material classification, however, the essential research bases were provided by the irradiations of the experimental sub-assemblies and the rigs containing samples of different materials, in the core of the Phénix plant.

These irradiations were generally followed by a series of non-destructive tests such as elongation and diameter measurements taken on the cladding, and bowing measurements on the hexagonal wrappers. These measurements were taken on a great number of sub-assemblies and pins in the irradiated element cell. They enabled a statistical evaluation of the deformations across the wide diversity of irradiation conditions, thus providing indirect access to creep and swelling deformations.

Then, mechanical tests (tension, impact strength, creep, density ...) were conducted in the Cadarache laboratories on the hexagonal wrapper samples, on the sections of cladding with the fuel removed, and on the samples irradiated in the rigs. In addition, on a smaller number of samples, electronic microscopy examinations provided greater in-depth information on damage-causing swelling events and microstructural phenomena.

Three **classes of materials** were studied as described above.

Austenitic steels have been the subject of the most research in France and abroad, where fast neutron reactors have been



developed. The main phenomenon limiting austenitic steel lifetime in reactors is swelling under the accumulated effect of damage caused by the fast neutrons at temperatures of approximately 500 to 600 °C, which is the clad and hexagonal wrapper temperature in normal operating conditions. Irradiation also contributes to the significant modifications in the mechanical strength and ductility of these steels.

The valuable effect on the swelling phenomena derived from the addition of small amounts of titanium and silicon was quickly observed. By adjusting the major elements (chrome and nickel) in the composition of austenitic steels, new grades were obtained which demonstrated much better swelling

behaviour. Metallurgical state modifications (cold working), improved manufacturing specifications, and better control over the entire production chain led to the 15 - 15 Ti steel which is the current reference for fuel pin cladding capable of achieving an average dose of 115 dpa for maximum volume swelling of 6 %. The steel becomes too brittle beyond this value. Optimisation of the specifications led to defining the AIM1 steel which was used to make the latest fuel loads for the Phénix plant.

The nickel alloys, such as the Inconel, show much better swelling strength and excellent thermal creep behaviour. However, irradiation causes extreme brittleness and research on nickel alloys has been suspended.

The use of martensitic ferritic steels was not initially retained due to their thermal creep behaviour, inferior to austenitic steel. However their centred cubic crystallographic structure endows them with great stability under irradiation and good swelling strength. Therefore these steels were first tested for hexagonal wrapper applications, where there are fewer thermal creep problems. EM10 steel was the stablest throughout the entire range of operating temperatures. In irradiated condition, it also has a ductile/brittle transition temperature below ambient temperature.

New manufacturing processes have been developed in order to improve the behaviour of martensitic ferritic steel at high temperatures. Experimental pins were irradiated in the Phénix plant reactor core, and the post-irradiation tests confirmed the good dimensional stability under irradiation. However the tests also showed undesirable brittleness. As a result, new alloys are currently being developed.

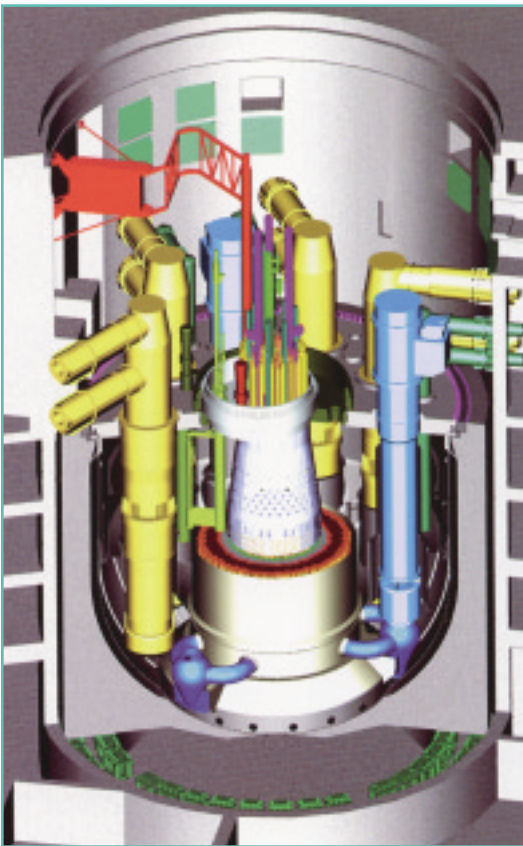


Diagram of European Fast Reactor project



4.8. A turning point

Although reactor power operation had become increasingly regular, taking full advantage of the feedback gained since the start-up, particularly due to the improvements made to the facility and the core materials, the four negative reactivity trips which occurred in 1989 and 1990 disrupted the thriving dynamics underway. From then on, innumerable safety files had to be presented and analysed by the Safety Authority before each new irradiation cycle could be authorised.

The executive secretary for the plant

The secretarial department of the Phénix plant, although organised in a "pool" for a long time, has always had a secretary working exclusively for the direction. With remarkable continuity, and paying scant attention to the many changes in director, the executive secretary has thus become a living memory for the power plant.

Although the tasks have changed little in thirty years, like everywhere, progress in terms of material has been considerable: from electric typewriters to micro-computers, from telex to fax, and then e-mail; filtering telephone calls, receiving contacts, typing monthly reports, dealing with the administration of the Operating Committee, classifying documents as they come in to, or go out of, the power plant (using an organisation system copied from the outset from that used at EDF and which has changed little since), organising ties with American, British, Japanese and Russian counterparts... a whole range of tasks that are essential to the good running of a unit. And throughout all this, the constant interest in the main objective: guaranteeing that the plant operates correctly, for the efficiency of the secretary plays a key role in helping the director of the plant and his assistants.

The executive secretary's role, however, is not limited to these essential tasks. The executive secretary is also the person to whom all the employees turn to recount their joy and sadness, in the good and bad times at the power plant. It is better to find out more about the current mood before going any further...

Reactor safety with respect to reactivity accidents had been thoroughly re-examined. And, at the end, neither the design choices nor the safety demonstrations were questioned. The sole risk was a reactor shutdown. It had been shown that under no circumstances would the potential causes of



Sodium-fire extinguisher

result in a hazardous nuclear accident. Yet, the failure to determine the exact causes of these trips undermined the confidence in the operator's ability to control the reactor, and turned the issue into a hovering Damocles' sword capable of interrupting power operations at any time.

In addition, the analysis of the sodium fire at the Almeria solar platform forced the visualisation of hypothetical situations involving large leaks of pressurised sodium leading, in the form of streams of sprayed drops, to much higher burn-up rates than those originally designed for. This was one important factor in the reactor safety upgrading, undertaken as Phénix approached its twentieth anniversary.

On the national and international scene in the early 1990's the winds were beginning to shift. Fast neutron reactor development was under pressure. The oldest plants were closing, other projects were halted. The future of the Creys-Malville plant was in doubt. Internationally, only the Russian and Japanese programs were being pursued. Support for electricity-producing fast breeder reactors was dwindling. Outside of the community of reactor operators, fully aware of the advantages inherent to this reactor system, advocates were increasingly few and far between.



André LACROIX

Director of the Creys-Malville nuclear power plant from May 1990 to January 1995 ^[1]

From Phénix to Superphénix

Rapsodie (40 MWt), Phénix (560 MWt), Superphénix (3000 MWt): even though Phénix has fourteen times the thermal power of Rapsodie (which did not generate electricity) and Superphénix has only five times the thermal power of Phénix, the real change of scale, in both technical and human terms, is between the latter two.



In technical terms, apart from their respective sizes, Superphénix differed from Phénix in the design of many of its components: primary auxiliary circuits incorporated in the reactor vessel, double rotating plug, fuel handling, secondary cooling systems, steam generators, cylindrical reactor building, start-stop steam/water by-pass circuit, water circulation systems, turbo-generator sets, fuel storage, instrumentation and control, etc. Though it seemed, on the face of it, a minor point, the lack of a chromatograph to monitor the purity of the reactor argon proved costly when the primary sodium coolant became polluted!

These differences did not prevent the high cost (in time, money and media controversy for Creys-Malville) of taking into account the negative reactivity trips that occurred at the Phénix plant. On the other hand, the safety analyses, theoretical studies and design improvements with regard to sodium fires conducted for Creys-Malville subsequently impacted on the Phénix plant.

In human terms, the ten or twenty engineers who had run Rapsodie or Phénix in the 1970s knew these facilities down to the last valve (or fuel pin). They could have instantly replaced any technician in the field or control room operator. But at Superphénix, given its size, the quantity of materials (equivalent to about two-and-a-half times one of the EDF's 1300 MWe pressurised water reactors) and its functional complexity, detailed knowledge of the plant and the workings of its basic systems was necessarily fragmentary.

But the size and technical "legibility" of Phénix are not enough in themselves to explain the dynamic of its start-up and the success of its first years in operation. What did herald this success was the complete integration of the CEA, EDF and GAAA teams during the design and construction stages and the

direct participation of the operating team at the end of the construction phase and throughout the commissioning tests. And we should add one human factor that was paramount at Phénix: the stability and team spirit – or even commando spirit – of its first operators.

At Creys-Malville, numerous factors prevented the same kind of dynamic developing: the number of organisations involved, the international organisation and industrial arrangements, the very hierarchical relations between the EDF Engineering and Production structures (which are quite appropriate structures for building a series of similar power plants, but much less so for a prototype), the lengthy construction and commissioning process, and the increasing size of the operating team as work progressed (it was initially planned as 250 staff not counting the Handling and Physics departments).

To sum up (and without going into the technical, political and media history of Superphénix and its lamentable closure, for political reasons, in 1997) it has to be said that the context was not at all the same. The Phénix plant was designed, built and commissioned in a France still imbued with de Gaullian spirit, proud of its large-scale industrial creations in general and its nuclear industry in particular. The Phénix fast breeder reactor was in the vanguard of civilian nuclear technology.

Between the start-up of Phénix and the commissioning of Superphénix, the organisation of nuclear safety had greatly advanced. In the 1970s safety was incorporated into the operator's know-how; though rigorous, it was based on fewer elements and required fewer procedures, necessary but often too detailed. And unlike Creys-Malville, Phénix (often called the "Marcoule reactor") never hit the headlines and was never a target for the antinuclear lobby, despite its technical and economic ups and downs, the sodium fires and leaks, negative reactivity trips, etc. As a result, its managers had the good fortune to be able to devote most of their working time to technical matters.

[1] André LACROIX a travaillé comme ingénieur à Phénix de 1971 à 1980, puis a rejoint Creys-Malville.



SAFETY UPGRADING
(1992 - 1998)



Landmarks

- 9 - 26 February 1993 Tests at 350 MWt in order to analyse shutdowns caused by negative reactivity transients
- 31 March 1994 Definitive shutdown of PFR (United Kingdom)
- 5 April 1994 First divergence of Monju reactor (Japan)
- 27 May 1994 Definitive shutdown of Bugey 1 reactor
- 12 July 1994 New decree authorising the creation of the Creys-Malville plant (Superphénix reactor)
- 24 December 1994 49th irradiation cycle re-started
- 7 April 1995 End of 49th irradiation cycle. Boitix 9 experimental sub-assembly reaches 144 174 MWd/t
- 25 July 1996 First divergence of Chooz B1 (N4 series)
- 19 June 1997 French Prime Minister announces closure of Superphénix (Creys-Malville)
- 2 February 1998 Ministerial notification of the definitive shutdown of Superphénix (Creys-Malville)
- 22 May 1998 Start of 50th irradiation cycle
- 24 October 1998 Award ceremony for the American Nuclear Society's Historical Landmark
- 12 November 1998 Discovery of leak from intermediate heat exchanger I and interruption of 50th irradiation cycle

In the early 1990's, Phénix became the senior member of the French nuclear power reactors. The GCR reactors (Chinon A, Saint-Laurent-des-Eaux A, Bugey 1), the EL4 heavy water reactor (Brennilis) and the Chooz A pressurised water reactor had all been shut down. The Phénix plant once again became the subject of a whole new task, which was also being conducted elsewhere on the EDF pressurised water reactors and on some older CEA facilities. This exercise consisted in extending the lifetime of the reactor and conducting the associated safety upgrading.

The Phénix plant had been shut down due to the negative reactivity trips. The operator wanted to start the reactor back up, however many successive prerequisites would be imposed, as new safety analyses developed. At each step of the way, issues were raised by the Safety Authority, and responses were provided, often eliciting new questions. Each requested file required long lead times to perform the essential research and provide the necessary content. And each new request added that much more delay to starting up power operations. The plant focused on the objective of having the very best safety reports in the shortest possible times, and on preserving the capacity to get the reactor operating without overwhelming delay.

This period was to last several years. And throughout the entire time, staff capacities had to be kept up, despite the reactor outage and the retirement of many employees who had been

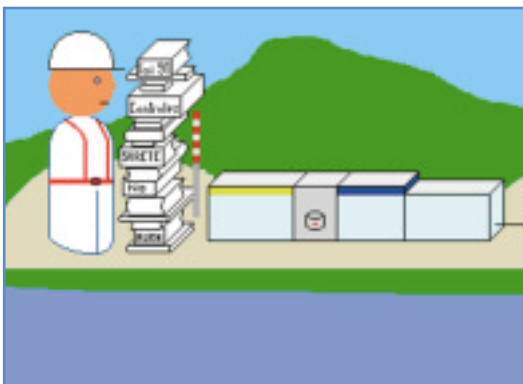
active since the start of the nuclear power plant operation. The number of people working essentially on safety increased, as did the number of engineers in the plant sections. The Quality Safety section was set up, then the Renovation project.

Another important change was the fact that the departments at Cadarache which previously obtained their information for their own research from Phénix were now being increasingly approached by Phénix to assist with its own information needs. In addition, the engineering staff at Novatome was increasingly requested to help prepare the reports, develop test and inspection means, design new systems and manufacture new equipment.

Above and beyond the fact that the Phénix reactor was now twenty years old, the grounds to the safety upgrading were myriad. Given the diversity and the complexity of the various issues at stake, it was decided to present them grouped into themes. The subjects included the reactor Lifetime Extension Project, the cracking mechanism found on the secondary circuits, analysis of the risk of major sodium fires and of the total loss of the normal decay heat removal circuits, and the new safety demands. At the same time, both the passage of the law dated 30 December 1991 on nuclear waste, then, later, the ministerial decision to definitively shut down Superphénix influenced the plant's objectives. During this period, the Phénix plant was authorised to carry out two irradiation cycles in special conditions.

5.1. The Lifetime Extension Project

Conducted in concert with the analyses of the negative reactivity trips, the thinking on the downstream side of the fuel cycle emphasised various characteristics of fast neutron reactors for plutonium management (the possibility of using the reactor as burner rather than as a



breeder reactor) and for transmutation of long-lived radioactive waste (cf. § 5.5). With respect to the Phénix plant, this motivated, end of 1993, the start up of a Lifetime Extension Project, whose objective was to define the steps which were indispensable to renovating the reactor in order to ensure safe operations for an additional ten years^[1]. This also meant using the operating time to add to knowledge on fast neutron reactor technology, and to attempt to provide a definitive answer to the origin of the A.U.R.N.

In fact, given the shutdown periods experienced since 1990, the reactor had accumulated 97,000 hours of power operations, whereas it had been designed for 140,000 hours (20 years of operation with an availability factor of 80 %). Yet the natural ageing of the structures had to be taken into account, and the stresses undergone also had to be accounted for.

Tangibly speaking, the **condition of the nuclear steam supply system** had to be evaluated, from the reactor block to the steam generators, and extended operations had to be justified, for the initial design studies had been based on a twenty-year life period. This work involved both the reconstitution of the context surrounding the design, manufacturing and operation of the materials, and detailed examination of the secondary circuits which, unlike the internal reactor structures, were easily accessible.

The consequences of high-temperature ageing (approximately 100,000 hours between 350 and 550°C, were studied for the various steels used in the nuclear steam supply systems. This involved studying rates of creep, embrittlement in operation, tensile strength, resilience, corrosion resistance and other phenomena on

stainless austenitic steels, ferritic and austeno-ferritic steels, and welding materials using chrome, nickel and molybdenum. These studies added considerably to the knowledge of these materials.

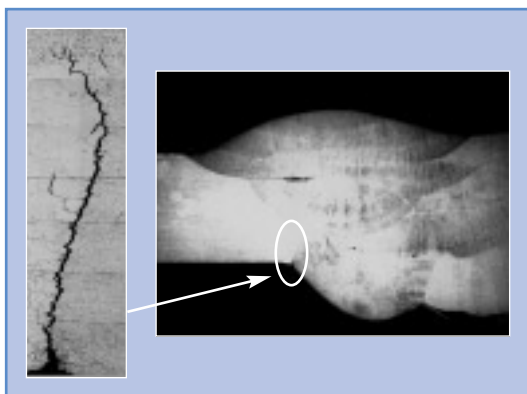
Recent developments in the **thermohydraulics** field were implemented, such as large-scale simulation of turbulence. The instabilities of flows in sensitive zones had to be calculated. A first application took place on the secondary circuits, to recreate the mixing of fluid streams at various temperatures which had led to cracks in a tee-piece in May 1986 (cf. § 4.3). The core outlet zone was also closely investigated in highly refined modelling to evaluate the effect of thermal stripping on the lower structures of the core cover plug. Then, on an even higher scale, modelling of the lower zone of the hot pool, with its highly complex geometry, evaluated the fluctuating aspect of the stratification on the primary vessel.

To confirm these computer simulations, a new thermometric device (a tube equipped



Secondary pump

[1] At the time, it was planned to operate the reactor up until 2004, the date chosen to have the time to summarise all the experiments conducted before the parliamentary debate scheduled for 2006 by the 30 December 1991 law.



Weld defect on 321 steel tube

with thermocouples along its entire length) was installed in the reactor in 1999. Similar modelling was done for all the hot and cold pool on the primary circuit, in order to evaluate the thermal asymmetry affecting the under-sodium reactor block structures, in normal operation with two secondary loops operating, and during an operating incident such as tripping of a secondary pump. A thermometric device was installed here too, near the sensitive zones of the core support structures, to validate the study.

In the field of **mechanical design**, a major experimental program focused on weld resistance, revealing how conservative the rules used for the ductile materials were and how unsuitable the simplified rules in the field of cyclic stresses on the singular welds were. The fluctuation zone of the sodium surface in the main reactor vessel was carefully analysed and confirmed by geometric measurements in situ. Special attention was paid to the problems of buckling and thermal stress ratcheting, mechanisms which could cause plastic deformation. Calculations reconstituted the damage to the main secondary piping, and the results were compared to the tests conducted on the zones which appeared to have cumulated the most damage.

After observing stress relief cracking on some of the **welds** in the 321 steel secondary circuits (cf. § 5.2), an experimental laboratory approach was used to develop and adjust a damage model of these joints. The case of creep was also examined. In addition to working on the current condition of the cracked welds, the evolution of hypothetical defects in various operation conditions included accidental condition such as a severe earthquake was also considered, in

The sheet metal workers

The sheet metal working unit is responsible for the maintenance of the pipes and associated elements (reservoirs, valves, welded valves and so on). This maintenance is based essentially, in terms of prevention, on checking the equipment under pressure. This check punctuates the outages. Certain elements, such as the valves responsible for evacuating excess pressure, are nevertheless tested and calibrated whilst operating. The inspection of equipment under pressure demands considerable preparation, as it also takes into account the logistics (scaffolding, opening and lighting the equipment). This unit is also responsible for monitoring the behaviour of the pipes when faced with variations in temperature. Readings are taken with tracers fixed to the tubes. They widen more or less from an initial point determined on a sealed plate nearby.

The corrective side of sheet metal work maintenance consists in repairing any pipes that have deteriorated. For example:

- replacement of a module in the steam generator,
- replacement of a steam tube on the start-up circuit,
- inspection and repair of a welded valve on the condenser's supply circuit.

The unit can also take charge of manufacturing piping elements to improve operating procedures. For example, installation of a neutral gas scanning system to preserve the pipes better during drainage.



view of the demands imposed by extended reactor operations. Such considerations were required both to verify structure resistance if an accident occurred at reactor end-of-life, and to determine the maximum size for small defects that could go undetected at no risk during inspections.

The early analyses, which worked with the assumption of elastic linear material behaviour and used formulas of existing stresses and intensity factors, were exceedingly conservative. According to the calculations, the structures were already in a state of ruin, which clearly wasn't the case. Complex codes were then developed, combining finite element and elastoplastic analysis which provided direct access to the magnitudes which characterised the harmfulness of the assumed defects.

One difficulty involved in applying these methods to the reactor block came from the need to model practically all the structures, due to the very large radii of the reactor block internal structures (several metres) and their very fine thickness (a few centimetres). These calculations demonstrated that extending reactor lifetime was acceptable, and gathered considerable feedback experience on the influence of certain manufacturing conditions, such as chemical and thermal treatments, forming and welding, on later damage to structures in operation.

5.2. Cracks on the secondary circuits

The experience gained from inspecting the secondary circuit welds showed that gamma radiography on 321 steel does not reliably provide information located at the root of the weld in the thermally affected zone. Between 1987 and 1989, the department of advanced techniques at Saclay developed, at



Inspecting secondary cooling circuit pipes

Phénix's request, ultrasound inspection of these welds. Pieces of the Phénix plant piping were used to test and validate the methods which showed themselves to be quite suitable, since, on 350 sample pieces, all the indications of depth over 500 μm were detected. These results were presented to and accepted by the Safety Authority (Bureau of Nuclear Control and Construction). However, this method did not allow for accurately determining the depth of the defects. Above all, no "zero point" was established during fabrication to use to compare the signals obtained on the welds in operation.

After the 1989 ten-yearly inspection, the operator took advantage of the many reactor shutdown periods to conduct non-destructive tests of the secondary circuits. The parts made of 304 steel were the first to be examined, for they were the biggest and some of them had been located in the reactor building. All the circular welds and the longitudinal welds in the elbows and straight sections were inspected. Very few small, non-evolving manufacturing defects were revealed, and some ten welds were repaired.

Next, the several dozen metres of main piping and equipment made of 321 steel were

examined. This stainless steel, to which titanium has been added to improve the material mechanical strength at high temperatures, was used to make some of the secondary lines around the steam generators, up to and including the buffer tanks. This steel, commonly used at high temperatures, has the widespread confidence of the specialists.

Yet the expert evaluations conducted on a tee-piece which leaked in May 1986 (cf. § 4.3) revealed a **delayed reheat crack** mechanism in the zones thermally affected during the welding of the tee to the steam generator inlet header. Early non-destructive tests (visual, dye penetrant test, gamma radiography, ultrasound) conducted after 1989 on the welds in the 321 steel secondary sodium circuits revealed defects in the roots of the welds, which were repaired when depth or length were deemed to be too great. Given the number of defect indications which were found, the inspections were gradually extended.

Related expert evaluations showed the generic nature of the delayed reheat crack in the thermally affected zones around the large diameter circumferential welds in the portions of the circuits in 321 steel, particularly in the complex geometry zones (triple point welds, weld connections with different thicknesses, ...). The higher the operating temperatures, the more cracks were found. Cracks were observed on 12% of the welds operating at 550 °C and on 3 % of the welds operating at 475 °C.

The mechanical resistance tests applied to weld samples showed that despite the defects observed, joint strength remained high after several tens of thousands of hours of operation at 550 °C. The radiographs taken at manufacturing for a reliable zero point were

routinely re-examined. The tests were often redone several times, due to improved test procedures and increased understanding of their limits, given the results of the metallographic expert evaluations which were conducted on the welds cut out for repair.

Studies were undertaken to justify maintaining the piping in good condition. However, it was hard to guarantee that the phenomena would not evolve and negatively impact plant safety for all the hypothetical situations which were taken into account. In the end, all the portions of the 321 steel circuits likely to have delayed reheat cracks according to the expert evaluations were replaced between 1995 and 1998 by new 316 steel elements (stainless steel with controlled nitrogen and low carbon content), for which the research done in the Superphénix reactor showed much better behaviour. An operating monitoring program was developed to examine certain welds every eighteen months of power operation. The welds were selected based on their heavy operating loads or on the presence of small defects whose lack of evolution had to be confirmed.

“There’s a problem...!”

Can you come and fix the (...)?” The people who get this type of phone call from the shift manager are generally on-call. This means that, outside their normal work hours, they must stay at home for a week, ready to come to the power plant to repair something within their field of knowledge. Sometimes, however, when it is a question of particularly sensitive materials, the skill and expertise of a particular technician is preferred. So he will be the one to get the call, whether he is on-call or not. Some complain a little, for the form, but then, regardless of what time of day or night it is, get in their cars and come to the power plant. There is a certain pride in being considered indispensable to the good running of the plant, and some such technicians even go further, keeping their professional “secrets” to themselves and only passing them on to the younger generation when they leave for retirement.



Other types of cracking were also found, and it quickly became clear that the welds done at the time of the replacement of certain piping sections were particularly prone to cracks. Indeed, these welds connected new material to material which had aged in operation. When these materials were assembled, the welds had been levelled at the root pass, that is, inside the piping, to eliminate any geometric irregularities, which generally can be the source of local stresses where defects begin.

In addition, in 1986, the buffer tank on the N. 2 secondary circuit had an argon leak in the presence of sodium aerosols at the point of the level measurement nozzle (cf. § 4.3). During an inspection in 1991, a through crack was detected on the upper convex

head. Each time, a zone corroded by the sodium-insulation mixture was found around the crack. More extensive inspections revealed the presence of defects on the tanks in the three circuits, caused by a delayed reheat crack mechanism.

These defects were first repaired by scouring and reloading the welds, then the three tanks were replaced by new 316 steel tanks, built to the new design rules, and equipped with reinforced leak detection (sampling under the insulation and analysis for the presence of sodium).

In 1992, two cracks were detected on a circumferential weld on the N. 2 secondary circuit, on a 304 steel section downstream from the hydrogen detection return. Expert evaluation showed that these cracks were due to thermal stripping associated with the temperature fluctuations of several tens of degrees of the fluid mixture. The sleeve was replaced on the three circuits after design changes to the nozzle.

No other potentially affected site was identified, except in the expansion tank, where thermal stripping was also found in 1993 on the spherical shell, where the hot sodium arrives through the balancing line from the buffer tank. The balancing line extends into the buffer tank to cool the sodium before it pours into the tank. It was realised that at the time the plant was built, the flow had been increased whereas the heat exchange piping had not been lengthened, which gave rise to this thermal stripping on the barrier, which was then replaced.

The isolation valves at the steam generator inlet (hot leg) were removed, for after discovering creep-related defects it was demonstrated that they were not essential to operations. Another crack-sensitive site was identified as the flange supporting the seal-



Replacing secondary cooling circuit pipes

ing bellows for the secondary circuit penetrations in the wall between the reactor building and the steam generator building. These flanges were inspected using ultrasound and dye penetrant tests and the indications found were scoured out. However, the flanges on secondary circuits N. 1 and 3 were eliminated in 1999 and were replaced by different design parts.

5.3. Major sodium fires and the loss of decay heat removal circuits

From the safety standpoint, a secondary sodium leak is acceptable. If a secondary sodium leak occurs, the piping is drained, the reactor is shut down, the non-radioactive sodium stops flowing, the sodium fire outside the insulation is extinguished and decay heat is removed through one or the other of the two secondary circuits.

Emphasis however is placed on preventing more serious accidents. For example if a crack developed in a weld, if it is not a through crack (no sodium leak, so no early detection which would enable the weld to be repaired), the crack could suddenly burst, causing a massive sodium spill and a major fire, especially if the sodium sprayed out in fine drops.

After the studies and work done at the Creys-Malville plant to guarantee safety in the event of a **major sodium fire** burning in spray form, a similar approach was required at the Phénix plant. The risk of a major sodium fire had to be taken into consideration, including a simultaneous reaction between the sodium from the leaking circuit and the oxygen from the surrounding air and water discharged by the circuits destroyed by the sodium fire. This reaction, sometimes called a sodium-water-air reaction, or sodium-



Sodium-fire extinguishing exercise

water reaction in open surroundings, is extremely powerful.

The results of the tests on the 321 steel pipe welds were interpreted as being due to poor manufacturing quality of the secondary circuits^[2]. The solution chosen to counter the risk of the preceding scenario called for building a wall with a two-fold purpose. One, to separate the zones in the steam generator building containing sodium circuits from the zones where the water and steam piping circulated; two, to reduce the amount of oxygen available to feed a sodium fire around the pipes.

To improve protection from major sodium spray fires in the reactor building, the tunnels for each one of the three secondary circuits were contained by steel flooring and insulation on the walls. Steel chutes extended the tunnels to the intermediate heat exchanger heads. An anti-splash shell and a retention shell were installed around each intermediate heat exchanger to reduce the risk of sodium pouring onto the reactor slab. Steel protecting covers were placed around the expansion bellows on the penetrations in

[2] It must be recalled that in the original design, since the secondary sodium circuits were not radioactive, they were considered as "classic" equipment, to which industrial state-of-the-art construction and inspection rules were applied, and not the far more stringent rules which were applied to the nuclear parts of the reactor.



the wall separating the reactor building from the steam generator building.

In addition, beaded wires were added to all the circumferential welds (to detect the presence of sodium outside the weld) on the secondary circuit main piping for closer, earlier surveillance of a potential sodium leak. Beaded wires were also added to the entire length of the auxiliary piping which had not previously been equipped. In addition to the fire detection systems already in place, new air sampling installations were put in to sample the air in the loft atmosphere and in each one of the secondary galleries and the steam generator building, in order to detect any sodium aerosols.

Insulation was added to the building frame

supporting the sodium piping to protect the frame from the high temperatures resulting from a major sodium fire. A separation was built between the sodium zone and the water-steam zone in the underground floors of the steam generator building, to guarantee the draining of the sodium from the secondary circuits into the storage tanks, and the steam generators drying out at the very outset of a major sodium fire. The steam generators' casings were reinforced. Three water tanks, first installed in the steam generator building -upper gallery, were replaced by new tanks in the turbine hall.

However none of these protections reduced the risk of cracks on the secondary circuits. If, in addition, it is assumed that defects exist

A shift team

A shift team is composed of:

- a shift manager, who is the authority for all the employees in his team,
- an assistant or multifunction operator, capable of dealing with all aspects of the control room,
- a reactor operator, who guides the operations of the reactor and all its auxiliaries,
- a machine operator, who deals with the electricity generating facility, from the steam generators to the 225 kV substation,
- a panels operator, who is responsible for the operating manoeuvres from a panel in the control room. He is also in charge of blocking [1],
- several field operators who monitor the facility, perform localised readings of physical values, guarantee that the materials are functioning correctly and perform manoeuvres at the request of the operators.

Each of the six teams does an eight-hour shift, operating the power plant. The shift starts with the changeover, that is, an interview between an employee and the person working at the same workstation in the previous shift. Then there is a short briefing during which the shift manager summarises the changeover and gives his orders for

the coming hours. Each employee at his workstation thus plays a role in operating the power station correctly, whilst remaining ready to act should an abnormal event occur (shutdown of an engine, a fire alarm, a sodium leak, etc.).

The operators remain in the control room; they collect information from the screens, and compare it with that collected during the changeover and briefing and then report it to the field operators. The field operators then spend several hours in the different parts of the facility (controlled area, steam generator building, turbine hall etc.) where they collect information before transmitting it to the operators, either immediately by telephone or at the end of their round.

Quiet periods are used to prepare future operations, write up documents, write reports and so on. A meal is taken in a room next to the control room and plays an important role in community living. Finally, the next shift team arrives and it is time to go home, around six o'clock in the morning, two in the afternoon or nine at night...until the next shift, either the next day or after a few days' rest.

[1] Blocking consists particularly in making one or more pieces of equipment non functional and guaranteeing the absence of any dangerous fluids (electricity, steam, pressurised water, sodium etc.), so that other people can intervene on the equipment safely. After critical analysis of the request, the blocking perimeter is defined and then locks are placed on the equipment (valves, circuit-breakers...) to ensure that they will not be manipulated incorrectly.



on each one of the secondary circuits, an event – the only identified event would be a major earthquake – could simultaneously break the residual filaments on the biggest welds (this is referred to as the critical failure length – beyond which earthquake stresses could break the piping) and all the operating secondary circuits would immediately burst. Not only would this cause a sodium fire in the steam generator building, but it would also be impossible to remove the decay heat from the reactor through the secondary circuits.

This hypothetical accident is referred to as a **total loss of decay heat removal circuit** (D.C.N.E.P. in French), a conservative hypothesis of the sudden and simultaneous loss of the three independent secondary sodium circuits. The reactor decay heat is then primarily removed via convection through the reactor vessels to the emergency cooling system^[3]. The temperature of the reactor block rises until the residual power removed by radiation compensates the residual power decaying with time. This temperature must remain below 720 °C to ensure structural integrity of the reactor. Although it is acceptable, this damage situation must be avoided at all costs.

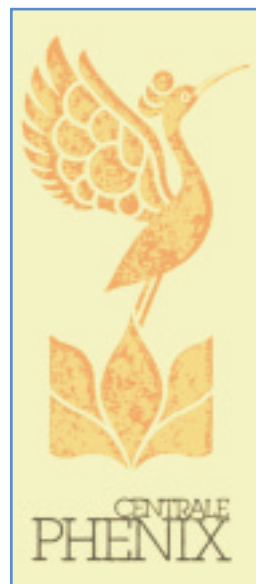
Going back in time, the Safety Authority had conducted the first reactor safety upgrading evaluation in 1986. This evaluation primarily consisted of re-examining the final safety report with respect to the current safety criteria, applying the methods developed in the previous years, in the framework of the safety analysis for the start-up of the EDF series of pressurised water reactors, and the Creys-Malville plant. This evaluation resulted in several requests to upgrade some studies, which were gradually undertaken.

[3] Plans called for reinforcing the safety of this circuit during the renovation shutdown between 1999 and 2002 (cf. § 6.2).

In 1989, calculations showed that the initial studies had minimised the effect of the neptunium β and γ radioactivity after reactor shutdown, and that the residual power to be taken into account in the D.C.N.E.P. studies was greater than that used heretofore. After criticism from the Safety Authority about the time it was taking to respond to its demand, the operator proposed to limit reactor operation to 500 MWt, by virtue of the precautionary principle, for the time it would take to finish the research underway. Indeed, at that level of initial power, the plant systems could remove the residual power as it was calculated, without the reactor block reaching the absolute limit of 720°C.

These were the conditions in which the 48th irradiation cycle was authorised in April 1990. During the shutdown period following the fourth A.U.R.N. (cf. § 4.5), tests were conducted to increase the heat removal capacity of the emergency cooling circuit, which included replacing the nitrogen in the innerspace between the main vessel and the containment vessel with helium, a better heat conductor. Use of the CREX experimental circuit was discussed (cf. § 4.4), but the transformations required for its use as a main safety system were too complex.

Concurrently, in order to obtain rapid authorisation to operate, a report justifying the acceptability of a D.C.N.E.P. accident was prepared for operations at 350 MWt (in



which case, reactor block temperature, in every scenario, remains below 670 °C). This file was validated by the Safety Authority in June 1993. Studies started up to raise the limit to 470 MWt, then 420 MWt, operating with two secondary circuits. These studies were however abandoned for the uncertainty of their results was incompatible with the physicists' requirements for firm knowledge on the reactor operating power to define the irradiation experiments. Thus, reactor power remained definitively limited to 350 MWt.

5.4. New safety requirements

In 1986, the Safety Authority had also asked the operator to re-evaluate the continued operation of the vital safety functions, in light of the current methods, in the event of the maximum historically feasible **earthquake** of VII – VIII intensity on the MSK scale. This level of seismic activity, one-half degree higher than the original design, corresponded to the recent redefining of regional seismicity. It covers an earthquake close to the one which occurred in 1873 (placing the epicentre directly -under the reactor), and to the 1909 earthquake which occurred in Lambesc en Provence (placing the epicentre 35 kilometres from the site, on the edge of the sismotectonic zone).

Earthquake resistance was not too difficult to demonstrate for the equipment, however it was quite complex for the plant buildings, which were designed and built according to the aseismic rules in effect in the late 1960's. This work, put to the side during the intense periods working on the A.U.R.N., was then recovered and provided with significant resources including in-depth studies subcontracted to Novatome, who had experience on the research conducted for the Creys-Malville plant.

As was shown, the facility required major work in order to guarantee the resistance of some of the building structures, with sufficient safety margins to compensate for the uncertainties on some of the construction techniques and on the evolution over time of the materials used in the frames and the traditional bolting. Earthquake-resistant renovation work began in 1996. The clearance space between the reactor building roof and the handling building roof was expanded. The annex building was strengthened. The earthquake resistance of the control room and offices building was completed to ensure the protection of the operating teams called in to maintain reactor safety after an earthquake, and to enable the staff to reach the remote operation panel, if need be. The thresholds for the automatic shutdown of the reactor in the event of earthquake were drastically lowered, dropping from a speed of 1.5 cm/s to 0.15 cm/s, which corresponds to an earthquake of III - IV intensity on the M.S.K. scale (which is an earthquake with 1000 times less energy than the energy level for which the plant is designed).

The emergency cooling circuit was also modified in order to guarantee operation after a major earthquake. Protection was added to the piping travelling through the reactor building infrastructures and the annex building. To lower the risk of failure, two standby generator sets were installed to feed the pump motors in the event of loss of the EDF electrical supply and the two main generator sets. Likewise, "firemen" connections were placed on the circuits, to be able to supply them with water from a mobile motor-driven auxiliary feed pump.

The Safety Authority completed a study on the **reliability of the automatic shutdown system** of the reactor, then, in March 1991,



recommended the installation of an articulated control rod like the ones on the complementary shutdown system (SAC) at the Creys-Malville plant. Such a step would markedly reduce the probability, which was already quite low, of a core meltdown. In January 1994, the decision was made to install such a system at Phénix.

In order to take advantage of the results of the design calculations, the tests and the return on experience, the design of the Phénix SAC rod assembly and mechanism was similar to Superphénix, but different from the six control rods. The SAC rod consisted of several articulated parts, one of which contained the boron carbide pellets. The absorber was inserted in the guide tube head at all times, which meant it could drop even if the core was distorted or off centre with respect to the core cover plug. The rod

mechanism had a spherically seated bearing providing for the centring of its supporting electromagnet with respect to the guide tube head. The complementary shutdown system could shut the reactor down all by itself.

To save time, available elements were put to use, including the boron carbide pellets, the control rod absorber sub-assembly guide tube and the spike of a breeder sub-assembly. Qualification benefited from the tests applied to the Superphénix system, and was also subjected to hydraulic tests at Cadarache and neutron tests in the reactor. The system was installed in the core centre in December 1996.

Further requirements emerged, this time concerning **maintaining staff capability**. The employee population at the Phénix plant was extremely stable, with very few requests for transfer to other sites, even among the EDF personnel accustomed to greater mobility. One consequence of this stability was the retirement rate of approximately ten retirements per year, primarily among the highly experienced engineers and technicians who had carried out the major part of their professional career at the Phénix plant. This vital transfer of expertise between outgoing and incoming specialists was organised for positions identified as strategic, through periods of overlapping presence adapted for sufficient training and coaching to take place.

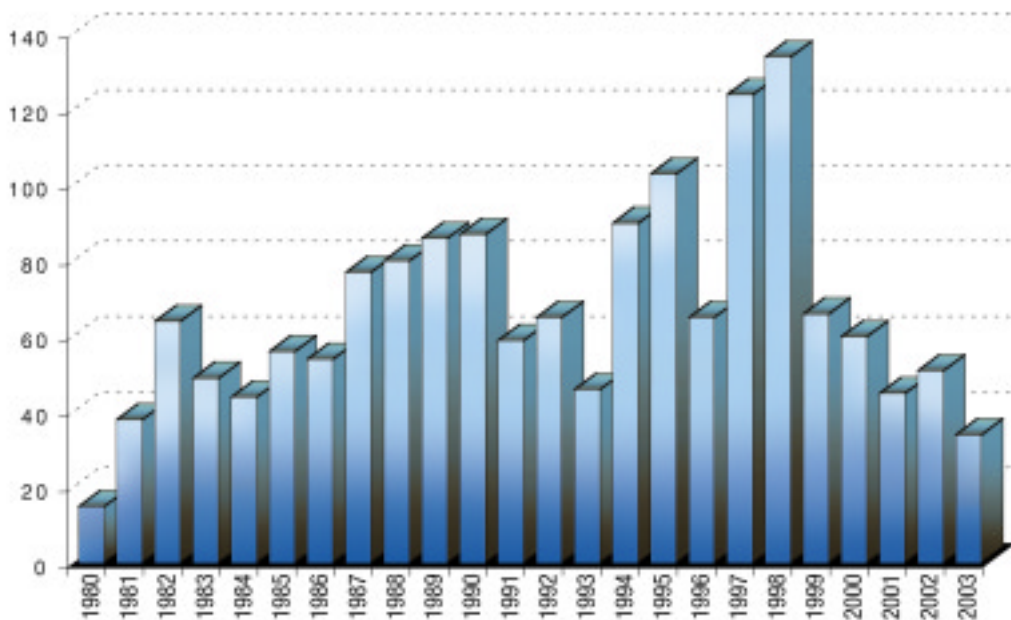
In compliance with procedures established in the EDF reactor system following the Three Mile Island accident, "beyond design" and "emergency" methods had been developed. These described the procedures to follow in the very unlikely event of accident which could lead to a deteriorated situation for reactor safety. Such cases include the total loss of pumping from the Rhône river, of



Complementary shutdown system



Safety files



Permanent Group meetings

Date	Main subjects discussed
26, 27 November and 10 December 1975	.. Launching of reactor operations
29 April and 26 June 1986	.. Safety re-assessment
12 September 1991	.. Negative reactivity trips
24 June 1993	.. Negative reactivity trips, sodium fire protection and safety re-assessment
1 st and 22 February 1996	.. As above + strength of internal structures
3 July 1997	.. Conditions under which reactor operations can be resumed
4 and 11 December 1997	.. Facility seismic resistance, core support and steam generators
12 July and 6 September 2001	.. Steam generator repairs, check of core support and core cover plug
31 October 2002	.. Review of improvements made to the plant



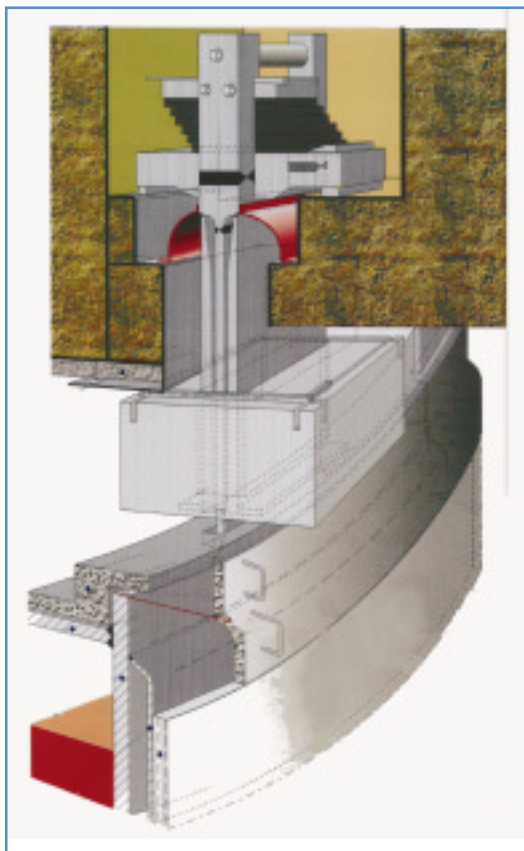
internal and external electrical supply, of nitrogen or argon distribution, of the storage drum cooling circuit ... Generally speaking, these procedures incorporated and classified elements which have been part of operating manuals since start-up. Other procedures dealt with maximal accidents such as sub-assembly meltdown, leaks in the main reactor vessel and its safety vessel. They are designed to limit the consequences of such extreme events.

The expert appraisals which were underway to find the causes of the A.U.R.N. led to questioning about **possible damage to the reactor block**. In connection with the Lifetime Extension Project, structural damage was re-evaluated based on precise accounting of all the situations the reactor had been subjected to. By situations was meant the cumulative length of operating time at the various operating regimes and transients (start-ups, normal shutdowns, automatic shutdowns, ...). This work focused on the reactor block, its auxiliary circuits, the secondary circuits up to the steam generators, and some fuel handling equipment.

The approach consisted in improving knowledge about the technical lifetime of the installation components, by a double approach to both materials and potential damaging mechanisms. Then the sensitive sites were identified, based on the consequences of their failure on the installation safety and also on the cost of repair or replacement. This work led to implementation of actions to prevent damage (for example, restricting operating temperature), to improve monitoring (for earlier detection of defaults), and to increase the effectiveness of reactor operations in accidental situations and limit their consequences (procedures improvement).

This was used to verify the resistance of each sensitive piece of equipment, ensuring that it can sustain a worst-case accidental load (earthquake, violent sodium – water reaction, ...) without irreversible damage, at the extreme limit of its forecasted operating life, during which it has already experienced the largest probably number of normal operating transients (start-ups, automatic shutdowns, ...).

Tests were defined for the most sensitive equipment emerging from this analysis. This was the framework for the inspection on the reactor vessel hangers, and other, more complex inspections were being prepared for the conical shell and the core cover plug (cf. § 6.1).



Upper hanger of main vessel



The CEA Nuclear Reactor Division which governs the Phénix plant, set up a working group which wrote a 1995 [report on the safety status](#) of the facility and on the actions underway to improve the safety level in the framework of maintaining reactor operations for another ten years. This provided overall consistency and emphasised the consequences of high-temperature operations: fatigue-related damage – creep, difficult to model, and accelerated aging of the materials, in particular the titanium-stabilised steels (321 steel in the secondary circuits). Accordingly, the decision was made to reduce the range of reactor operations by lowering the average hot pool temperature from 560° to 530° C, which brought about a drop in the secondary circuit temperatures.

The archives

The archives are not in either an attic or a basement, and they are not piles of papers that risk collapsing at the slightest breath of wind. The “archives” are the power plant’s documentation, responsible for guaranteeing the permanence and communication of all the documents necessary for operating the power plant, including in particular the plans and electricity diagrams. In other words, all the documents produced internally, as well as those drawn up by the constructors and suppliers. There are even occasions when sets of documents are retrieved or copied at a constructor’s to eliminate the risk of losing the corresponding documentation in the course of a restructuring.

Since the power plant was built, almost 120,000 documents have been added to the archives. All these documents are saved and stored in duplicate for security reasons (in case of loss or fire damage, for example – one copy is stored on another site belonging to the CEA). Each document can be consulted on site or photocopied for later, more detailed study. Some documents have never been requested since their arrival in the archives, others are photocopied regularly. Sometimes, an in-depth search for “the” document required is necessary, perhaps with only a vague idea of its content or origin, along the 470 meters of documents (more than 23 tons of paper).

5.5. The 30 December 1991 law

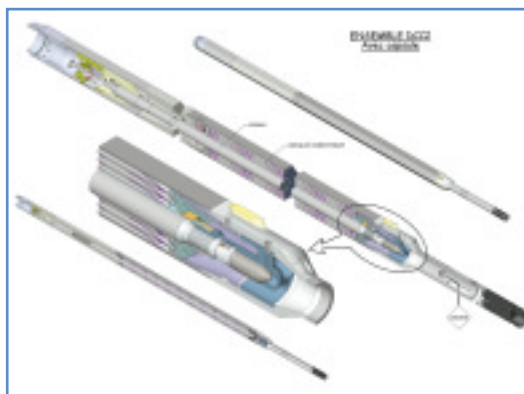
Up until the end of the 1980’s, the two main experimental programs conducted at the Phénix plant were the qualification of fuels with increasingly high specific burn-up, to improve the cost of future fast neutron reactors, and tests on other core configurations with the triple objective of stability, extended campaigns between refuellings, and breeding optimisation (heterogeneous cores). These two programs were conducted for the European Fast Neutron Reactor program.

Two [new programs](#) came into being in 1990. The first, named CAPRA, which stood for increased plutonium consumption in advanced reactors, sought to qualify and quantify the specific reactor contribution to general plutonium management – inventory control, multi-recycling, isotopic vector regulation. The second, originally called SPIN, dealt with the separation and incineration of radioactive products. The part which dealt with the feasibility of transmutation of minor actinides and long-lived fission products rapidly became the number 1 axis of the 30 December 1991 law, still referred to at this time as the “Bataille law” named after the reporter to the National Assembly.

This law stipulated that long-lived radioactive waste had to be scientifically studied in order to be able to come to a decision, within fifteen years’ time, on their long-term management. The law specified that three lines had to be explored during this fifteen-year time period – that is, by 2006 – in order to propose a range of management choices to the political decision-makers. The three directions defined by the law were:



- The search for solutions providing for the separation and transmutation of the long-lived radioactive elements which are present in waste (axis 1),
- The study of the possibilities of reversible storage^[4] in deep geological repositories, using underground laboratories (axis 2),
- The study of conditioning and long-term interim surface and sub-surface storage^[4] of waste (axis 3).



Irradiation rig

The law assigned the CEA with the responsibility for piloting the research on axes 1 and 3. Axis 1 was devoted to reducing the toxicity of the waste, with the underlying idea of extracting the long-lived radionuclides to transmute them into non-radioactive or short-lived nucleides. This was broken down into two major research fields: advanced separation (as compared to the current treatment) of the long-lived radioelements (minor actinides^[5] and fission or long-life activation products^[6]) and transmutation. The law also set the framework and a national objective for the research conducted by the CEA in an area challenged by those who call for an end to nuclear energy.

The two programs, plutonium burning (CAPRA) and incineration (the 1991 law), entailed the development and qualification of entirely new fuels (mixed oxides with containing high plutonium content or americium) under high representative flux. The Phénix plant was the most suitable facility for the CEA, and beyond them, for western Europe as a whole, to conduct such research. The few other replacement solutions (Russian and Japanese fast neutron

reactors, thermal power plants, non-reactor research...) did not meet the needs identified in these programs. Furthermore, the Superphénix reactor also served as a unique resource in these fields, though on the scale of the industrial validation of solutions which must first be qualified in a smaller, more flexible reactor like the Phénix plant.

Within the CAPRA programme framework, studies were conducted to make the reactor a burner, eliminating all or part of the breeder sub-assemblies, and even the upper axial blankets. The breeder sub-assemblies were replaced by steel sub-assemblies, those farthest from the core contained absorbing materials, retaining their roles as neutron reflectors and protections. This operation has a certain economic effect, for it avoids production, then reprocessing or storage of the breeder sub-assemblies (the recycled plutonium is no longer a usable material). The steel pins could also replace the upper axial blankets in the fuel sub-assemblies, as they themselves were replaced. Following a few adjustments, the incidences on the reactor core were acceptable. These studies were however abandoned when the CEA made the decision to reduce the operating life of the Phénix plant (cf. § 5.7), in order to use up the stock of available breeder sub-assemblies.

[4] Additions made to the law dated 30 December 1991 by a government decision in December 1999.

[5] The main so-called minor actinides, as opposed to the actinides used as fuel (uranium and plutonium) are neptunium, americium and curium.

[6] This essential applies to iodine 129, technetium 99 and caesium 135.

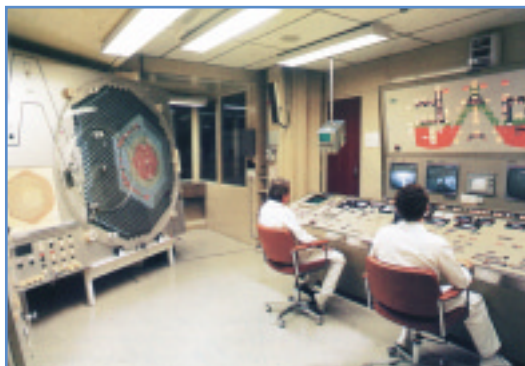


Results on the subject of burning had already been gathered, particularly through the Superfact experiment, which was done in the Phénix reactor between 1986 and 1988 (cf. § 3.4). The program was continued with irradiation during the 49th cycle (December 1994 – April 1995) of two pins containing fuel pellets enriched with 45 % plutonium (Caprix) and 16 pins containing inert matrices (Matina 1) useable for the burning of minor actinides or the consumption of plutonium which has not been mixed with the uranium. The first were left in the reactors to pursue their irradiation throughout the following cycles. The second ones were removed from the reactor to be examined, then fourteen pins were put back in the reactor for the 50th cycle, under the name of Matina 1A.

The chemists

The Chemistry – Environment group in the Phénix plant focuses its activities on three fields: “classic” fluids, gas and sodium, and finally the environment. It is thus necessary to monitor the chemical quality of all the fluids used regularly: water, oil, gas-oil (for the generator sets with diesel engines), cooling fluids, liquid argon and nitrogen, and so on. The chemists also operate the demineralized water generation facility needed to the electricity generation facility. They manage the power plant's chemical products and monitor the chemical conformity of waste emissions to regulations.

The good quality of the sodium and cover gases (argon and nitrogen) is monitored using specific equipment: plugging indicators and Tasténa samples for the sodium, and chromatographs for the gases. To decontaminate components, this group is responsible for preparing the appropriate chemical solutions. Finally, the Chemistry – Environment group is also in charge of monitoring the impact of the power plant, by means of readings of gaseous waste emissions, follow-up of the radiation monitors situated around the site, samples of plants and agricultural products (such as milk) and keeping the registers demanded by the Safety Authority up to date.



Handling control room

During this time, the **Creys-Malville plant** carried out extensive renovation work between late 1992 and the first half of 1994, to bring the facility into conformity with the new safety requirements on sodium spray fires. A public enquiry was held in spring 1993, and was in favour of renewing the operating license. The safety files were processed by January 1994, with the publication of the Nuclear Installations Safety Direction, concluding that the start-up of the Superphénix plant could be authorised. On 12 July 1994, a new decree which “authorised the creation” of the Creys-Malville plant was published. The reactor diverged on 4 August, then very gradually built power up, within the framework of a substantial test program.

In 1994, the CEA and EDF defined a **programme of knowledge acquisition** (P.A.C.) that included three aspects :

- Demonstrate the fast neutron reactor's capacity to produce electricity and contribute to plutonium management and the reduction in volume and radiotoxicity of long-lived radioactive waste,
- Study the flexibility of fast neutron reactors using plutonium and qualify the technical solutions developed within the framework of research programs which aim to operate this type of reactor as a net plutonium burner,

- Study the possibilities for the destruction of long-lived radioactive waste, in particular minor actinides, within the scope of axis 1 of the law dated 30 December 1991.

This program resulted in the construction of three experimental sub-assemblies for irradiation in the Superphénix core: two had high (31 %) plutonium content (Capra), and one contained 2 % neptunium (Nacre). These sub-assemblies were delivered to the Creys-Malville plant in late 1996, just as the reactor was preparing for a maintenance shutdown after a nearly flawless year with 95% availability and 3.5 billion kWh produced. The experimental sub-assemblies meant to be placed in the core during the scheduled maintenance shutdown.

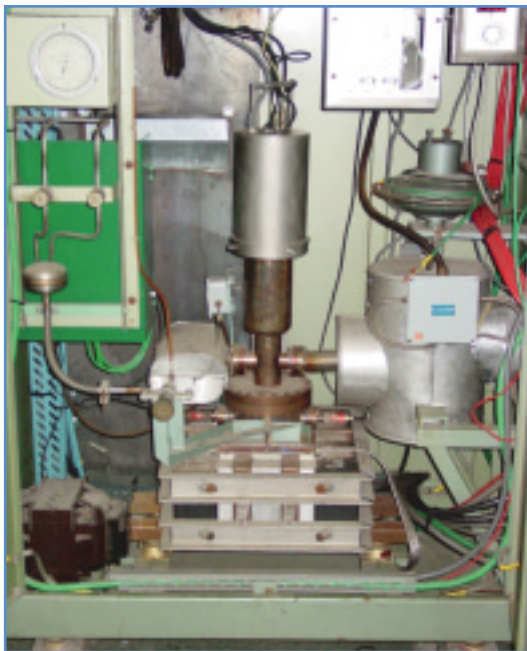
5.6. The 49th irradiation cycle

During this period, dominated by production shutdowns, each irradiation cycle became an event in and of itself, and thus is covered here in more detailed description than were the 48 preceding cycles !

The 49th irradiation cycle was interrupted after 5 days' time, following the 4th negative reactivity trip, in September 1990. Then the reactor was often kept in a state which was ready to start up on short notice. However, the operator found it problematical to make the point that the reactor had to operate if knowledge was to be gained on the cause of the A.U.R.N., even if the trips had not yet been fully understood. Temperature variations were carried out to obtain measurements on reactor block emissivity in the framework of the re-investigation into the loss of the normal heat decay removal circuit (cf. § 5.3). The primary spare pump was placed in the reactor after being supplied with

a gas injection device to introduce gas in the volute, to test this hypothesis of the origin of A.U.R.N. (cf. § 4.5). The last irradiated fuel sub-assembly available in the storage drum was dismantled in March 1991 (approximately 1500 fissile, breeder and steel sub-assemblies had been dismantled since the origin)^[7].

As of October 1991, the reactor frequently diverged to conduct tests at "zero" power (which in fact was power below 20 MWt). These tests included neutron measurements to specify the characteristic parameters of the core, automatic trips to compare with the A.U.R.N. occurrences, gas injections in the reactor to measure the influence of the passage of a gas volume in the core. The inspection and repair work on the secondary circuits took increasing amounts of time each month, and the scheduled date for the availability of these circuits, to bring the reactor back up to



Hydrogen detection

[7] Only forty or so breeder sub-assemblies and a few experimental sub-assemblies were dismantled in the following years, lacking an interim storage solution for the pins and the structural waste.



power operations, was regularly postponed. The N.1 and N.3 secondary circuits were finally ready in September 1992, and the electric generation facility was ready in October, while the reactor was maintained critical^[8]. However, new demands for the installation of sodium leak detectors on the secondary circuit pipes required them to be drained.

The plant was again ready in early 1993. The 49th cycle could then get underway, during ten days of **tests at 350 MWt** conducted in February, still within the framework of continued analysis of the negative reactivity trips. These tests first consisted of monitoring the start-up transient until the thermal and neutron equilibrium of the core was reached, accompanied by monitoring structure displacement. The tests were used to qualify the SONAR device in rated situation and to conduct additional neutron measurements. They provided for gathering a great deal of information for the continued investigations, yet did not contribute any decisive elements toward the identification of the A.U.R.N.

In the fourth quarter 1993, the Simfonix operations simulator started being used to improve personnel training and fulfilled the need for simulating real operating situations.

On 17 June 1994, a fatal accident occurred at the N. 3 secondary sodium circuit expansion tank worksite. A technician working for a subcontractor firm was found lying inanimate at the bottom of the glove box used for the work under argon atmosphere on this tank. The Gendarmerie and the Labour Inspectorate conducted investigations. The worker was supposed to work outside of the glove box, and knew the risks and procedures. No confirmed explanation has yet been able to elucidate the circumstances of death.

[8] 27 divergences took place in 1992, which maintained staff competence in reactor operations.

In addition to all the **work** described above, the five original heat exchangers (exchanger C was dismantled in 1985, cf. § 3.6) were replaced in 1994 by recent designs, after having been in power operation for approximately 80,000 hours. Secondary circuit N. 1, which was not intended for use in the upcoming two cycles, was equipped with “dummy exchangers”. Secondary circuits N. 2 and 3 received the new, 2nd generation heat exchangers H, I and J, which were equipped with a mixer at the outlet of the tube bundles. Their primary sodium flow was limited on the periphery^[9]. Exchangers A, B, D, E and F were dismantled in 1997.



Pumping station, external view

The 49th cycle resumed on 24 December 1994, after completing the required works, notably on the secondary circuits, and after the Safety Authority had processed all the files and reports. The reactor was still operating at 350 MWt with two secondary circuits, and there were no noteworthy problems until 7 April 1995, at which time the reactor had shut down normally at the end of the cycle. In three and a half months, the plant had operated at 61 equivalent full power days (EFPD), and produced 314 million kWh, with 86% availability during this part of the cycle. And then, the Phénix plant

[9] The second generation intermediate heat exchanger G, had equipped the N.3 secondary circuit since 1980.

And what if it were a negative reactivity trip?

On May 24, 1995, at 8 h 31 am, during a divergence, an automatic shutdown occurred. This shutdown remained unexplained for several very long days. No one at the time could understand what had happened. The initial analyses of the recordings and other log books gave no clues. The teams from the Physics department were called to study the recordings. All that could be observed and described was a certain phenomenon: there had been a sudden variation in the counting rate of the intermediate range neutron measuring channel, which had gone from around 2 counts per second to more than 3,000 counts per second in less than 500 milliseconds, before going back down approximately one second later.

Everyone tried to find the solution. The different recordings were analysed and correlated with each other. There were coherent and contradictory elements. A certain number of trends could be identified, but nothing more specific. The initial conclusions, as something has to be produced quite quickly, were that the phenomenon could have a real neutron origin. The threat of A.U.R.N.s was on everyone's mind. As for the team responsible for the maintenance of the neutron measuring channels, "their" material had been singled out. Had it been in good working order? Had they reacted correctly? Had it been used correctly? Had it been maintained properly? Had it been correctly adjusted? Many tests and demonstrations were needed before all these questions could be answered with a definite "yes".

All the teams were mobilised to try and find more clues to the mystery, but the dossier remained as empty as ever. Certain measuring channels reacted (the pulse channels), and others did not (the current channels). The teams from the Physics and Maintenance departments then launched what was almost a police investigation. What was each person doing at the time of the automatic shutdown? Interesting information turned up and another line of inquiry began to take shape. Two welding sites in the reactor building had been in progress on the day of the event. Welding machines have equipment that emits high frequencies. The shutdown took place at a time when many worksites were starting up. These elements were sufficiently disturbing for the decision to conduct a simulation test to be made.

The test was conducted on June 2, 1995. A measuring channel similar to the intermediate range neutron measuring channels was set up in the reactor building. It was connected to a fast recorder. A source of neutrons was installed so that the channel would be at an initial counting level identical to that of the intermediate range neutron measuring channels at the time of the automatic shutdown. A welding machine was installed nearby. The welder was then asked to start using the machine as he had done at the time of the incident. The results were not long in coming: the automatic shutdown was observed in the control room and the signals recorded were very similar to those of May 24. An explanation could at last be given and the case closed by forbidding all authorisations for welding work during the reactor's future periods of operation.

once again entered an extended phase of studies and work.

In 1997, a computerised system replaced the work orders which had been in use since the plant had opened, greatly facilitating the mass of work being done on the equipment, particularly with respect to safety. This computerised maintenance management system (G.I.E.) was the same as that installed in the EDF thermal power plants, and the same prescriptions applied to the personnel. The system also prepared the locking out of the equipment^[10]. During this same period, the warehouse where the essential spare parts

were stored was thoroughly renovated. Warehouse management was outsourced and storage quality significantly improved to comply with the new prescriptions applying to safety-related equipment.

5.7. Abandoning Superphénix

On 19 June 1997, following the victory of the Socialist party in the legislature elections, the new Prime Minister announced his decision to abandon Superphénix, as had been stipulated in the Socialist and ecologist government platform. As of the very next day, CEA general management announced that they would examine the measures required to successfully

[10] This covered the disconnection of electric power, neutralisation of hazards from mechanical power, etc., which ensured worker safety when intervening on equipment





conduct the research required to meet the provisions of the 30 December 1991 law, without Superphénix.

As a result, the experimental program at Phénix was increased and new experiments were defined in the framework of axis 1 in the 30 December 1991 law (cf. § 7.5). However, the industrial validation planned on the scale of the Creys-Malville plant was quite simply abandoned. Likewise, the experiments on plutonium burning were not resumed. The Caprix rig was the only one to remain in the reactor and pursue irradiation. The days and months which followed were marked by many protests by the staff from the Creys-Malville plant, demonstrating in support of their facility and of their faith in the future of their reactor. Despite their deep empathy, the agents at the Phénix plant did not take any collective actions to express their solidarity.

This was the context in which the meeting of the [Permanent Group](#)^[11] was held in July 1997. The Permanent Group was to rule on the safety of the Phénix plant, and discussions were rife about the impact of the Prime Minister's decision on Phénix. Definitely shut down, or on the contrary, a fresh start? In addition, the Institute for Nuclear

[11] To successfully conduct safety analyses, the Safety Authority works with Permanent Groups of experts who meet to examine important files.

Protection and Safety, which delivered its analysis to the Permanent Group experts, had not been convinced by the documents presented by the operator, in particular on the quality control of the core support structures inside the reactor block. Was it worth pursuing the analysis in these circumstances?

The meeting nevertheless took place. The operator stated that he had begun work on a new means of inspection for the conical shell, the structure that connects the core support plate which bears the core weight to the main vessel. If there was a crack measuring a few metres long in the connection welds, a strong earthquake could ram it to the breaking point, sending the core sliding to the bottom of the reactor, with the risk that the control rods lack the time to completely drop into it. In addition to an indirect geometrical measurement of the absence of movement, via the lateral neutron shielding assemblies. The operator proposed an ultrasound inspection through the shell itself. Such a method required substantial developments whose feasibility results were expected at the end of the year. Despite the doubt expressed by some experts in the Permanent Group, the appointment was set.

Two new meetings were scheduled in December 1997. In the meantime, the CEA announced that it would definitively shut down the Phénix reactor in 2004 at the latest. Given the required shutdowns for refuelling and maintenance, for reinforcement and renovation work, this sharply restricted the actual operational time remaining. The first meeting of the Permanent Group was devoted to the file on the facility's earthquake performance. The second meeting focused on the method for the ultrasound inspection of the conical shell and on the risk of leak in the steam generator modules. The Institute for Nuclear Protection and Safety



The Creys-Malville plant

was highly critical on these two points, and considered that the operator's approach was not sufficiently compelling, lacking an R&D program which would require several years of metallurgical experiments.

After the discussions between the Permanent Group experts, it was found that, subject to certain confirmations to be provided prior to the work, the reinforcements planned for the plant buildings were acceptable to ensure installation safety in the event of earthquake. Operations at the Phénix plant could thus be authorised up until its definitive shutdown, subject to the performance of the earthquake reinforcement work and the inspection of the conical shell welds according to the method described by the operator. This work was to be done during the scheduled shutdown between the 50th and 51st irradiation cycles.

The definitive shutdown of the Creys-Malville facility was officially announced in 1998, at the same time as the Government confirmed the reorientation of the Phénix reactor for research work on transmutation. This decision was not followed by a transfer of EDF personnel experienced in the field of fast neutron reactor operations, with few

exceptions (of which the present author is one). Indeed, EDF redeployed its personnel primarily to other electricity-producing nuclear power plants equipped with pressurised water reactors, which were crucial to its electricity production. However, the Phénix plant support group (GRAPH), which was temporarily created to help the operator manage the shutdown works for renovation, reinforcements, inspections and maintenance (cf. § 6.7), was widely made up, on the EDF side, of agents who had worked on Superphénix.

Losing neutrons and information

For a certain time, the condenser cleaning system using "Tapproge" balls had been causing a few problems. There were in fact increasing numbers of these blue balls, which normally circulate in a closed circuit, escaping into the Rhône. Some of the fishermen who saw them interpreted this as a leak of neutrons from the Phénix plant. As a result, one weekend in June 1997, a specific configuration of the system was planned to analyse the dysfunction.

But because communication between the different people involved was insufficient, and also because of a certain lack of experience, two operations employees opened the drainage valves in the volutes of the condenser's cooling pumps when the pumping station had not been isolated from the Rhône by sluice gates. The water rose quickly and the two employees evacuated the pumping station as it was flooding. In the next few hours, the sluice gates were installed and the water pumped out with mobile units on the site. It then took several days to revise all the equipment that had been submerged.

In conclusion, this operation, which had been designed to prevent the loss of the so-called neutrons, highlighted a risk of losing information through the renewal of staff and the power plant's intermittent operating. The incident was the subject of an analysis and corrective measures, in human, technical and documentary terms. The dysfunction of the "Tapproge" system was also cleared up.



5.8. The 50th irradiation cycle

On 9 April 1998, the Safety Authority announced its approval for the new start-up of the Phénix plant to operate its 50th cycle of irradiation. The power resumption operations were performed in April and May 1998, after three years during which the turbo-generator had remained silent and the reactor had diverged just a few times. The operations successfully completed with reactor divergence on 23 May, connection to the electrical grid on 25 May and achievement of full rated power (137 MW) on 28 May, followed by a few problems requiring disconnection, then return to power on 31 May. The event was proudly greeted by the publication of a new in-house journal entitled "Renaissance".



The primary sodium seen through a slab penetration

Several disturbances disrupted normal reactor operations. These included an isolation flaw on a preheating wire on a secondary loop, problems with the operations of the ionic pump on the hydrogen detection system on the steam generator reheater, two fire starts following an oil leak on a turbine bearing, clogging of the lubricating filters for the turbine safety stop, and a plugged NaK sparger on the primary argon analysis circuit. When the risk of a rapid spurious trip was identified, temporary modifications were made to the processor logic for

the hydrogen detection at the steam generator reheater and superheater outlets, with approval from the Safety Authority.

In late September, a ten-day reactor shutdown period was put to use to perform an intermediate fuel handling campaign. Several maintenance operations were also conducted, and the staff reviewed the upcoming organisation for the renovation, reinforcements, inspections and maintenance. Power operations resumed, often switched off by small problems including the failure of a module in the feed pumps' speed regulation, insulation defect on a preheating wire, failures in the regulation of the main feedwater, turbine tripping following a changeover manoeuvre on a lubricant circuit exchanger.

On 24 October 1998, the President of the American Nuclear Society awarded the Phénix plant with a **Historical Landmark**. The ceremony took place in the presence of several French and foreign leaders, and many present and retired employees were on hand. Phénix was the third facility in France to receive the award, after the PIVER pilot vitrification facility in Marcoule and the ZOE reactor in Fontenay-aux-Roses. It was proudly stated that the Phénix plant had "*accomplished all it set out to do as a prototype and even exceeded expectations in many areas*".

On 9 November 1998, plant management made the decision to voluntarily stop the reactor in order to empty the buffer tank, in order to safely reinforce the frame supporting the tank, an operation which had been deemed necessary based on the results of the earthquake resistance calculations for these frames.

During the operations to start the reactor back up, signs of a **decrease in the volume of sodium** became clear in the N. 2 secondary sodium circuit (between 5 and 10 m³), and an increase in the volume of sodium in the reactor vessel

was also seen (on 13 November). The event was identified as due to a leak in the intermediate heat exchanger on this circuit (approximately 50 litres a day, reactor operating) the origin of which went back to the beginning of the 50th irradiation cycle. The CEA then made the decision to move up the shutdown for renovation, reinforcements, tests and maintenance, given that this exchanger had to be replaced and that the 50th irradiation cycle had

made sufficient progress to prepare the upcoming research program.

After the circuit was drained, the inspection and monitoring of the sodium levels in the two intermediate heat exchangers on the N. 2 secondary sodium circuit identified heat exchanger "I" as the source of the leak and undertook the preliminary steps to replace it (cf. § 6.5).

All in all, during the 50th irradiation cycle, the reactor had operated 77 equivalent full power days (EFPD) and produced 382 million kWh, bringing the net production of electricity to 20.88 billion kWh since the beginning. It had had a 70% availability factor when this cycle was cut short.

Handling the fuel in sodium

Positioning the rotating plug and the handling arm for the handling of the sub-assemblies in the core is a delicate operation. It is performed by operators who connect the different automatic sequences and make a certain number of checks to avoid any positioning errors. To do this, they use notably a mimic panel for simulating movements.

The angles of rotation of the rotating plug and handling arm are calculated for each position in the core. Before each manoeuvre, the controller on the simulator performs the movements described in the handling program and positions the dummy arm of the model on the resulting position. The number of the sub-assembly is then checked to make sure it corresponds to the one programmed.

It is only then that the operator performs the rotation movements, operating the engines manually. Using binoculars, he reads the values of the verniers as they go past. He stops at the right place and perfects the angle displayed with precision. This vernier is also controlled by a camera which retransmits the image to the handling control room, where the handling operator compares the angle of rotation with the value given in the handling program.

Furthermore, small plates have been engraved on the floor of the reactor hall. These plates bear the numbers of the positions of the sub-assemblies. When all the movements have been performed, with a camera it is possible to read the number corresponding to the position reached and to compare it with the number of the position requested. After this final check, the rotation commands are blocked and the sequences for removing or positioning the sub-assembly in its place can begin.

5.9. Safety first

The upgrading of nuclear power plant safety consists in verifying that, after several years of operation, it still presents sufficient guarantees of the absence of risk for the public, the personnel and the environment. Upgrading was also the time to carry out many modifications required to bring the safety level up to the same level as a new facility. It's somewhat as if, during the compulsory technical check done on vehicles, the latest safety equipment (airbags and other equipment) required on new cars was added to the old ones.

A safety upgrading is not easy for an operator who is used to his facility and knows its reliability better than anyone else. The plant had worked so well up until now, what was the point of all these studies and work, especially since their purpose was not to increase electricity production but simply to serve as a guarantee in the event of highly unlikely accidents? In the case of the Phénix plant, the upgrading was complicated by the fact that it was not a clear-cut operation conducted from start to finish in a cohesive, exhaus-



tive manner. The evaluation had been done little by little, from negative reactivity trip to powerful earthquake, from cracks discovered on a pipe and small sodium leaks to worst-case scenarios of loss of all means of decay heat removal and major sodium spray fire.

As a result of the immense efforts deployed by the small team at the Phénix plant, assisted by the technical support from Cadarache and the many engineering firms who also worked on the plant, the safety upgrading dossier gained consistency. Nearly ninety files were sent to the Safety Authority each year. Power operations at the plant were authorised for a few more irradiation cycles, with the objective of conducting the experiments required by axis 1 of the 30 December 1991 law on long lived nuclear waste manage-

ment, that the abandon of Superphénix had transferred to the Phénix plant. Reactor power was definitively restricted to 350 MWt. However, after the 50th irradiation cycle, conducted in 1998, the substantial work of meeting the new standards and facility testing had to get underway with the arrival of the third ten-yearly outage.





Commemorative plaque of the American Nuclear Society



Dominique GOUX

*Head of safety and quality service
at the Phénix plant since May 1990*

Safety at the Phénix plant

The safety of a nuclear facility like the Phénix plant depends on three basic factors: a safe reactor, competent staff and a vigilant national safety authority.



To start with, Phénix had a sound architecture that made it possible to remove and revise or replace many components such as sodium pumps, intermediate heat exchangers and steam generator modules. This is made easier by the fact that there is no pressure in the reactor – a feature that is in itself a safety factor (no risk of a pressure drop causing a cooling failure) and also makes this a radiologically clean plant. Day to day surveillance and maintenance of the plant is greatly facilitated.

Designed at a time when there was as yet little feedback from experience in running power reactors, the Phénix plant incorporated well-developed safety arrangements. These were then strengthened as experience was amassed (e.g. sodium-water reactions), knowledge advanced or the degree of safety demanded increased. Thus a considerable proportion of the safety upgrade work concerned the secondary sodium circuits and protection against large sodium fires, taking additional but highly unlikely situations into account; as a result, considerable reinforcement and partitioning work was carried out.

A nuclear power plant also needs competent staff to run it safely, whether it is in power operation, shut down for refuelling, undergoing engineering work, or running tests. This particularly applies to operating and maintenance staff, who work directly on the plant. As staff who had worked at the plant from the outset retired, there was great deal of work to do to transfer their knowledge to newcomers; this work still continues today.

This applies to all the expertise involved in and around the plant, especially on the safety side. During the 1990s new issues arose, and the safety team was considerably strengthened, going from one "safety officer" to five "safety engineers". This team is also backed up by the skills of several CEA and EDF units and external engineering consultancies.

From 1990, following four negative reactivity trips, a certain number of safety upgrade studies and checks on the reactor block were carried out. Then other

issues gradually came to the fore, such as seismic upgrading or the resistance of the sodium circuit materials, which by then had been functioning under power operation conditions for 100,000 hours. In late 1993, a "Phénix Lifetime Extension Project" was set up in the Cadarache Reactor Research department to conduct an overall safety and availability upgrading of the plant's equipment. The project was supervised by a committee of high-level experts under the chairmanship of the late Xavier Élie^[1], then Director of the Phénix plant. Alongside this, the plant's safety engineers were conducting update studies of accident situations. In this way, starting from relatively disparate case studies and as safety analyses progressed, we achieved a coherent whole in which each hazard is estimated at its true value.

All this was done under the watchful eye of the Safety Authority and its technical support structures, as witness numerous technical meetings and the ten meetings of the Reactor Standing Committee that were held at the Phénix plant between 1991 and 2002. Although relations between the Safety Authority and the plant have not always been easy, the basic reason for this has been the technical difficulty of the issues dealt with and, in some cases, the innovative nature of the methods employed. But thanks to clear explanations and the involvement of representatives of the Administration, we have been able to work in a climate of mutual trust.

It can be said that authorisation to return to power operation, first in 1998 and then in 2003, was due to the motivation and active mobilisation of all the teams involved: the Phénix team, the CEA and EDF teams and our external partners' teams, who all worked together to carry the renovation work through to a successful conclusion and prepare the resumption of normal operation.

All we have to do now is make good use of this authorisation and carry out the planned experiments.

[1] Died in October 2000.



RENOVATION
OF THE REACTOR
(1998 - 2003)



Landmarks

- January 1999 Start of renovation work
- 5 - 10 April 1999 Test on uncovering the reactor block
- 19 September 1999 Mini-tornado whips through the site
- 27 October – 23 December 1999 Inspection check on conical shell
- March 2000 End of ten-yearly overhaul of turbo-generator
- 23 November 2000 Discovery of internal leak in intermediate heat exchanger H
- 28 November 2000 Inspection of module of steam generator N. 2 reveals cracking
- December 2000 End of main seismic reinforcement works
- January 2001 Decision to repair steam generators
- 15 March – 6 April 2001 Visual inspection of core cover plug and upper internal structures of the reactor
- November 2001 End of tests on new emergency cooling system
- 29 March 2002 Delivery of last fresh fuel sub-assemblies
- December 2002 End of repair work on steam generators
- February 2003 End of sodium fire protection work



In late 1997 the Safety Authority accepted in principle the renovation work to be done.

Time was short, because the operation of the plant was part of the CEA's research into transmutation of long-lived radioactive waste and the law of December 30, 1991 provided for a Parliamentary debate in 2006 to decide on this issue for the future. The Phénix engineers and technicians, backed up by those at Novatome^[1], threw themselves into the new challenge. They had to define and carry out, in the shortest time possible, a considerable number of checks and reinforcement, renovation and maintenance operations, some of which were the first ever of their kind.

After the long period of uncertainty punctuated by hopes of a quick start-up and successive disappointments, the start of the renovation work was welcome. It began in a well-organised manner, restoring confidence in the plant's future. The atmosphere was like the early days, with a heavy workload offset by the varied nature of the tasks and human contacts. However, it soon became clear that much more work was required than had been expected.

The stakes were high, however: the Phénix plant was the only facility in the world capable of running the irradiation experiments needed in the early 21st century^[2]. So, despite the many misadventures that caused delays

and increased the cost of the work, the teams kept to target. Keeping to schedule was a race against time; it started as a sprint, but gradually the length of the course stretched out to marathon dimensions, so the runner's strength (the budget!) had to be managed very precisely.

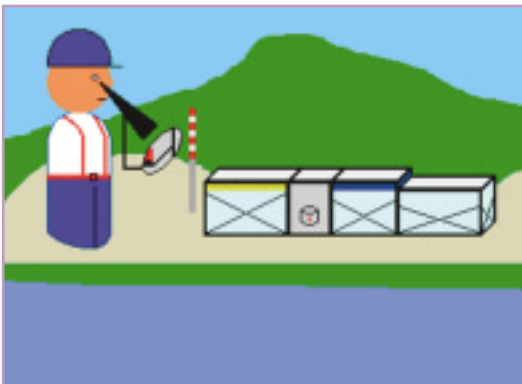
This period does not lend itself to a chronological narration. Instead, we present the different tasks carried out – inspections, seismic reinforcement, sodium fire protection, the incidents on the steam generators and intermediate heat exchangers – before briefly covering the other main operations and ending with an outline of how the work was organised.

6.1. Inspections

The first major inspection, conducted between October and December 1999, was that of the welds on the **conical shell** supporting the core and connecting it to the main vessel. The inspection was designed during the second half of 1997, as part of the safety upgrade. Because the structures supporting the core are of vital importance for control of the nuclear reaction, the least defect found, and even the absence of a reliable conclusion, could lead to definitive shut-down of the plant. The method chosen was inspection by ultrasound, a method developed especially for operation by the CEA's Advanced Technologies Department in 1998 and 1999 and implemented by Novatome and the company COMEX Nucléaire. The principle was to use the shell itself to guide the ultrasound waves from the outside of the main vessel,

[1] Later to become a division of the company Framatome ANP (an Aréva/Siemens joint venture).

[2] The only other similar reactor is the Monju reactor in Japan, which has not yet received authorisation to conduct improvement work following a sodium leak in December 1995. The BOR 60 reactor in Russia and the Joyo reactor in Japan are lower-powered units. The Russian BN 600 reactor is designed only for electricity production. As regards future reactors, it will be at least fifteen years before they are operational.

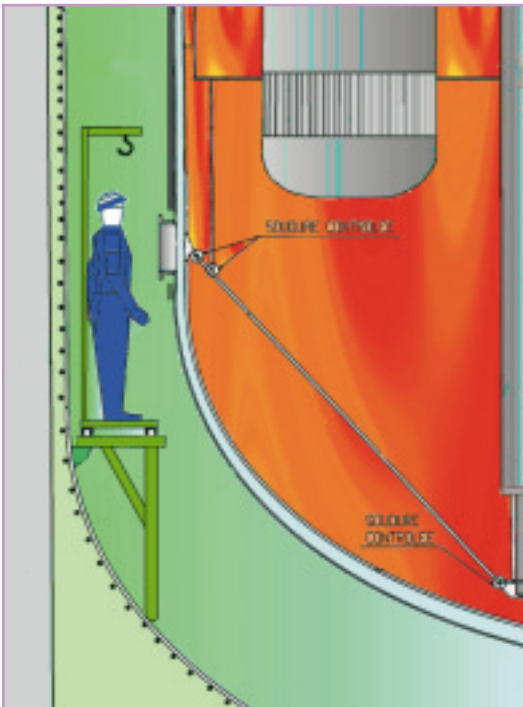


over a distance of more than three metres to the heart of the primary sodium, which was kept at a temperature of 155 °C.

Starting in the first half of 1998, a work platform was installed in the space between the safety vessel and the containment vessel to prepare for the operation. In August and September 1999, the safety vessel (at a temperature of around 130°C) had its lagging removed and five holes made in it, giving access to the main vessel. Nozzles, whose purpose was to let through the inspection carriers, were welded (on a saddle-shaped junction, which meant that the operator had to permanently adapt the distance between the electrode and the vessel wall), then machined (cutting the safety vessel in a 360 mm circle) and checked (by X-raying the welds). Given the environmental constraints, as many as possible of these operations (especially the welding) were

remotely controlled from a control post in the hall of the reactor building. After the inspection the openings were hermetically welded.

Two special carriers - a "short" one for inspections near the opening and a "long" one for inspections up to five metres away) were developed to bring the sensors into contact with the vessel in a 10-cm-wide annular space. This involved the operators going down into an 80-centimetre-wide space between the safety vessel and the containment vessel, so that catwalks had to be installed and the area lit and air-conditioned. Furthermore, because of the radiation emission rate in the area of the intervention, an ALARA^[3] approach was taken so as to limit the radiation doses the operators received; all the operations were simulated in advance and optimised on mock-ups to keep exposure time to a minimum.



Schematic of the conical shell

Validation of the inspection method using two carriers (one long, one short) was finished in September 1999. The data acquisition was achieved between 27 October and 21 December that year and the worksite closed during the first quarter of 2000. The inspection confirmed the qualification of the method. The quality was excellent, and from the results it could be concluded that there was no defect in the welds examined. The operation also proved that inspecting the internal structures of a sodium reactor, although difficult, is perfectly possible.

The **core cover plug**, directly subjected to jets of hot sodium leaving the sub-assemblies, is one of the structures under heaviest wear when the reactor is operating. It is also of crucial importance for safety, because it

[3] The ALARA approach ("as low as reasonably achievable") estimates the dosimetric consequences of an activity and looks for "economically reasonable" ways to reduce them to a minimum.

carries the control rods. As with the conical shell, any crack or disorder in the core cover plug might lead to definitive shut-down of the reactor. Because of the temperature and irradiation conditions in the cover gas, electronic equipment cannot be used here. Novatome therefore carried out a video inspection using optical devices rather like inverted periscopes up to 19 metres in length. These were introduced inside the reactor block after draining out half the primary sodium (about 400 metric tons).

To make sure it would be possible to carry out the inspection, a reactor block draining test was performed in April 1999 : 130 metric tons of sodium were transferred from the reactor block to the storage tanks and back again; this test provided the opportunity to check the thermohydraulic state of the partly uncovered primary circuit. To carry out the inspection, three inspection devices were ordered: one panoramic device (providing an overall view of the visible structures), a lighting device and an inspection device (imaging a 5 cm x 5 cm area at a resolution of about one-tenth of a millimetre). Developing these devices involved both technical and contractual problems, which were solved, but which delayed the job by eighteen months. In particular, on the inspection device, the light sent out for imaging took the same optical path through the tube of the device as the incoming image. In tests with the device in late 1999, parasite reflections spoil the quality of the image received. Successive modifications had to be made to the device before the problem was definitively solved.

The devices were delivered to the site in December 2000 and underwent temperature qualification tests in a heating mock-up set up in the hall of the handling building. They were then introduced into the slab

penetrations and used in turn, according to need. The inspection concerned not only the outside of the core cover plug (external shell, bottom grid assembly, thermal shield bolting), but also inside, by removing a control rod mechanism from the reactor and introducing the inspection device in its place. Also inspected were some of the reactor block's internal structures: primary vessel, the penetrations of primary pumps and intermediate heat exchangers, etc.

Sodium, an opaque fluid

For the operator of the Phénix plant, the major disadvantage of using sodium is the fluid's opacity. Arrangements were made at the outset, and have been reinforced since then, as regards taking into account the oxidation of sodium in the presence of air, the risk of fire associated with sodium and the reaction between sodium and water. On the other hand, as it is impossible to "see" through sodium, a whole range of indirect checking methods are necessary, for both operations and in-service inspection.

The handling of the sub-assemblies in the reactor block can thus not be checked directly by the operator. There is therefore increased monitoring of the movements of the machines to guarantee that each sub-assembly is in the right place. Ultrasound devices (Visus) are used to make sure that there are no obstacles beneath the free level of sodium. Similarly, the movement of the sodium in the circuits (filling and transfer for example) must be particularly well monitored, using notably thermal instrumentation.

Finally, controlling the internal structures situated in the lower part of the reactor block can only be done with the help of highly sophisticated means, as there is no way of making a visual check. For this reason, ultrasounds are used here too. It should be noted that progress in the use of ultrasounds to check the welds covered with sodium has been exceptional since the Phénix plant started operating (partly out of the necessity mentioned above). Today, a sodium-cooled reactor could thus be designed in which the means required for the in-service monitoring of these welds are integrated.



The visual inspection of the core cover plug was conducted between 15 March and 6 April 2001. Draining involved transferring about 400 m³ of sodium at 180°C from the reactor vessel to the primary sodium tanks to bring the level in the reactor down as far as the sub-assembly heads. Advantage was taken of this to make a geometrical verification of the core lattice, especially its flatness, to make sure there was no disorder that might be linked to the negative reactivity trips (cf. § 4.6). Once the operation was completed, the sodium was transferred back to the reactor.

The images of the structures observed inside the reactor block were of excellent quality considering the extreme environmental conditions for the optical instruments in the reactor block cover gas. The structures looked very good. It is worth noting that the free surface of the sodium was used as a mirror to see the underside of the core cover plug grid. All the images obtained showed that the structures examined were in good condition. The only significant finding was that one of the eight screws of the upper flange of the core cover plug was missing; these screws were needed for assembling the apparatus but have no subsequent function, and it was shown that even supposing lost screws reached the hot pool, this would not compromise safety.



Core cover plug reflected on the free surface of the sodium

Other verifications further demonstrated the good condition of the plant. In many cases the equipment used was purpose-designed. Ultrasound inspection of nine of the twenty-one upper hangers of the reactor vessel required a special carrier to insert the ultrasound sensors into a ten-millimetre gap. A method for detecting any argon escaping from the main vessel into the nitrogen in the space between the vessels was developed, to ensure that the cover gas zone of the main vessel was leak-tight. Checking the welds of the storage drum vessel confirmed they were in good condition, as had been noted in 1989 and 1991. A major inspection programme was run on the secondary circuit pipes and the steam generators, resulting in numerous repairs (cf. § 6.4). Lastly, the third ten-yearly inspection of the plant involved regulatory checks on components, such as hydraulic testing of the steam generators, revision of the turbo-generator set, tests of the handling devices under load, etc. These operations revealed no abnormal defect after thirty years in operation.

6.2. Seismic reinforcement

Following the seismic safety upgrade study (cf. § 5.4), all the plant's **buildings** had to be reinforced, especially the structural steelwork and the steel reinforcements in the reinforced concrete, but also disconnecting the buildings to prevent them from knocking together in the event of an earthquake^[4]. Parasismic building methods and regulations having changed since the civil engineering of the late 1960s, the study recommended strengthening a considerable number of structural steelwork elements and steel reinforcements in the concrete. Once it had been defined, this work was not particularly complicated; but the fact that it had to be

[4] The earthquake intensity to take into account is VII–VIII on the MSK scale, equivalent to the most severe earthquake ever recorded in the region, with the Phénix plant very close to its epicentre.

Seismic calculations

At the beginning, it was said that the power plant's buildings, which had been built in accordance with the parasismic regulations in force at the end of the 1960s, should be able to withstand a slightly higher earthquake without any significant damage. The specialists were thus asked to make the corresponding calculations.

A few months, and several simulations, later, the first results came in: certain parts of the buildings would not withstand earthquakes of a level VII – VIII. These initially pessimistic results were obtained essentially from the margins introduced into the calculation codes to take into account the uncertainties as regards construction methods, the real dimensions of the parts used most, and on the evolution of their compartment over time, for example. Sometimes, there is quite simply no calculation method compatible with the construction techniques used.

In certain cases, the support and recommendations of experts make it possible to find solutions that are acceptable for all parties. In other cases, as a way of reducing the degree of uncertainty, it is said that "you only have to" collect more information, do additional calculations, conduct ageing tests and so on. But it soon turns out that things are not so simple. You would have to develop an R&D program, and that is obviously out of the question given the time schedule that interests the operator!

And then, at the end of the day, it seemed less of a burden, and quicker, to strengthen the facility in such a way that the seismic calculation models would guarantee that it would hold. All that was then necessary was to find the most intelligent and efficient means of doing the strengthening on the existing facility, in the midst of all the apparatus, pipes, various cables and so on that the operator obviously did not want (and was not able) to dismantle. The whole business was quite an art form!

carried out inside a plant that was in continuous operation made the task far more complex, especially as the companies contracted to do the work were not used to working in nuclear facilities. However, they quickly adapted to the conditions and under Novatome's supervision they worked well and carefully.

Inside the reactor building, the columns were reinforced wherever there were lap joints in the steel reinforcements, so that the transverse stresses from an earthquake would no longer risk shearing the columns. The operations consisted of locating the steel rods precisely (they were not all in the places indicated on the plans), taking cores from the concrete, anchoring the support plates, welding the steel casings and finally injecting resin between the casings and the concrete. Outside the building, after putting up scaffolding from bottom to top of the east and west fronts, the facades and roof were reinforced. This work was done (whenever the wind permitted!) between April 1999 and June 2000.



Hall of the reactor building during seismic reinforcement work

The structural steelwork of the north handling building was reinforced in 1999 and 2000, especially in the area of the irradiated elements cell, the upper hot cell, the metrology room and the fresh fuel storage room. Earlier, meas-



Measurements of surface contamination of the walls had shown that no particular radiation protection precautions were needed, either during the work or for disposing of the waste produced in the process. To prevent the south handling hall and the adjoining north handling building knocking together in the event of an earthquake, a spacer seal between the two structures was sawn off. Seismic reinforcement of the south handling hall was carried out in 2002.

The most extensive reinforcement work was in the steam generator building; because it is not a nuclear building, the safety requirements at the time it was built had been less severe. The purpose of the work was to prevent the building collapsing in the event of an earthquake, which would cause the secondary sodium pipes to collapse and so cause a major sodium fire and also the loss of the decay heat removal circuits. All this work was done between January 1999 and August 2000:

- reinforcing the skeleton of the wall next to the reactor building,
- increasing the gap between the superstructures of the steam generator building and reactor building, by sawing off protruding portions of concrete,
- strengthening column base anchorings in the sodium zone,
- strengthening the structural steelwork of the east and west outer walls,
- removing the water tanks from above the handling bay^[5],
- strengthening the roof and skeleton of the handling bay,
- installing a new row of columns in front of the steam generators. The superstructures of the handling bay now rest on these columns,
- increasing the gap between the foundation slabs supporting the superstructures.

The latter three operations separate the superstructures of the sodium zone of the steam generator building (which is on the same foundation slab as the reactor building) from those of the handling bay (which shares the same foundation slab as the turbine hall).

In the turbine hall, the purpose of the work was to prevent it collapsing on the steam generator building. The column bases and some ties on the structural steelwork were reinforced, as were some parts of the foundation slab to which additional steel reinforcements were added. Metal liners were anchored around the concrete columns supporting the foundation block of the turbo-generator set. The breeze-block wall separating the steam generator building from the turbine hall was taken down block by block, and replaced by metal cladding.

The **emergency cooling circuit** serves to cool the reactor pit and the reactor slab. In the event of an accident involving loss of all the secondary sodium circuits, it would also serve to remove residual decay heat from the reactor. The design of that part of the original circuit that was outside the reactor, and which included the water-to-water heat exchangers taking their cold water from the Rhône, were not in compliance with current standards for proving resistance to a high-intensity earthquake.

Consequently, two new, entirely independent circuits were built, at some distance from the building housing the secondary circuits. Each one incorporates two pumps and an air cool-

[5] New tanks had already been installed in the machine room (cf. § 5.3).



er with ventilators and is linked to its own diesel-driven generator. These systems are designed to function in the event of an earthquake or major sodium fire. Seismic standards and the nature of the ground make pile foundations necessary, with the piles sunk as much as 18 metres down. To test their heat exchange capacity, the air coolers were tested in summer, with the reactor reaching a temperature of 400°C. The heat exchange capacity was less than expected, but was acceptable once the adjustment of the circuits' operation had been modified.



Emergency cooling system air cooler (on the right, near the pump room)

By contrast, fine-tuning of the air coolers, which can operate on a water-spray basis, required repeated studies and tests to make sure they would work efficiently both in extreme cold (-17°C) and during a heat wave (+35°C). As a result, reheaters were added to the water headers at the inlets to and outlets from the air coolers, and changes were made to the instrumentation

to allow the air coolers to function in water spray mode across a wider range of temperatures. At the same time the instrumentation of the system was redefined because the operator had criticised the first version as being too complicated.

Seismic reinforcement of a complex facility is difficult owing to the interaction between the different **components** that make it up. For example, work on the buildings changes their particular vibration frequency and movements and hence the stresses imposed on their component parts. So several sequences of calculations are needed. This concerns, for example, the travelling cranes; but it is particularly difficult to carry out surveys and make the consequent alterations to these cranes because they are so useful as handling equipment for the rest of the work. Particular circuits such as the oil cooling circuit of the storage drum also had to be consolidated to make sure they would continue to function in the event of an earthquake. Lastly, to eliminate all risk of fall onto the new western emergency cooling circuit, the argon and nitrogen production stations were altered and moved.

6.3. Protection against sodium fires

The purpose of **partitioning the steam generator building** was to separate the area containing sodium pipes from the area containing pressurised steam or water pipes^[6]. The operation also separated the two secondary sodium circuits that will be used in future (circuit N. 1 to the west and N. 3 to the east). Limiting the future power of the reactor to 350 MWt means that two steam generators

[6] In the original design, there were two lines of defence against the risk of reaction between the secondary sodium and the air in the building or the water of the steam generators: prevention and protection in the event of accident. The additional measures are intended to add a third line of defence: limiting the damage from a particularly serious accident.



will be sufficient. In these conditions, it is best to definitively abandon secondary circuit N. 2 rather than keep all three (two in service and a spare one), as this would generate a lot more work without giving many more guarantees of plant availability.

The partitioning consisted of metal structures carrying insulating panels (about 1250). These panels can be taken down to allow maintenance work on plant inside the sodium zone. With the new partitions, fire due to an accident on a sodium pipe cannot affect a pressurised steam or water pipe that would exacerbate the sodium fire if it failed. Similarly, a sodium fire caused by a leak in a secondary sodium circuit cannot spread to the neighbourhood of the other circuit; this way there is no need to empty the second sodium circuit and leave the reactor without a decay heat removal circuit. The partitions are designed to resist an earthquake plus a

major sodium fire raising the ambient temperature to about 1100 °C for 30 minutes.

The first structural steelwork was put up in April 1999. The work was disrupted several times, first by all the other work going on in the same area (seismic strengthening of the building, pressure tests on the steam generators etc.), then by changes in priority between the two zones owing to analysis of the critical path of the work following incidents that had arisen, and finally because the sub-contractor responsible had gone into liquidation. This last matter greatly slowed down the work from October 2000 and halted it in May 2001. Work began again, under a new contract, in April 2002 and was completed at the end of January 2003.

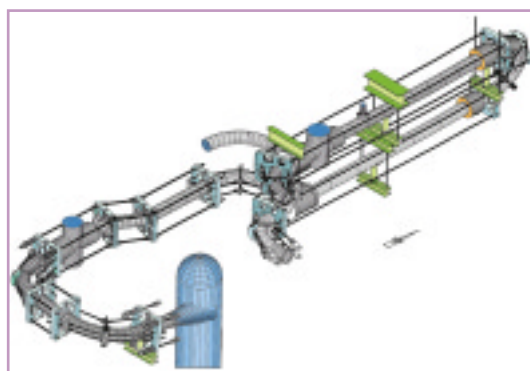
Other methods to protect the steam generator building from major sodium fires were also introduced. These were dampers for closing the ventilation air intakes, failure panels on the sides of the building, a new conventional fire detection system and a sodium fire detection system (Na 101: flame spectrometers), heat insulation under the ceilings of basement rooms in the building, insulated channels around the secondary sodium circuit draining lines and protection for the electrical conduits.

Anti-whip devices were installed to prevent a failed pressurised steam or water pipe slapping violently against support structures. A first study suggested devices tried and tested in other facilities, but the Safety Authority wanted devices that would work in the combined event of earthquake and high pressure pipe failure. The consequent changes to the dimensions of the frames and stops and the necessary reinforcement of the structural work meant that there was not enough room to install them in the steam generator building.



Steam generator building partitioning

An innovative solution was found by Novatome: each circular weld is encircled by a strap to prevent a clean break and the strain is taken by longitudinal tie-rods leading to anchor points, to prevent pipe whip. These devices also have to allow the pipes to move freely in the event of temperature or pressure changes. All this made the design work a delicate task. To install the devices, the lagging had to be removed from the steam pipes; the work lasted from October 1999 to February 2001.



Steam circuit anti-whip device

On verification, the **resistance of the steam generator casings** was found insufficient, once the new seismic resistance standards were taken into account, to guarantee against the risk of collapse. These casings support both the sodium pipes and the high pressure steam and water pipes. All the steam generator casings were reinforced between March 1999 and April 2001, with a few final touches in late 2002:

- installation of internal tie bars to limit the gap between the east and west sides of the casings, and modification of cross-beams on these sides,
- addition of failure panels on the south side to evacuate excess pressure due to sodium fire in the event of module failure,

- addition of three more valves to prevent pressure drop in a casing at the end of a hypothetical major sodium fire,
- local reinforcements to compensate for the preceding two modifications,
- heat insulation of the posts and struts of the casings in the southern area,
- installation of anti-whip devices near the water and steam sub-headers.

Scaffolding

The scale of the work undertaken throughout the facility, particularly in terms of civil engineering, required huge scaffolds of more or less bizarre appearance in order to adapt to the facility's different structures. The amount of scaffolding, along with its great height and complexity, soon made it a permanent source of worry for the safety of everyone involved.

One day in 1999, an employee burst into the safety engineer's office, white as a sheet, saying that he had just had a lucky escape from a fall. Once on site, the safety engineer observed that a guard rail had been removed and that the platform, which was narrow at that point, thus had nothing to protect the workers from falling. Despite frequent visits to the worksites, this obvious lack of protection had not yet been noticed.

It in fact turned out that one hour earlier an area of scaffolding had been dismantled by the sub-contractor who had not noticed that it had a double function and acted as a protection against falls at the level of the platform. All's well that ends well, as they say! But this "near miss" clearly shows that, despite very strict monitoring of worksites, the configuration of the site changes so quickly that it is difficult to always keep an upper hand.

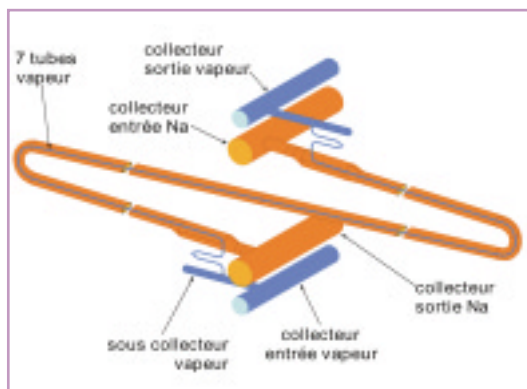


Lastly, some auxiliary lines of the secondary sodium circuits were moved so as to include them in the partitioning of the sodium zone. Gaging devices were installed on the balancing lines between the buffer tanks and expansion tanks and on the circuit monitoring for hydrogen in the sodium, to limit flow in the event of a leak or failure. Portions of auxiliary circuits that had rarely or never been used since the plant opened were removed in order to limit the risk of leaks and consequent sodium fire in the building. The hydrogen detection mixers were replaced by better-designed devices.

6.4. Repair of the steam generators

From 1998, an **inspection programme** was run on the modules of the Phénix steam generators to assess their condition after 100,000 hours of power operation and make sure they were safe for the 720 effective full power days to come (about 30,000 hours). Thus a module of steam generator N. 1, and twice four modules of steam generator N. 2 (which will no longer be used) were removed, cut up and examined one after the other. The first examinations showed a few shallow cracks on the welds of the sodium piping (in 321 steel), linked to the manufacture of these devices, and growing only very slowly in service; studies conducted by the CEA showed that they would be stable under accidental load (earthquake) at the end of their lives.

However, during the last examinations in late November 2000, a more serious crack was found, penetrating two-thirds of the way through the steel. This crack was caused by delayed reheat cracking which has been shown to be connected with high operating temperatures at the



Schematic of a steam generator module

steam generator modules inlet and the geometrical features of this area, including some triple weld points.

Consequently, given the possibility of a similarly serious crack existing on the modules of generators N. 1 and 3, the corresponding safety report could not be issued. The CEA therefore decided not to seek authorisation to resume power operation in the summer of 2001 as planned. The cost and the delay of changing the steam generators' reheater modules would be prohibitive, and the palliative solutions examined, although more reasonable in terms of cost and performance time, all presented uncertainties that might, if they were implemented, prevent the resumption of power operation.

The solution finally chosen, with a view to limiting the extent of the repairs, was to eliminate those parts of steam generators N. 1 and 3 that might be affected (the shell of the sodium inlet and the first elbow in the superheater and reheater modules) and replace them with identical parts made from 316 steel, a type of steel that is less subject to cracking. Following discussions with the Safety Authority, it was also considered necessary to carry

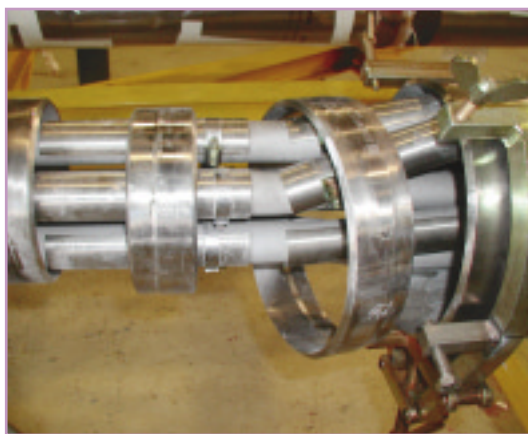
out an ultrasound inspection of the welds on the cold elbows and the longitudinal welds on the sodium inlet shells in order to ensure that there were no defects in those areas.

The main difficulties of these repairs were ensuring that the sodium washing was both effective and harmless, and the welding work. All risk of caustic cracking (due to traces of soda left in the module and occurring either during the repair and on resumption of power operation) had to be eliminated. The welding work required special attention to avoid causing new incipient cracks in the aged metal of the modules' sodium pipes. Feedback from replacing parts of the secondary circuits made of 321 steel showed a certain susceptibility to the formation of welding defects in joints where new 316 steel was welded onto aged 321 steel. Special precautions were taken to control this risk (use of intermediate sleeves, specially qualified welders etc.) and the post-welding inspection was particularly rigorous. The reagent used for washing was an increasing injection of water in nitrogen, sweeping across the module from bottom to top. To control the reaction, temperature and hydrogen concentration were monitored and the relative quantities of water and nitrogen were adjusted accordingly.

The decision to repair the steam generators, and the method chosen, were validated in January 2001. That year was devoted to re-examining the reports on the modules and the consequent extent of the repairs, developing and qualifying operating procedures for washing and repair and drawing up safety report for the steam generators, taking the new adjustments into account. This report

were the subject of close and highly critical examination at two meetings of the Permanent Group in July and September 2001. The engineering work itself, carried out by a group of companies, began laboriously in November 2001 and was completed industrially in November 2002.

The actual repair work on the 47 superheater and reheater modules of steam generators N. 1 and 3^[7] consisted of replacing, for each module, the divergent and convergent cones of the sodium inlet shell; these are the only zones where any cracks of significant size had been found. However, the hot elbow was also replaced, partly to take into account weld shrinkage occurring when replacing the inlet shell and partly to eliminate a zone where operating temperatures may have fostered creep phenomena. Replacement of the two longitudinal welds of the sodium inlet shell and the cold elbow were also qualified in case the non-destructive tests of these components showed some defects, but in the event this was not the case.



Repairing a steam generator module

[7] Superheater module N. 9 of steam generator N. 1 was new (it was installed in October 1998). Consequently, its sensitive zones were checked and no repair was carried out.



To keep **control of the repair operations** and to demonstrate that control, prior tests were run on a module of steam generator N. 2, which was then examined destructively. All the repair operations were tested and validated and all the operators and operating methods were qualified according to the standards codes in force, with draconian selection criteria. Lastly, overall qualification was performed for the operations on a module, covering all the operations planned during the repair and up to requalification.

The repair principle had already been employed at the Phénix plant, particularly for replacing the sodium headers of the steam generators. It was based on interposing intermediate sleeves between the parts left as before (aged 321 steel) and the new parts (316 steel). These sleeves are slightly

conical and of varying thickness so that, with a little final "sculpting", they precisely fit both the dimensions of the reductions or ends of the new elbows and the dimensions of the old shells. The presence of these sleeves makes the "new on old" welds accessible for effective non-destructive tests. Then the sodium piping was rebuilt with the help of half-shells of 316 steel; the corresponding "new on new" welds were checked externally. All in all, the repair of the steam generator modules required some 1500 welds, and about 8500 gammagraphy tests to check those welds.

6.5. Tightness of the intermediate heat exchangers

Two intermediate heat exchangers were found to **leak from the tube bundles**, the first (I) in November 1998 (reactor in operation, cf. § 5.8), the second (H) in November 2000 (reactor shut down). Each of these intermediate heat exchangers in turn was taken out of the reactor, washed, decontaminated, dismantled, taken to the hall of the old G2 reactor serving as a workshop, and inspected so as to locate and inspect the leak. Meanwhile they were replaced by new heat exchangers of recent manufacture.

The inspections showed that the cracks were in the tube inner wall in the expansion zone^[8] under the upper tube plate. The origin of the cracking was corrosion under stress due to the presence of soda or polluted sodium, on the secondary side, during the phase when these intermediate heat exchangers was emptying (secondary sodium circuit N. 2, to which they were con-

[8] "Expansion" refers to the mechanical operation of expanding each tube so as to force the tube wall tight against its hole in the tube plate, so keeping it in place and achieving a preliminary level of tightness. The seal is perfected with a ring of welding.

Novatome

All the work of strengthening, renovating and performing the numerous essential safety checks were the responsibility of the contractor Novatome, the engineering division of the reactor sector at Framatome ANP (a joint Areva - Siemens company). A structure was installed on site as early as 1994, and approximately forty permanent staff were integrated in 1999 to make a project team of 90. It included staff for supervising the works and interventions, as well as part of the project operating team in order to remain close to the power plant's orderers and in direct contact with the sub-contracted companies.

Other people joined this structure for specific missions of variable duration. An example of this type of mission was the ultrasound checks of the welds in the reactor block's conical shell. Beyond Novatome's expertise in the field of fast neutron reactors, it also made use of the wide range of skills available at Framatome ANP to do monitoring missions, non destructive checks, quality assurance, follow-up, tests and so on. Novatome thus participated in the success of the renovation and reinforcement work, as well as in that of the checks made on the Phénix plant.





New intermediate heat exchanger

nected, had been repeatedly opened for work between 1995 and 1997); the cracks were worsened by increases in reactor temperature. Directives were issued to avoid allowing this type of risk to arise again on any sodium circuit in the plant.

In view of the prolongation of the reactor's life, the **manufacture of three new intermediate heat exchangers** and two dummy heat exchangers to plug the penetrations was begun in 1994. Intermediate heat exchanger K was delivered to the plant in

December 1998 and replaced exchanger I in the reactor. In January 1999, the tightness test carried out by the manufacturer on intermediate heat exchanger M, just before its delivery, failed to meet the criteria. A search to locate the defects was carried out in the factory. Two cracks and six non-through-wall defects were repaired. In spite of this, a further tightness test proved negative again. The tightness test on intermediate heat exchanger L also failed to meet the criteria. A through-wall defect was repaired. A new test then found satisfactory tightness on this exchanger.

An X-ray survey of tightness welds between the tubes and the lower tube plate was carried out on intermediate heat exchanger M. It showed a significant number of porosities in the welds. The manufacturer was therefore

Supervising the sub-contractors

Supervising suppliers has always been one of the primary concerns of the preparators et activity managers at the Phénix plant. This is because such supervision is essential for guaranteeing the quality of the sub-contracted services. From the days of its construction, the project controllers visited the suppliers of key elements to make sure that the technical specifications were being respected. This practice continued throughout the operating of the power plant when replacement equipment had to be ordered.

Consequently, and in accordance with the requirements of the quality directive dated August 10, 1984, this practice has become codified. An action of systematic selection of suppliers (based on the objective criteria of organisation and aptitude) has been developed, with the emphasis on planning and scheduling actions and keeping any proof of supervision. Be it a complex service such as the inspection of the conical shell of the reactor's vessel, or the simple replacement of a filter, any supplier whose activity has an impact on safety is assessed according to very strict procedures. The suppliers' work, either in their own factories or on site, is supervised by employees from the Phénix plant or an independent organisation, and all the quality documents are examined closely and archived.



asked to check the tightness welds between the tubes and upper and lower tube plates of intermediate heat exchangers L and M, to manually repair any defective welds and to also check intermediate heat exchanger K, installed in the reactor in February 1999, which risked having the same defects since it was of identical manufacture.



Inspecting the tubes of an intermediate heat exchanger

During the summer of 1999, inspection revealed non-compliant welds on heat exchangers M and L. The defective welds of intermediate heat exchanger M were repaired. Even then, the tube bundle did not meet the tightness criterion. In January 2000 a second series of investigations revealed cracks at the edge of the melted zone of the same welds. After repair, the tube bundle was put through a further tightness test and this time passed the test. Reassembled and tested, intermediate heat exchanger M was delivered to the site on 19 July 2000. A few months later it was introduced into the reactor to replace defective heat exchanger H.

Given the fact that the porosities in the welded tube/plate seals are unrelated to the lack of tightness found in intermediate heat exchanger M, and allowing that exchangers K and L passed the tightness tests, acceptance of these two exchangers was finally

pronounced; heat exchanger L and the two dummy heat exchangers were delivered to the site in July 2001 to be kept as spares.

6.6. Other work and innovations

The **operating data acquisition platform**, developed in 1991 to record parameters that might explain the negative reactivity trips (cf. § 4.5), was replaced by a new system in 2001, as part of the renovation of the plant. This gives the plant an efficient and reliable monitoring tool to the end of its life. Five hundred measuring channels are covered, at frequencies between 1 and 250 Hz. Acquisition of the most important measurements (80 channels) has redundancy. Reactor operation is only enabled if a minimum set of these measurements is operational.

Meanwhile, the computer handling fast processing of core temperatures and the computer handling centralised processing of information from the plant to make it available to the operators in the control room have both been replaced by hardware of recent manufacture. The old and new equipment was tested in parallel throughout the 50th cycle before the old one was definitively stopped and removed from the control room. The original computers had functioned for nearly thirty years, an exception lifespan for this type of hardware. The batteries and switch cabinets of the 220 V a.c. instrumentation were also given new-generation hardware, as were the pneumatic plant and the 170-bar compressed air distribution network. And as with all other nuclear power plants, software and automatic controls important for safety had to be checked for the passage to the year 2000.



SARA

SARA (fast acquisition system) is a computerised system which makes continuous recordings, 24 hours a day, of the almost 1,500 signals emitted, in most cases by sensors (flow, temperature, pressure, neutron flow, vibrations and so on) placed inside the reactor block. The system was installed in 2001 and has two main functions:

- in-service monitoring of the reactor block, in order to detect as early as possible any abnormal evolution capable of having an impact on the availability or safety of the power plant. To do this, the signals are compared with each other and in relation to the situation of the power plant. For example, it checks that evolution in reactivity is perfectly synchronous with the movement of the control rod, or monitors the vibrations in the core cover plug to control their stability,

- recording all the signals that could be useful for determining the origin of an emergency shutdown by negative reactivity similar to those that occurred in 1989 and 1990, should such an event occur again.

The frequency of acquisition depends on the sensors and transmission systems. Every 4 milliseconds, 72 readings are taken, and 232 every 10 milliseconds. The others have an acquisition period that varies between 1 and 10 seconds. Certain important channels are doubled. One day of acquisition represents 3.5 billion measurement points (15 Gigabytes before compression). A signal processing and visualisation system provides the engineers and technicians in the Physics department with information in either synthetic form or, on the contrary, as detailed as desired. Certain examination tasks have been automated. All the readings are examined every working day and are the subject of a formalised follow-up.

Fire protection was re-examined in the same way as the analyses performed on EDF pressurised water reactors. A number of significant alterations were made to prevent a major fire or reduce its safety impact: fire-stop doors were installed, smoke removal systems improved, the structural steelwork insulated, automatic

fire extinguishing devices installed in certain sensitive areas, fire sections for fire containment were created, etc.



Fire-fighting equipment

Lastly, the **third ten-yearly regulatory inspection** was carried out, with all the mechanical and electrical activities that entailed. These were scheduled in the periods between the priority renovation and inspection operations. Not everything in the ten-yearly inspection went smoothly. For example, the stem of some sodium valves failed to push the cap snugly into their seats, there was seepage from the steam pre-heater coils of the secondary dump tanks and the speed regulation of a primary pump suffered random breakdowns which the electricians had difficulty explaining.

In addition, the reactor's sodium components also had to be overhauled. The primary pump removed from the reactor in 1997 was completely dismantled, its parts examined and some replaced; its reintroduction into the reactor was postponed several times for scheduling reasons and finally took place in May 2002. Two control rod mechanism bellows were not perfectly tight and had to be replaced. The tightness valve of the fuel transfer lock, on the reactor side, was overhauled. The fuel handling arm lifting tube was cleaned of sodium compound deposits that were preventing it from working smoothly.



The steam pre-heating coils of the primary sodium tanks were replaced by electric trace heating. The sparger (containing a liquid sodium-potassium alloy) on the primary argon circuit was replaced by a new device brought from Creys-Malville, though its environment had to be adapted.

The **turbo-generator set** was given its ten-yearly overhaul between October 1999 and April 2000. Following the results of the magnetic particle testing, three rotor blades of the turbine's low-pressure cylinders were replaced. Then in August 2002, slight corrosion was observed on the rotor blades of the turbine's low-pressure cylinder. This was due to saturation of the air dryer of the low-pressure cylinder during tests with the condenser. Both low-pressure cylinders were opened to clean the rotors and remove two blades to inspect their attachments to the shaft, which proved to be in satisfactory condition.



Turbine undergoing overhaul

When the **steam valves** were examined for their ten-yearly maintenance check in mid-1999, the measurements taken suggested that the stellite coating of the valve seats had become too thin owing to the many machining performed earlier to ensure that they would be leak-tight in

service. After verifying that this situation was unacceptable, the decision was taken to repair the twenty valves involved in the rapid depressurisation isolation sequence of the steam generators. It then emerged that it would cost about the same and take about as long to repair the valves as it would to replace them with new ones (owing to the environmental constraints, the size of the justification files to be put together, etc.). Other steam valves were examined using more precise methods; fortunately, these examinations showed that there was still a sufficient thickness of stellite on the valve protection caps and seats except in a few valves, which were replaced.

In June 2001, the requalification tests on the condenser's raw water circuit was interrupted owing to fouling of the filters caused by silting up of the **pumping station**. Divers discovered a 1.5 to 3 metre layer of mud; the pumping station would have to be cleaned out. This was done between February and June 2002. The silt in waterways upstream of the filters and in the suction lines between the Rhône and the pumping station turned out to be much thicker than expected: the pipes were clogged with mud to half their height. About 750 m³ of silt was removed from the suction pipes and 400 m³ from the waterways.

The raw cooling water circuit was returned to service after cleaning the condenser. The flow rates in the raw water pipes were measured in September 2002 to evaluate the plant's heat balance, using devices installed for the purpose.

All the **logistics** for this major plant overhaul had to be organised with the plant in operation. This included

- installing temporary buildings for the shut-down support team and reinforcements,
- installing and managing additional cloak-rooms for the contractors,
- fitting out a reception area for the contractor companies,
- building, monitoring and dismantling scaffolding,
- various safety works (guardrails, lighting etc.),
- keeping the premises and surrounding area clean,
- removing ordinary waste and waste from the controlled area,
- and more.

6.7. Organisation of the work

The shutdown of the Phénix plant for renovation, reinforcement, inspections and maintenance began on November 9, 1998. The target date for completion of the work was set for January 31, 2000 and the target date for re-connection to the grid was April 1, 2000. Almost immediately the schedule had to be changed to allow for removing intermediate heat exchanger I for inspection. It was changed again with each unforeseen development that arose: removal of four modules from steam generator N. 2 for inspection, factory repair of the new intermediate heat exchangers L and M, delays in manufacturing the optics for the devices required for visual inspection of the core cover plug, additional work to upgrade some of the sodium and steam valves, delays in completing various renovation tasks whose difficulty had been under-estimated, etc. In September 2000, the new start-up of the plant was scheduled for

the end of the second quarter of 2001. Because of these delays, the CEA decided to postpone the definitive shutdown of the plant. Rather than setting a new date (previously set for end 2004), the CEA announced that six irradiation cycles would be run; this would take about five and a half years from resumption of power operation.

In January 2001, after the crack had been discovered in the sodium pipe of a module in steam generator N. 2, it was decided to replace all the potentially affected parts of the

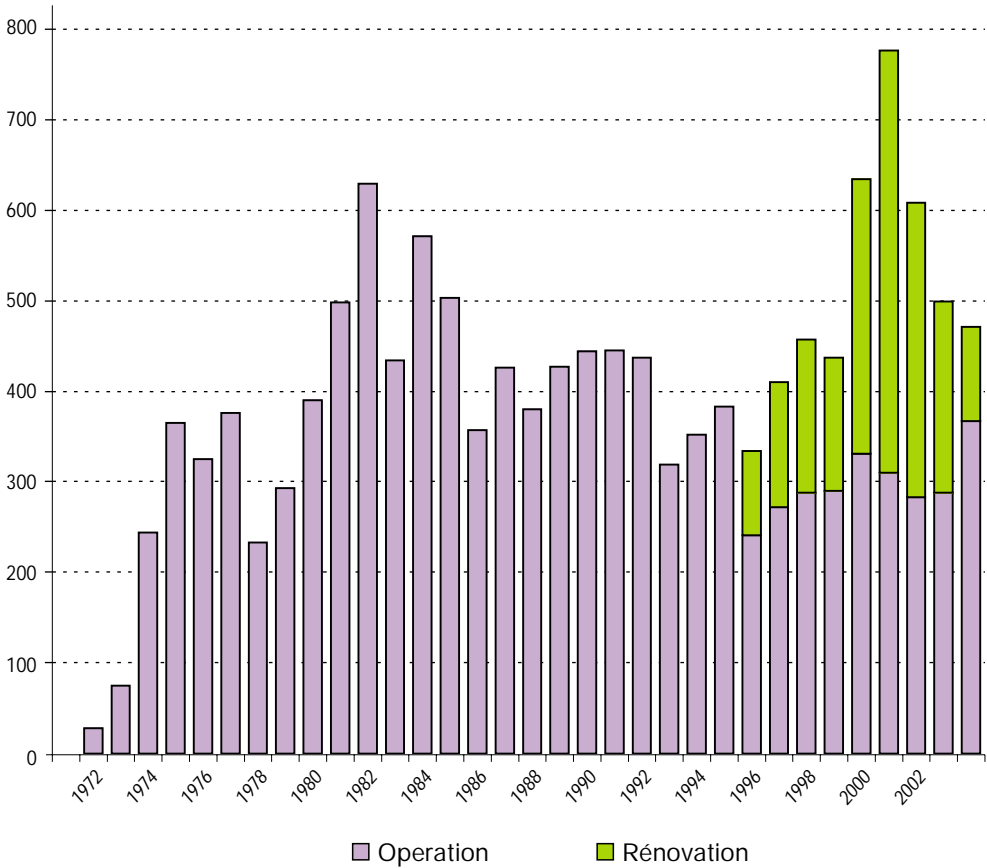
The cost of the work

The project for strengthening, renovation, verification and maintenance was initially scheduled over 18 months. In the end, it lasted 4 years and its cost was also increased. The main reasons for this were:

- the desire to make the timetable as short as possible, which led to starting one piece of work before the studies had been completely finished,
- the additional requests from the Safety Authority in the course of the project, with a direct consequence on the cost of the work and an indirect consequence on the global optimisation of the project,
- the fixed rates for the worksite, proportional to the duration of the work. The cost of the ten-yearly regulatory inspection and overhaul, estimated on the basis of that of 1989, turned out to be very different,
- the prototype effect: certain operations, such as checking the conical shell and core cover plug, had never been performed and thus required prior development. Their cost was difficult to estimate,
- the co-activity between worksites and worksites / operating activities, which led to disturbances, leading in turn to getting behind schedule and extra costs,
- the condition of the material, requiring more renovation than initially thought,
- the preparatory works that were sometimes underestimated,
- the problems involved in justifying that the work was not necessary; on certain occasions, the calculations did not make it possible to provide undeniable proof of the resistance of some structures, so reinforcement work was what was chosen in the end.



Phénix plant expenses (Millions of Francs₂₀₀₀)



The plant expenses were counterbalanced through the generation of electricity (22 billion kWh at the end of 2003, equivalent to 5 billion Francs₂₀₀₀), the acquisition of data on fast neutron reactors and the development of irradiation experiments for the CEA and other third parties. Expenses also cover plant renovation costs (green colour) from 1995, equivalent to €160 million (or 1 billion Francs₂₀₀₀).

Phénix plant construction costs totalled 800 million Francs₁₉₇₄ (approximately 3.2 billion Francs₂₀₀₀)



steam generator modules, i.e. the sodium inlets to the superheaters. It was estimated that this work would take about a year. The other activities on the site were reorganised to fit the new priorities. When the definitive extent of the repair of the steam generators was validated following Permanent Group meetings in the summer of 2001, completion of the works was announced for the end of 2002, with resumption of power operation in the first quarter of 2003. The repair of the steam generator modules kept to schedule, but with several adjustments along the way. After final verifications, the plant was ready to resume power operation in the second quarter of 2003. The renovation, reinforcement, inspection and maintenance of the Phénix plant had taken four and a half years.

The **organisation of the plant** was profoundly revised during the works. Organised into four departments right from the beginning, the plant now also had a temporary project structure, partly funded by the engineering departments of the Military Applications Department of the CEA and the EDF's Lyon Engineering Centre^[9]. After a learning process, all the activities of the shut-down for renovation, strengthening, inspection and maintenance were directed by a project leader for the "resumption of power operation" project and included:

- A shutdown manager, responsible for coordinating all the works. He was backed by groups to deal with planning, accommodating contractors, logistics, work safety and local assistance to workers,
- the Renovation Project, which supervised Novatome's activity as prime contractor,

[9] In 2000 the EDF's Lyon Engineering Centre (Centre Lyonnais d'Ingénierie) changed its name and its purpose, becoming the Engineering Centre for Dismantling and Environment (Centre d'Ingénierie Déconstruction et Environnement CIDEN).

- the Ten-Yearly Maintenance Project which covered almost all activities carried out by the plant technicians directly, or which they sub-contracted,
- the Conical Shell Inspection, Core Cover Plug Inspection and Steam Generator Module Repair Projects, which were more short-lived,
- organisation of the requalification tests.

As well as the technical side, there was a great deal of work to be done on the **contracts and administration** side. Because of the scale of the project and the types of work involved - often very different from the work of running the plant - extensive use of subcontractors was essential. But even a prime contractor



brought in from outside has to be supervised; ultimately it was the plant's activities managers who took responsibility for all operations. The engineers and technicians of the Maintenance department, who usually had full control of such work down to the smallest detail, had to play a customer's role in some major contracts, and had to learn their new role as the work progressed. The high monetary value of the contracts, the CEA's increasingly rigorous contractual regulations and the tightening of the worksite admission system after the New York terrorist attacks of September 11, 2001, considerably increased the time devoted to preparation and supervision of contracts, sometimes to the detriment of presence on site. All in all it took three or four times as many person-hours to prepare each contract as it had in 1989.



A few figures sum up the **scale of the works** during the shut-down for renovation, reinforcement, inspection and maintenance:

- € 250 million in studies and works,
- roughly 150 renovation items and over 300 ten-yearly maintenance items,
- 350 sub-contracting firms,
- 5000 orders placed,
- over 4500 contractors who worked on the plant site,
- between 600 and 950 people on the site every working day,
- 3.5 millions hours worked on the power plant site.

The **radiation assessment** for the shut-down period was 0.31 man-Sv for about a thousand people (CEA, EDF and contractors). This raised the total dose at the Phénix plant since it first came on line to 1.8 man-Sv - equivalent to the annual collective dose from a pair of pressurised water reactors). The job giving the highest radiation dose was the inspection of the conical shell (0.10 man-Sv). As regards work safety, some sixty accidents giving rise to sick leave were recorded during the period. Accident frequency was average (13 accidents per million hours worked), and all in all the accidents were benign, as is confirmed by the fact that they gave rise to only 0.17 days' sick leave per thousand hours worked). This is an acceptable record when, for example, there were up to 200 people working simultaneously on different worksites closely spaced on every floor of the steam generator building.



Sub-contractors reception area following the mini-tornado of September 1999

On Sunday September 22, 1999, at about 10 pm, a mini-tornado ripped through the Phénix site. No one was injured and the industrial facility suffered no damage. But offices were devastated, prefabs in the contractors' reception zone were moved or bowled over, trees were uprooted and vehicles damaged. The human cost would clearly have been far more serious had the incident happened during working hours. Later, although the plant was untouched by the torrential rains and flooding of September 2002, some of the staff suffered major disasters at home.

6.8. The renovated reactor

This non-productive period was above all a time of intensive work on the plant, for the regular staff, the many reinforcements and everyone employed on the site by contractors. Increasingly efficient project structures were organised.

Considerable difficulties were overcome. These were mainly connected with the following:

- The late completion of the preliminary studies; so as not to delay the work, many operations were started before the studies were finished.



- The way the list of work to be done lengthened as the work progressed, owing to the Safety Authority's ever more stringent demands.
- Discovery of defects during overhaul of several equipment items and large components (including the intermediate heat exchangers and steam generators); this meant unscheduled replacements or additional repairs,
- Late delivery of some essential components, such as the devices for visual inspection of the core cover plug and the new intermediate heat exchangers.
- Worksite management required constant effort, especially for coordinating joint activity in the different buildings.

In spite of all this, there were some exceptional technical successes that deserve highlighting:

- Ultrasound inspection of the welds on the conical shell of the reactor block, conducted across a distance of 3 metres in sodium kept at a temperature of 155 °C, using the conical shell itself to guide the ultrasound waves. The robotic sensors were introduced into the 80-millimetre space between the main vessel and the double envelope of the reactor, with a 5-metre off-set.
- The visual inspection of the core cover plug, using inspection devices more than 19 metres long and an entirely optical process inside the reactor block. This involved removing 400 metric tons of primary sodium and then returning them to the primary circuit.
- The repair of the steam generator modules, replacing portions of sodium pipe, without damaging the steam tubes. An operation conducted within the scheduled time and budget thanks to the establishment of a suitable project structure.

- The simultaneous performance of quite a number of very varied tasks (civil engineering, mechanical engineering, boiler-making, non-destructive inspections, electricity, electronics etc.) in the steam generator building. This required good management of the interfaces, especially as people on different jobs often had divergent objectives.

All in all, the work proved that inspecting the internal structures of a sodium reactor, although difficult, is perfectly possible. The same applies to renovation a nuclear reactor in depth to meet new safety requirements, some of them very different from the original design.



Dominique PÊPE

*Head of the Renovation Project at the Phénix plant
from January 1996 to December 2000*

Renovating the Phénix plant

The renovation of the plant was a period of intense activity at Phénix, and also of considerable anxiety, first concerning how long the renovation would take (in the end it stretched over nine years, from December 1993 to February 2003, with a few months in operation during that time), and also the scale of the work, the design and execution difficulties involved, the unpleasant surprises and each successive recommendation of the Safety Authority.

Back in December 1993 it looked like a simple matter. In two years we would complete "*most of the studies and work required to continue running the plant for ten years*", i.e. an assessment of the damage to the reactor block structures, re-examination of the design of the secondary cooling circuits, a few jobs such as replacing some sections of secondary sodium pipe made of 321 steel, supplying two intermediate heat exchangers and a supplementary shut-down system, and "*a little corrective work to validate the seismic resistance of the steam generator building*".

But the project soon became more complicated as the results of the studies and inspections trickled in, until there were thirty separate subjects of study and about two hundred separate jobs of work to be done.

So great was the difficulty in some domains as regards design, calculations and performance of the work, and so innovative the solutions found, that it's fair to say the impossible was achieved. First of a kind operations were the visual inspection of the core cover plug inside the reactor vessel and the ultrasound inspection on the primary structures (required making openings in the safety vessel) and the repairs to the superheater and reheater modules. Other examples were the seismic strengthening of all the buildings, which meant harmonising the opinions of dozens of experts on the sizing rules to be applied, some of which had to be invented; the emergency cooling circuits, which had to make it possible to evacuate an almost instantaneous fivefold power increase in the event of a total loss of decay heat removal circuits, without any human intervention including in the event of earthquake or major sodium fire; and the partitioning in the sodium circuit steam generators and the anti whip systems of steam pipes, which necessitated several design changes and years of calculations. The fact



that the steam generators building combined all the hazards – earthquake, sodium fire and steam pipe failure – necessitated a host of calculations to ensure a coherent whole, as an altered parameter in one field affected the others.

For the same reasons, the actual renovation work was particularly delicate. Dozens of firms were working at the same time in small spaces with all the problems imaginable of cohabitation, co-ordination, safety, cleanliness etc. The state of the buildings and circuits sometimes caused doubts as to a successful outcome, with the presence of hundreds of tons of scaffolding and rubble and hundreds of metres of cables pulled out and hanging loose.

This period in the life of the Phénix plant greatly perturbed the staff. At first, during the years of study when nothing visible seemed to be happening on site, there was a certain impatience. Then came the sometimes difficult cohabitation with teams from the CEA's Military Applications department and the EDF's engineering centre in Lyons who came to strengthen the workforce on site but had different methods and work habits; staff felt they were being dispossessed of their workplace. Then, when the work was at its height, there was the discouragement of finding the plant each day more deteriorated rather than renovated, with lagging and small pipes deformed by the repeated passage of feet, cables ripped out, thermocouples mistaken for wire and cut up for use as makeshift binding. Then there was the belated discovery, for some, of the new equipment, the appropriation of the modified circuits, and the long and difficult resumption of power operation.

To cope with all this, organisation, which had initially been very simple, was gradually extended, refined and professionalised, with the introduction of high project-leading skills, under the leadership of successive plant directors (four in all, and as many CEA chairmen!) who skilfully guided the plant through those final years and the power build-up process.

In early June 2003, the arrival of the letter from the Safety Authority authorising the resumption of power operation for a final six irradiation cycles was experienced by many staff as a release from years of working under pressure – and with real and legitimate pride and satisfaction.



RESUMING
OPERATIONS
(2003 - 2009)



Landmarks

- 7 January 2003 Safety Authority authorises return to power operation
- 15 June 2003 Start of 51st irradiation cycle
- 1 September 2003 100,000th hour supplying EDF electricity grid
- 13 September 2003 Sodium-water reaction in steam generator N. 1
- 22 December 2003 51st irradiation cycle re-started

To be continued ...



Having repaired and upgraded the entire facility to such an extent, is it reasonable to think that the plant will be capable of re-operating at a steady power rate for a long period of time? This question was raised from the very beginning of upgrading, inspection and maintenance activities: a considerable programme including requalification tests, data updating, personnel training, etc. was conducted at the same time and in close collaboration with overhaul activities in preparation of the plant's start-up.

With overhaul activities accomplished, all circuits functioning properly and licences obtained (first semester 2003), the plant teams are once again occupied with reactor operations and operational problems - hopefully few and far between - and chiefly with maximum operations periods using two secondary cooling circuits. This time however, experimental irradiation tests have priority over power operations: this requires implementing and managing programmed shutdowns for refuelling activities and loading and unloading of experimental sub-assemblies.

Furthermore, science must adapt to available technologies as nuclear power plants must always respect strict operating regulations and answer to Safety Authorities. The number of inspections generally increases with the reactor's age to make sure that all equipment

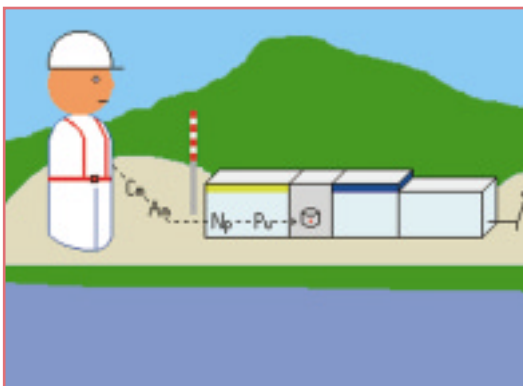
remains in good working order. If schedules are not optimised nor strictly respected, programmed shutdowns for maintenance can drag out and begin to resemble something of the ten-yearly outage performed at the beginning of Phénix's operations. Finally, it is extremely important that the last phase of a nuclear facility's life-span - its dismantling - also be well prepared.

Having discussed how the operator managed the start-up operation after overhaul activities, this chapter will examine the 51st irradiation cycle which is currently underway. The preparation of future reactor operations, core physics and irradiations will then be explained, before highlighting Phénix's potential contribution to the future Generation IV reactor project. Dismantling issues will be discussed in guise of a conclusion.

7.1. Preparatory work

Preparatory work carried out before beginning reactor power operation was proportional to the extent of overhaul work performed between 1999 and 2002 and was instigated very early on in the process. Early 2000, a specific organisation was implemented to define and carry out exhaustive **requalification tests** of all equipment and circuits related to shutdown activities. Each component affected by outage work was tested phase by phase during specific test programmes, as was case, for example, during the very first operational tests, only this time tests were limited owing to the volume of work to be done: it was not necessary to test components that were knowingly in good condition.

The facility was classified into different operational groups of equipment and components sufficiently alike to be analysed in an autonomous manner, with however the constant apprehension of making sure that components on the limits between these groups



were not dealt with twice nor forgotten: reactor, secondary cooling circuits, steam generators, fuel handling equipment, ventilation circuits, back-up cooling circuits, electricity generation facility, electrical installations, etc. Such an organisation was also implemented during the main operational periods such as criticality and secondary cooling circuit refilling following long drained periods. This organisational procedure, during which more than a thousand documents were produced, was concluded by the follow-up of the reactor power build-up.

During outage work, three quarters of the plant's operational **personnel** was replaced. A considerable number of highly specialised staff possessing a thorough knowledge of all plant items retired after having worked twenty years at the plant. Therefore, the reactor was started up again with a new team of operational engineers that had not worked on Phénix before. The reactor and the electricity generation facility began operating with a quarter of the personnel being new to Phénix and the other quarter having changed positions within the former teams prior to power operations. Thus, as the plant had operated on and off for a period of about ten years, it felt like the Phénix plant was going through a commissioning phase once again.

Personnel at the Phénix plant gradually reached a total of 270 people, owing to new legislation that brought the working week down to 35 hours. Engineers and technicians from specialised schools were hired in priority in order to build up a new "stock" of Phénix experts. Engineers from the Renovation project were assembled to form an engineering section in 2001 in an attempt to maintain a pool of expertise required in the medium term. A shortage in



The Simfonix simulator

office space soon arose due to both the increase in the overall number of plant staff and the fact that maintenance staff was mobilised for preparatory work. Several temporary buildings were set up around the plant, including the biggest building complex for the outage support team.

In terms of **competencies**, considerable effort was made to train staff, through on-the-job training, situational training and "apprentice coaching", as well as conducting specific knowledge transmission classes and practical training using the Simfonix reactor control simulator. Furthermore, long periods during which important reactor components were prepared for operation - such as secondary circuit refilling and reactor criticality activities - were used to "refresh" staff competencies and provide specific individual instruction necessary for new personnel. Therefore, numerous criticalities for neutron and physic tests and training shift teams were performed: all qualified and trainee reactor control operators performed at least one approach to criticality and one reactor criticality. Generally speaking, functional requalification tests were carried out by the plant staff, which helped personnel better assimilate the facility.

Staff training

Once the major renovation work on the Phénix plant had been completed, staff training took on particular importance, all the more so because most of the employees had never seen the reactor in operation before and the specificity of the reactor in the French nuclear series meant that help from outside was not possible. Under these conditions, it was necessary to develop in-house training for the staff in different domains.

The largest and most important area concerned operating the facility. The training program on the Simfonix simulator was intensified and additional training programs were organised on the shutdown circuits and the new or modified functions within the facility. Advantage was taken of the identity of the equipment to send machine operators to conventional EDF power plants of 250 MWe to allow them to complete their training on the electricity generation facility. The shift teams and security teams from the Marcoule centre did communal intervention exercises on simulated accidents or catastrophes. In 2002 alone, every operations employee thus did an average of ten days of training in addition to his regular program.

All the other employees in the power plant also receive training adapted to their function: security, running the reactor, maintenance management, nuclear safety, quality assurance, management of sub-contracted business and services, procedures in case of accident and so on. In order to compensate for even the slightest "in the field" training program for engineers, a training program on incident- and accident-related situations was developed on the simulator for the engineers on call. This special effort in staff training, which will be continued throughout the final years of operation, is one of the keys to the success of the resumption of power operation of the power plant.

It is nevertheless important to point out a fact that seems obvious for nuclear power plant operators but not for the general public: operational teams do not stop working when a plant is shut down, but rather see their workload increase in volume. Reactor control teams are constantly involved in reactor operations and also ensure that equipment is

available for upgrading and maintenance activities. The same applies to the maintenance teams who manage or carry out all maintenance and inspection operations. The plant personnel continues working as usual during shutdown, only the general context is different as the reactor is not operational: facility monitoring, particularly the monitoring of remote components, is reduced and the notion of urgency is less pressing.

In parallel, more than a thousand different reactor control documents were updated: operational notes, instructions (general, specific, incidental and accidental), periodic tests and locking procedures etc. In terms of maintenance-related documents, the reception, validation and classification of end-of-operation files was carried out in priority to be sure that all necessary data had been collected. Periodic maintenance worksheets - over a thousand documents - were updated at a later date unless required beforehand. A specific software programme was developed to help identify documents to be modified, as well as prove the exhaustive manner in which essential safety documents are updated.

The Permanent Group for nuclear reactors held three more meetings to discuss Phénix reactor **safety** matters. On July 12 and September 6, 2001, steam generator repairs were evaluated. The Permanent Group also remarked the quality of results obtained during the inspection of the conical shell and the core cover plug. The third meeting was held on October 31, 2002. Repairs made to the steam generator modules, inspection of steam generator welds that had not been repaired and the related in-service monitoring programme were all deemed satisfactory, as were the studies concerning the strength of the steam generator casing under accidental and "beyond-design basis" conditions. From a safety viewpoint, the reactor was deemed





The turbo-generator set

capable of completing the remaining six irradiation cycles.

Even now, the volume of files referred to the Safety Authority is phenomenal as more than 60 files are drawn up each year. During the four-year shutdown period for renovation, inspection and maintenance work, the Safety Authority carried out over thirty inspection visits, most of which focused on assessing the quality of reactor operations, as well as covering more general matters such as fires, periodic tests and the Phénix plant Internal Emergency Plan, etc. Around half of these inspection visits were successfully passed, which highlights the overall good working order of the plant and the quality of work carried out by the personnel.

As part of the strategy to improve the plant's "safety culture" among staff, top management and plant organisation, an audit was carried out to evaluate the non-conformities between real work practices and safety objectives. Assisted by EDF's Nuclear Inspection Authority, this audit was carried out by ten auditors and peers on site at Phénix during the first three weeks of the year 2001. On the last day of the audit, observations and recommendations were presented to the plant's management staff and the audit report was handed over. Conclusions were based on concrete observations and all positive aspects from which progress could be made were pointed out and

problems were underlined. Recommendations and suggestions were drawn from these conclusions, which were implemented in the following months and integrated into other plant activities. The plant shutdown period was concluded with an overall cleaning programme - removal of remaining worksite bits and pieces, painting, sign posting, etc.

7.2. 51st irradiation cycle

As was the case during all the other plant shutdown periods, the fuel remained in the reactor during outage work. The fuel sub-assemblies required to operate the reactor during the remaining irradiation cycles were delivered between December 1999 and March 2002. Two handling cycles were performed in June 2001 and early 2003 to refuel the core and load the new experimental sub-assemblies.

The secondary circuits were filled, heated and drained several times until the sodium was deemed pure, as it reacts with traces of oxygen remaining behind in the pipes that have to be opened for maintenance and repairs. The electricity generation facility - excluding the turbo-generator set - was tested at high temperatures by using steam from the steam supply facility on site at Marcoule.

Two **sodium leaks** occurred during this period. On March 16, 2003, a valve bellow in the sodium purification circuit N. 1 of the secondary circuit was no longer leaktight. The sodium leaked through the bellow and rapidly found its way into a space between the valve shaft and the packing. Approximately twenty litres leaked out of the purification circuit which provoked a small sodium fire. The aerosols generated during the sodium fire spread into the partitioning of the secondary circuit N. 1 and even leaked into the secondary circuit N. 3, not to mention the area where the water and steam pipes are located.



Valve with insulation removed after a sodium leak

The leaktightness defect in the packing seemed to be a generic problem with all bellow valves, therefore approximately fifty valves had to be upgraded. Furthermore, a thorough cleaning of the regions into which sodium aerosols had spread was performed to prevent them from corroding equipment, especially smoke detection equipment and electrical connectors. This incident revealed a few defects in the steam generator building's partitioning (cf. § 6.3). Even though this partitioning must not be completely leak-tight so that hot gases can be evacuated in the event of a serious sodium fire, it must nevertheless help delay the transfer of sodium aerosols generated from the affected area into the circuit in operation. This delay is used to identify with certainty the faulty circuit and avoid draining the wrong circuit. A systematic research of smoke transfer possibilities from one area to another was carried out and specific improvements to the partitioning were made.

A blocked door...

Following a leak in the metal leaktightness bellow of a sodium flow regulating valve in the n.1 secondary purification circuit on Sunday 16 March 2003, the valve was rapidly dismantled and replaced with a new, but identical, valve. In the workshop, the dismantling operations of the cable stuffing box (made of asbestos) and reassembly with a new cable compatible with the sodium are validated. A leaktightness test was also validated (with a maximum hypothetical leakage rate of one kilogram of sodium per hour).

The replacement of this cable had to be performed on the 46 bellow valves in the power plant. The cables chosen were manufactured by a British company but the diligence of the teams was counteracted by... a blocked door on the hold of the aircraft transporting them! After several fruitless attempts in several different airports (Geneva, Marseilles, Metz), the hold was finally opened by an expert. The delivery delay was around one week.

The assembly was completed. To validate it, leaktightness tests for the space between the metal bellows and the packing situated downstream (secondary leaktightness) were conducted by controlling the evolution of argon pressure injected into the space in question, by disassembling the spark plug type sodium leak detector. In addition, a sample of twelve valves were X-rayed to check the quality of their bellows. The operation was finally completed on April 18.

On May 4, 2003, another sodium leak appeared in the electromagnetic pump of the steam generator N. 3 hydrogen detection circuit. The sodium remained confined inside the pump casing. The pump was consequently replaced.

Having issued the **operational licence** in January 2003 for the remaining six irradiation cycles, the Safety Authority then authorised reactor power operations on June 5, 2003 after having finished all assessments and visited the plant. Despite all these precautions, problems continued to arise in the plant. The sodium leak detection beaded wires in the

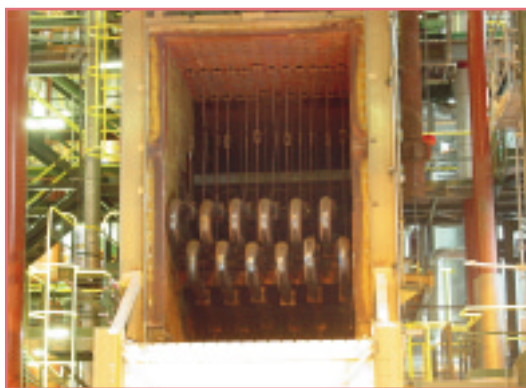


secondary circuits, some electrical equipment, water and steam circuit equipment, and various other components had to be inspected by the plant technicians due to successive failures as feared. The reactor diverged on June 15, the turbo-generator set was connected to the electric grid on July 4 and the plant attained its maximum authorised power of 350 MWt on July 6 even though power operations had to be interrupted several times. In August, a programmed shutdown made it possible to complete various different overhaul tasks and correct problems that had occurred during power operations. The turbine clocked up its 100,000th hour connected to the electric grid on September 1, 2003.

However, the plant incurred a fifth **sodium-water reaction** just when the personnel had begun hoping Phénix would continue along a path of steady power operation. On Saturday 6 September, 2003, the hydrogen detection monitors for all three levels of the steam generator N. 1 gradually began rising in a linear manner. Reactor power was reduced to about 40% of its rated power as to avoid exceeding threshold limits provoking the automatic shutdown of the reactor. The following week was devoted to investigating the reactor in an attempt to pinpoint the problem. On the morning of September 13, the hydrogen detection measurements in the reheater of the steam generator N. 1 rose suddenly and rapidly, provoking a moderate sodium-water reaction. The hydrogen detection circuit automatically triggered the reactor shutdown, as well as the rapid isolation and depressurisation of the steam generator N. 1. The rapidity with which these actions were accomplished helped limit the water mass that came into contact with the sodium to approximately 2 kilograms. All the detection and automatic safety circuits functioned

correctly. According to safety instructions, the shift team dried out the steam generator N. 3, drained the secondary cooling circuit N.1 of its sodium and controlled the removal of residual power from the reactor via the secondary circuit N. 3.

The faulty module was quickly identified as module N.12 in the reheater. This module was then removed and the pierced tube was assessed by metallurgists at Saclay. Except for the weld responsible for the leak, all other welds were in perfect condition. The cracking was probably caused by an initial manufacturing defect. Several millimetres of wastage in the shell across the hole was also remarked: considering the leak time, these values confirmed calculation estimations obtained for this type of phenomenon. Furthermore, analysis of past operations did not highlight any possible transient that could have caused such damage to the reheater modules.



Open superheater of a steam generator

All results were submitted to the Safety Authority for approval. In parallel, the module was replaced with a new module which was inspected and pressure-tested prior to installation. All plant teams were highly effective and demonstrated great dynamism, both during operations on the facility and when it came to writing up reports.



Permission to increase power was issued on November 6, 2003. The sodium in the secondary circuit N.1 revealed to be rather polluted caused by air that entered the circuit while the module was being replaced. It took the entire month of November to purify the sodium coolant. On December 1, a lightning bolt hit a transformer and the Rhône River's one hundred year flood waters (10,000 m³/s) delayed the delivery of spare parts for several days after this incident.

On December 11, a turbine trip accidentally triggered the main line breaker opening in the switchyard, which left the plant without its main offsite power supply (220 kV power line). Automatically, the reactor was shut down, the power supply was diverted to a 20 kV auxiliary line to supply the auxiliary equipment needed to ensure the cooling of the reactor and the diesel-powered generator sets were started up in case the power supply could not be diverted. Tests revealed that a dysfunctional relay was the cause of the problem. This incident is worth pointing out as a) the facility underwent a significant transient - the fifth transient since its commissioning, remembering that the plant was built to resist ten transients - and b) it proved to be very difficult in locating the cause of the problem (non-repeatable).

The plant was connected to the electric grid once again on December 22 and has been operating since then at the maximum authorised power of 350 MWt without any particular problems, generating approximately 145 MWe. After several ups and downs, the plant staff was relieved to see the reactor operating correctly and gradually regained confidence in their reactor. The smooth running of the reactor was considered a reward for several years of hard work.

7.3. Operational programme

An operational programme of six irradiation cycles each representing 120 equivalent full power days (EFPD) was approved. The reactor is currently operating at two thirds of its rated power - 62% precisely - and the secondary circuit N. 2 is not running. The period of time during which the plant is in operation in future greatly depends on its availability, that is to say, on the frequency of unforeseeable complications occurring during power operations and the number of shutdowns for refuelling, maintenance and inspection. Based on an average hypothetical availability equivalent to 70%^[1], each irradiation cycle should last nine and a half months, including short shutdowns (two weeks) required to reshuffle sub-assemblies during the cycle.

At the end of each irradiation cycle, it is necessary to **reload** approximately one sixth of the sub-assemblies, including the relocation of sub-assemblies within the reactor core in relation to their burn-up rate. Sometimes, even the replacement of a control rod or its guide tube is required. It takes a little under three weeks to complete all these operations, taking

Exporting

The Japanese from the JNC O-arai centre wanted to study the ageing of the materials in a fast neutron reactor cooled with sodium. It was for this reason that, after a certain amount of discussion, the Phénix plant sent them two sections of piping, approximately one meter fifty in length each. The first section came from a main pipe (diameter 500 mm) in a secondary cooling system, with some 100,000 hours' service on its clock. The second was a control of a weld on a new sleeve on a pipe of the same diameter. This type of welded joint is something in which the Phénix plant has considerable experience.

[1] This degree of availability - shutdowns excluded - is based on the availability factor obtained during the previous operational cycles.



Top management staff

	<i>Reactor Construction Department Managers, Phénix Project Managers</i>	<i>Phénix Project Deputy Managers</i>
1968	Rémy CARLE	Jean-Claude MOREAU
1971	Deputy Manager: Jean MÉGY	
1975	Jean MÉGY	Francis LICHTENBERGER ^[1] Edmond ROBERT

[1] died in August 1971

	<i>Plant Managers</i>	<i>Plant Deputy Managers</i>
1970	Fernand CONTE	Jean-Louis GODDET
1975		Michel BROSSON
1980		Bernard GIRAUD
1985	Michel GELÉE	Michel BOURRIÈRES
1990		
1995	Xavier ÉLIE	Alain PAUL
2000	Alain LAFON	Jean-François SAUVAGE
	Jean-François SAUVAGE	Patrick MARITEAU
	Joël GUIDEZ	



into consideration the time needed a) for the reactor to cool down to 250°C, b) to start-up and put away handling equipment in a sodium environment and c) to requalify specific functions, especially the control rods.

Furthermore, regulations vis-à-vis pressurised equipment stipulate the need to perform shutdowns for inspection and maintenance at least every 18 months. It was decided to programme these shutdowns independently from reactor operations on a calendar basis. This strategy makes it possible to fix the date for such shutdowns, which simplifies matters when it comes to ordering parts from Phénix's main companies, as well as extending the deadline for preparatory work with the intention of optimising such work and work interfaces. It is true that more and more work - from facility operations to project management - is being subcontracted to outside companies. It has therefore become necessary to take into account their workload on other facilities, site access procedures and deadlines for placing contracts, etc. The critical path for such shutdowns must take into account the time needed to a) drain the secondary circuit N. 2 (removal of residual power), b) carry out non-destructive tests on the secondary cooling circuits and steam generators and c) restart these circuits. The duration of each shutdown for inspection and maintenance has been estimated at two and a half months.

Of course, considerable effort is made to **optimise shutdowns** each time the reactor enters an operational phase. It is always preferable to synchronise refuelling periods with inspection and maintenance periods. To this can be added the importance of optimising fuel by making sure that "unburnt" fuel sub-assemblies are not unloaded prematurely so that all six irradi-



ation cycles can be completed correctly and the remaining stock of available fresh fuel sub-assemblies can be used. The duration of the irradiation cycle also determines the life-span of the experimental sub-assemblies. A minimum duration is necessary to obtain data on irradiation activities and a maximum duration was also set according to safety study recommendations. Nevertheless, adjustments are possible in order to avoid shutting down the reactor just to remove experimental sub-assemblies from the core.

Moreover, it is important to keep the duration of each shutdown to a strict minimum. In order to prepare and optimise the organisation of the future shutdowns, the plant held an observation mission at the Cruas Nuclear Power Plant in 2002. This mission aimed at understanding the organisation of an EDF nuclear power plant unit shutdown and how such a shutdown is managed successfully in order to transpose these conditions to the Phénix plant. Therefore, a new organisation was partially implemented during the unexpected shutdown following the sodium-water reaction in September 2003. This new organisation was then fully implemented during the first shutdown required to rearrange the core sub-assemblies in February 2004, as well as to help prepare the shutdown for maintenance and inspection in the summer of 2004.

The **irradiated fuel-assemblies** accumulated in the storage drum since 1997 had not been dismantled due to the lack of storage outlets. Dismantling and removal activities were resumed in July 2003 seeing that the spent fuel pool at the Marcoule Pilot Facility which had stopped reprocessing activities was once again able to store fuel pins containers. These





The Japanese Monju reactor

fuel pins shall be stored here until they can be transferred to the La Hague plant for reprocessing.

Sub-assembly steel elements such as spikes, hexagonal wrapper scraps and upper neutron shielding were placed in waste bins and are currently stored in storage pits N. 1 and 2 on site at Marcoule. Storage space is still available in the storage pit N. 2 but will not be used upon request from the Safety Authority, owing to the fact that these pits make it difficult to recuperate waste once placed. The storage pit N. 3, based on a modern concept, will be operational from 2005. Until then, waste bins are also stored in the Marcoule Pilot Facility.

Last of all, the storage drum now contains about twenty special sub-assemblies for which special measures must be taken in terms of dismantling, not to mention the fact that a storage management strategy must also be defined.

7.4. Core physics

The core, the turbo-generator set, the intermediate heat exchangers and the steam generators are all necessary components ensuring the transformation of heat into electricity. The core, as always, is the vital organ enabling all the other components to work. If all the previous chapters have said little on the reactor core, it is because the Phénix core has functioned perfectly well over the past 50 irradiation cycles performed to date.

Both the reactor designers and plant operators have demonstrated great care, the former in terms of recommendation and the latter in terms of how the machine should be operated, which explains the fact that the core has cost the plant very little in shutdowns, excluding the scrams due to negative reactivity trips for which the cause - core, structures, instrumentation, etc. - has yet to be found. At the same time, the core performance has constantly improved. More specifically, the increase in the specific burn-up of fuel sub-assemblies, from 50,000 MWd/t to 90,000 MWd/t at the core centre and 115,000 MWd/j on the periphery made it possible to double the time these fuel sub-assemblies spend in the core. However, core behaviour has varied little since the beginning, despite the various different experiments conducted in the reactor: parameters such as the negative reactivity of the control rods, the effects of the thermal reactivity feedback effect and the loss of reactivity due to spent fuel have remained extremely stable since the very first irradiations cycles.

Core follow-up during irradiation cycles focuses on both thermics, based on the constant measurement of the sodium temperature at the outlet of each sub-assembly, and reactivity which is controlled by raising or lowering the control rods in the core.

Temperature measurements at fuel sub-assembly outlets are taken into account by two computers that rapidly process all core temperature data (Fast Temperature Monitoring Circuit). Other than for safety measures, these measurements are used to assess the overall situation in terms of thermics that determines the heat, power and temperature of fuel assembly cladding. This follow-up also helps detect gradual shifts in temperature and provides information on the behaviour of fuel subjected to irradiation. For example, when a fuel pin bundle deforms, the corresponding reduction





The control room corridor

in flow rates leads to an increase in the heat in the fuel sub-assembly in comparison to expected sub-assembly temperatures.

Such measurements revealed a second swelling phenomenon in the solution heat treated 316 stainless steel or the lengthening and deformation of the cold-worked 316 stainless steel spacer wires welded to the Ti_6 316 cladding. Such phenomena were then characterised by post irradiation examinations performed on the equivalent sub-assemblies.

Differences between reactivity computer calculations and actual measurements remain reasonable, the only significant differences being caused by variations in the negative reactivity of one or several control rods in the vicinity of the experimental sub-assemblies. Thanks to numerous tests, the Phénix plant has greatly contributed to knowledge of reactivity feedback effects and helped establish new reactivity testing methods for control rods.

In terms of **core management**, the Caphé code was used from the very moment the Phénix plant was started to solve all neutron, thermal and hydraulic core calculations. This code was designed to provide the characteristics of each sub-assembly during the irradiation cycle underway: power, sodium flow, rated power, cladding hot spot temperatures, maximum linear power rating, burn-ups, fluence, specific burn-ups, etc. The Caphé code

was also developed to assess the reactivity of subsequent loads and therefore check whether safety criteria were respected and enough reactivity remained to eventually compensate for a loss of reactivity in spent fuel. This code was rapidly installed on office computers seeing that the data to be entered was very basic: the code used comparisons with the core load - whose flux distribution is well known - established at the very beginning of operations.

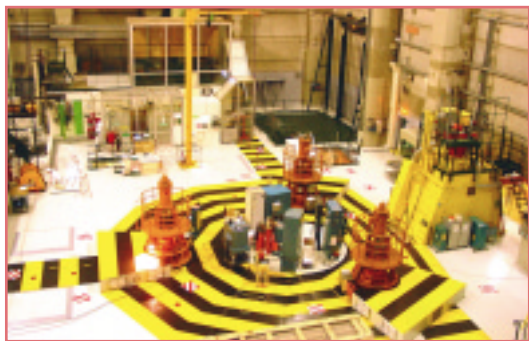
Over time however, the Caphé code became obsolete. Its field of application was undermined by progress made in fuel sub-assembly performance levels. The code was no longer able to correctly take into account the new core management strategies programmed for the end of the 1980s - high burn-ups, axially heterogeneous core concept for sub-assemblies, elimination of radial breeder blankets etc. Modifications that have been made since then - introduction of a complementary control rod in a central position, the gradual move to "small-core" configurations, insertion of sub-assemblies containing moderators into fissile regions - led to the decision to substitute the Caphé code with a more appropriate tool.

Consequently, the Gephix computer code became the **new core management tool** with the beginning of the 50th irradiation cycle. The tool was developed by the Reactor Studies

Phénix: a new given name ?

From 1991, staff exchanges were set up between the Phénix plant and the Japanese power plants Joyo and Monju. Japanese engineers thus arrived in a steady stream for missions of one to two years. They brought a touch of exoticism to the power plant, as well as good manners and considerable enthusiasm, both in professional terms and as regards the French way of living. To prove this, one of the engineers who was lucky enough to become a father during his stay in Bagnols sur Cèze, chose Phénix for the fourth name of his child.





The hall of the reactor building

Department at Cadarache and was a result of the Eranos fast neutron reactor core neutron calculation system. Therefore, this code takes into account recent and more effective models and is specifically developed using:

- a 3D representation of the core with hexagonal geometry,
- precise modelling of each fissile and breeder sub-assembly,
- direct calculations of the neutron flux distribution in the core.

The Gephix code is designed to access the main neutron, hydraulic and thermohydraulic characteristics of the reactor based on a set of standard data of the reactor (geometrical data, nuclear cross sections, flow laws, etc.). This code manages each fuel sub-assembly individually (breeder and absorber sub-assemblies, standard and non-standard sub-assemblies) and enables:

- the follow-up in real time of the core load being irradiated in the reactor,
- the projected follow-up of the current core load by anticipating the time variation curve related to the load,
- research into reactor configurations focusing on defining the characteristics of future loads,
- archiving loads that haven't been irradiated but on which tests deemed interesting were performed (specific criticality, reactivity testing, etc.).

7.5. Transmutation

The French law dated December 30, 1991, more commonly known as the "Bataille" law (cf. § 5.5) is partly responsible for a vast research programme focusing on the separation and incineration or transmutation of minor actinides and long-lived fission products. This programme aims at developing alternative solutions to deep geological waste disposal methods after having embedded such waste in highly resistant glass - a storage solution that was developed in the 80s.

The transmutation of long-lived radioactive isotopes helps reduce their toxicity to a level that is similar to the toxicity of natural uranium in a much shorter period of time than through the natural radioactive decay process (approximately 10,000 years instead of one million years). The transmutation of minor actinides - americium, neptunium and curium - and long-lived fission products has proven to be much more efficient in fast neutron reactors than in thermal neutron reactors such as water reactors (lower production of heavy isotopes, greater neutron fluxes and the presence of surplus neutrons, etc.).

From this point of view, the Phénix plant is aiming to demonstrate the technical feasibility of using this type of nuclear waste management technique. The Phénix plant hopes to lead the way in radioactive waste destruction techniques: various other research programmes will be launched to optimise this approach - if approved by the Safety Authorities - particularly through the application of advanced reactor concepts (cf. § 7.6).

Several experiments are being conducted to cover the different aspects of the programme:

- the incineration of minor actinides using a heterogeneous recycling mode, which involves irradiating a target containing a single minor actinide such as americium for example,

- the incineration of minor actinides using a homogeneous recycling mode, which involves irradiating minor actinides in small quantities with the fuel,
- the transmutation of long-lived fission products, which helps reduce their radiotoxicity while transmuting them into stable elements or elements with short half-lives.



Ecrix experimental rig

The **Ecrix B and H** experiments are being developed to test the feasibility of incinerating americium using a heterogeneous recycling mode. Two identical experimental fuel pins are inserted into the reactor and subjected to different irradiation conditions. Each fuel pin contains a stack of fuel pellets composed of fine particles of americium oxide evenly distributed in an inert magnesia matrix. In order to increase the efficiency of the transmutation process, these experiments are performed in a device capable of locally moderating the neutron flux by slowing down neutrons. It is therefore possible to take advantage of the great flux of neutrons available in the core by combining the high thermal neutron cross sections, which help obtain a fission rate of 30%, associated with a transmutation rate over 80%. The Ecrix B experimental device is inserted in the center of the core with a boron moderation, whereas the Ecrix H experimental device is located on the core periphery and neutrons are slowed down using calcium hydride.

Atalante and the Phénix plant

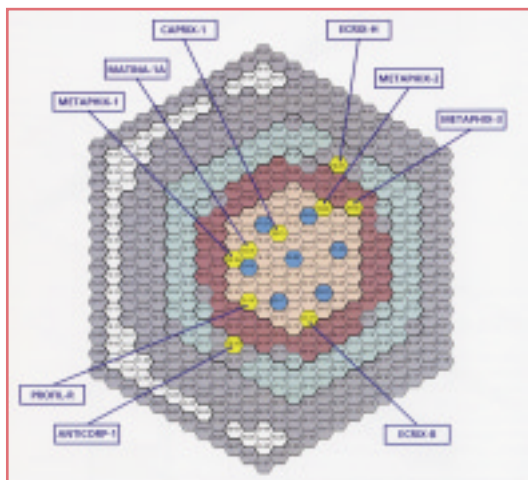
In operation since November 1992, Atalante groups together in a single facility all the research means in high activity chemistry or nuclear chemistry necessary for studying downstream of the nuclear power cycle. Atalante has more than 200 researchers, engineers and technicians, enjoying high performance, modern investigation means, with the latest laboratory becoming operational in December 2003. Atalante makes it possible to conduct both fundamental and applied research from the laboratory study stage (using very small quantities of radioactive material) to demonstration experiments using large quantities of samples of spent fuel. This gives the centre an exceptional position in the world nuclear research arena, and makes it a strong point in the construction of the European research area.

The research conducted in the context of the law of 30 December 1991 deals in particular with:

- developing processes for sorting long lived radioactive elements (essentially minor actinides),
- preparing and characterising "targets" for actinides separated in advance and destined to be used in transmutation studies (reduction of the life expectancy of waste matter, even transformation of such matter into stable elements),
- defining, drawing up and characterising the housing matrices of actinides and fission products (glass, ceramic and vitroc ceramic),
- studying the long term behaviour of the different matrices imaginable, with the aim of making possible deep geological storage or long term storage.

The Atalante facilities were used in particular for the Ecrix program. Sixty grams of americium oxide were purified in the C9 shielded cells. The pellets produced were then manufactured and sintered in the L6 laboratory. Finally, the cladding of these pellets in rigs, which were then welded, was performed in a helium atmosphere in laboratory L7. These operations required the development of specific techniques to make it possible in particular to control the phase changes in the americium oxide when it was heated. In the very near future, Atalante will also take part in the Futurix program.





Location of the experiments for the 51st irradiation cycle

Along the same lines, the **Camix-Cochix** experiment is developed to test the behaviour of optimised targets under similar conditions. Variations will focus on the microstructures of particle dispersion in the inert matrix and on stabilising americium oxide.

The **Anticorp 1** experiment is developed to demonstrate the feasibility of transmuting long-lived fission products into stable elements. Three fuel pins each containing two ingots of technetium-99 in its metal form are irradiated with a moderate flux until a transmutation rate of 20% is reached to produce stable ruthenium-100.

The **Profil R** experiment is performed on a practically standard fuel sub-assembly to obtain basic neutron data required for transmutation-related calculations. Two experimental fuel pins are placed among the rest of the bundle fuel pins. Each pin contains about fifty canisters holding several milligrams of minor actinide isotopes or fission products under investigation. Likewise, the Profil M experiment will be devoted to irradiating these isotopes with a slightly moderate spectrum.

Metaphix is a rare earth and minor actinide (neptunium, americium and curium) transmutation experiment using a homogeneous recycling method. Three rigs, each containing three experimental fuel pins, are placed in the reactor for different irradiation periods. This experiment is carried out in collaboration with Japan and has the specificity of using a metallic fuel composed of a uranium-plutonium-zirconium alloy similar to that used or envisaged in some American fast neutron reactors.

Last of all, the **Matina 2-3** experiment is conducted to complete results on the behaviour of new inert matrices and target concepts designed for minor actinide incineration using a heterogeneous recycling method.

Most of these experiments, as well as ten or so extra experiments related to other programmes, have been loaded in the reactor during the 51st irradiation cycle and will be unloaded during the next refuelling shutdown according to programmed irradiation rates. Several experiments (Profil M, Camix - Cochix) are currently being developed and will be irradiated from 2005.

7.6. Generation IV nuclear power reactors

Since the beginning of 2000, the CEA has clearly illustrated its commitment to developing the nuclear systems of the future, from 2030 and beyond. Considering the international consensus not only on the sustainable use of nuclear energy but also on the scope of the scientific and technological investments to be approved, this approach has become the subject of a large international cooperation. This became possible after the United States recognised the importance of the issue following the decision on behalf of the American Department of Energy to launch the Generation IV International Forum. This international "think-tank" now regroups ten coun-

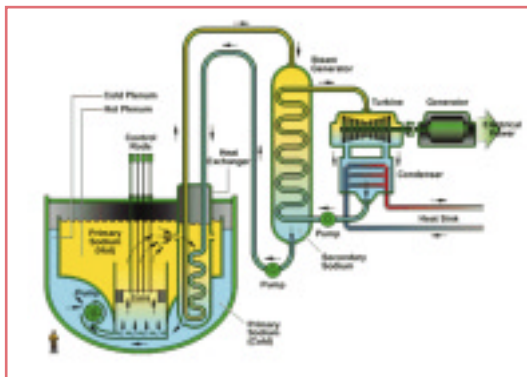


Diagram of Sodium Fast Reactor project

tries: South Africa, Argentina, Brazil, Canada, South Korea, the United States, France, Japan, Great Britain and Switzerland.

This international committee is devoted to defining future energy solutions and approving technological innovations to go beyond current limitations in existing or planned nuclear reactors by reinforcing the sustainable aspect of nuclear energy by: a) optimising fossil fuel resources via the use of fertile materials such as uranium-238 and even thorium, b) reinforcing safety aspects, c) reducing ultimate waste, both in volume, in toxicity and in half-life, d) following a stricter approach to nuclear proliferation risks, e) improving thermodynamic efficiencies by recovering thermal energy at higher temperatures and f) developing new fields of applications other than electricity production (hydrogen or soft water production). The reactors that have been dubbed the fourth generation should, for the most part, resort to fission produced by fast neutrons, with a liquid coolant such as helium (or liquid sodium) for example, capable of removing heat at temperatures higher than the 320°C of current water reactors.

Six technological options were selected in 2002:

- GFR (Gas-cooled Fast Reactor): a fast neutron reactor using a helium gas coolant,

- LFR (Lead-cooled Fast Reactor): a fast neutron reactor using a lead or lead-bismuth coolant,
- MSR (Molten Salt Reactor): a reactor whose fuel is composed of molten salt (fluorides),
- SFR (Sodium-cooled Fast Reactor): a fast neutron reactor with a sodium coolant,
- SCWR (Super Critical Water-cooled Reactor): a reactor using a light water coolant and operated under temperature and pressure thermodynamic conditions surpassing the critical point of water (374 °C, 221 bar),
- VHTR (Very High Temperature Reactor): a reactor with a helium gas coolant and operating at very high temperatures (1,000 to 1,200 °C).

Three of the above-mentioned systems are in fact reactors designed to operate with fast neutrons. The absence of a moderator reduces the loss of neutrons through captures, which has the double advantage of producing less waste while enabling a higher conversion rate of breeder material (uranium-238) into fissile material (plutonium-239). The flux of fast neutrons in the core also transmutes long-lived waste extremely well, regardless of how the waste was produced - waste generated on site or resulting from previous reactor generations and having been mixed with the fuel.

Furthermore, future reactors will both be more economical in terms of natural resources (consumption of uranium-238) and cleaner (recycling of long-lived waste). By using coolants permitting higher temperatures than water, it is possible to obtain better thermodynamic efficiencies and even create new uses such as the production of hydrogen via thermochemical processes or cogeneration of electricity and heat.



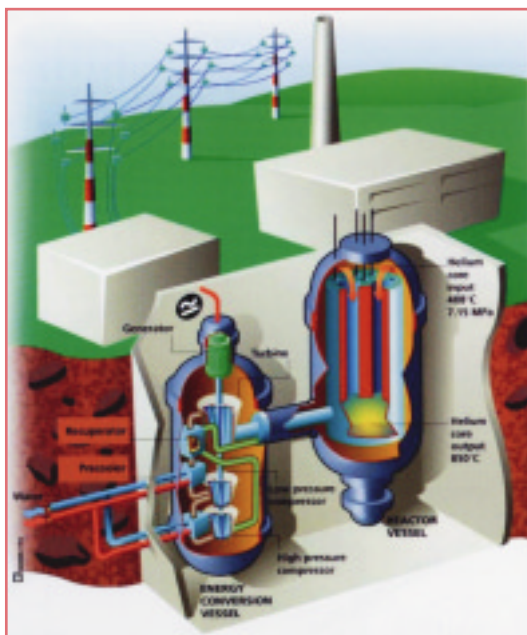


Diagram of Gas Fast Reactor project

Designed with cost effectiveness, safety and anti-proliferation factors in mind, these reactors consolidate the sustainable characteristics of the nuclear energy by fission.

The first phase of selection is then followed by a second phase that involves defining the terms of the cooperative process and allocate research programmes to the different partners for the next five to ten years. The CEA is focusing on gas coolant systems such as GFRs and VHTRs, while maintaining its expertise in the field of sodium-cooled fast neutron reactors (SFR). Thus, Phénix has been attributed a new objective. Though the materials and concepts to be implemented over the next ten years will certainly be different from those used in the past, analysing the behaviour of materials and concepts under real conditions and over a long period of time provides precious and irreplaceable knowledge.

In terms of using **sodium** as a coolant, several fundamental points concerning Phénix's operating experience are worth underlining.

First of all, the choice of metals used - 321 stainless steel is ruled out of course - must be validated through accelerated ageing tests of samples that have been subjected to stress representative of real conditions present during fabrication.

In-service inspection must be taken into consideration in the design phase so that structural ageing can be controlled as required. However, the greatest possible amount of prerequisite studies must be performed beforehand, so as to limit such in-service inspection to the strict minimum, for it is very costly in terms of resources and reactor availability.

Last of all, it is worth re-examining the interposition of the secondary sodium cooling circuits between the core and the steam generators. If safety measures concerning these circuits and steam generators are as strict as they are when carrying radioactive sodium, the interposition of such equipment becomes superfluous. Nevertheless, such circuits are very useful for the operator in avoiding the accumulation of radiation protection and sodium-related protection problems in the buildings.

Furthermore, three new irradiation experiments called **Futurix** have been developed to study the behaviour of fuel for future nuclear power system - gas cooled reactors, hybrid systems - and are programmed for the Phénix reactor. These experiments fall within the framework of an international collaboration with the Americans, Japanese and the European Union. The first experiment (Futurix FTA) will evaluate the behaviour of the different fuel types dedicated to the transmutation of high-content actinides, which will either be implemented in critical power reactors or in hybrid systems. The eight types of fuel will each be placed in a different fuel pin and

placed in one unique irradiation rig. Each partner will fabricate a fuel type in their own country. The Karlsruhe TransUranian Institute will be in charge of filling the fuel pins with the different fuels, whereas the final rig assembly will take place in the Phénix plant cell.

The aim of the Futurix MI experiment is to obtain a database on the behaviour of several inert materials under irradiation that are likely to be compatible to be used in the composition of core components of gas-cooled reactors. It is important to conduct irradiation experiments in the Phénix core to evaluate the possibility of reaching high temperatures (800 to 1,000 °C) and irradiation doses that are representative of gas-cooled reactor operational

conditions. The materials to be irradiated are inert (free of fissile material) and have been identified as potential candidates in the composition of structural materials and fuel elements for future gas-cooled reactors (carbide-type ceramics, nitrides, refractory metals, etc.).

Last of all, the third branch of the Futurix programme involves studying the behaviour of the different fuel designs that may be retained in the design of future reactors cores. The designs to be tested are currently being defined. All irradiation experiments have been programmed for 2006.

7.7. Dismantling preparation

Though the Phénix plant will finish its programme around 2009, much work still remains to be done after this date. Contrary to certain industries that abandon their industrial sites at the end of their activities, all plant life-cycles in the nuclear industry end with a comprehensive process of cleaning - known as cleansing - and dismantling. All the same, a leaktight containment would be largely sufficient in most cases to guarantee against the dispersion of the accumulated stock of radioactive products. However, for commendable ethical reasons, now labelled as sustainable development, both radioactive and dangerous chemical products are gradually removed from the facility. The facility is then dismantled and all waste and remaining products are transferred to waste storage centres or final waste repositories in relation to their level of toxicity.

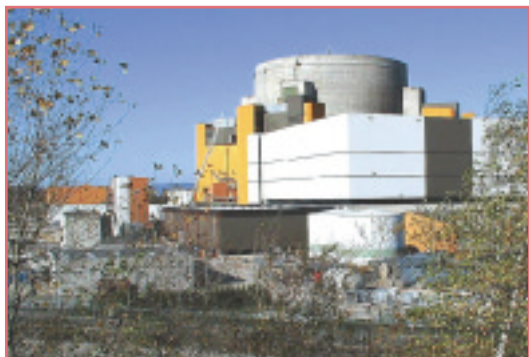
Such a large-scale operation must be carefully planned, and in the case of the Phénix plant, **studies** corresponding to this dismantling phase were launched as early as 2003 and have gradually been amplified so that dismantling operations can take place without delay as soon as the reactor enters its decommissioning phase. As soon as technical solutions

From Generation I to Generation IV

It is generally considered that nuclear power generation has already gone through two generations of reactor. The first corresponds to the prototypes built between 1950 and 1965 more or less (natural uranium gas graphite reactors in France). The second corresponds to the current commercial reactors (the pressurised water reactors in the EDF series). The third generation is for the reactors currently in the design stage, which take into account the feedback gained from the second generation reactors. The European Pressurised Reactor (E.P.R.) is an excellent example of this; it is a French-German project, and the first construction has just been ordered by the Finnish electricity company TVO (commissioning in 2009). The fourth generation is that of the systems of the future, those that are yet to be designed and which are liable to be built somewhere between 2030 and 2050.

The Phénix plant is the demonstration reactor for what was meant to be the French fast neutron reactor type from the 1990s onward. However, of the concepts for future reactors retained by the Generation IV forum, there is a fast reactor cooled with sodium. The performances required for fourth generation reactors (preserving natural resources, recycling waste products, high thermal efficiency and so on) are very similar to the advantages of the Phénix plant, as so proudly publicised by its promoters. From there to saying that the Phénix plant is the precursor for the fourth generation reactors, there is only a small step to make and some people are already more than willing to make it.





Dismantling the Creys-Malville plant

for storing waste are up and running, cleaning and dismantling activities may begin: the potential gain in radioactive decay of short-lived products, which simplifies operations after about ten years, does not however minimise the need to monitor the facility while waiting to safely handle radioactive waste. Furthermore, it is important to take advantage of the skills acquired by the plant teams so all operations can be carried out rapidly with minimum risk. Last of all, as the Creys-Malville plant was closed prematurely, EDF was required to start dismantling operations without delay. Much to everyone's dismay, the situation was reversed and it is now the Phénix plant that will benefit from the experience acquired by the teams working on SuperPhénix as dismantling activities progress.

Before actually dismantling the Phénix plant, two important operations must be performed beforehand: all fuel sub-assemblies must be completely unloaded from the core and the sodium coolant must be removed from the primary and secondary cooling circuits. Unloading is a typical part of core refuelling activities and involves transferring fuel sub-assemblies in the reactor to the storage drum, before being transferred to the irradiated elements cell to be cut so that both the fuel pins and structural waste can be removed. The reactor is therefore emptied of its sub-assem-

blies stage by stage, including the steel sub-assemblies on the core periphery.

As Phénix uses a **sodium** coolant, special procedures must be implemented. From the viewpoint of those responsible for designing the plant, it had been envisaged to reuse this sodium - not at all altered by reactor operations - in a third of fourth fast neutron reactor that France might have decided to build. However, as this reactor type has been disqualified, other solutions must be found.

Currently, preferences point towards the destruction of the metallic sodium by chemically transforming it into a stable compound to be stocked in a nuclear waste storage site. A process transforming sodium into soda was developed by the CEA and applied to the sodium from the Rapsodie reactor. This same process has been used since 2003 on an industrial scale to destroy sodium from the British reactor PFR located in Scotland and will be

Marcoule: a huge dismantling worksite

With the G1, G2 and G3 reactors, the UP1 reprocessing factory, the Pilot Plant and more, the Marcoule site has, since 1958, been a key production site for the nuclear industry, particularly in terms of National Defense. These facilities have now been shutdown and are in the process of undergoing radioactive drainage and dismantling. The aim is to reach stage 2 deconstruction by 2030, as defined by the International Atomic Energy Agency (I.A.E.A.). In concrete terms, this means facilities free of all radiological constraints.

For the UP1 facility and its facilities alone, this represents more than two million hours of work, with remote controlled intervention arrangements, 5,000 tons of equipment and 20,000 tons of structural materials to package in low and very low activity waste packages, 1,500 high or very high activity drums of waste and so on.

All of which goes to show that, when the time comes, all the know-how needed will be on hand to dismantle Phénix.



used to process sodium from both the Creys-Malville and Phénix plants. Soda can either be integrated into concrete blocks that can be stored at the ANDRA site built for very low radioactive waste, which is the case for Creys-Malville sodium, or be transformed into salt (sodium sulphate for example) before being sent to a liquid effluent waste treatment plant on site at Marcoule.

Furthermore, all equipment (pipes, pumps, heat exchangers, valves, etc.) having been in contact with the sodium retain a thin sodium film on their exposed surfaces and sometimes even sodium deposits in retention areas. This sodium and any other remaining radioactive products can be eliminated through the washing and decontamination of components in the handling building, as has been done since the start-up of the plant. The most complicated operation concerns the primary cold trap. This trap can either be placed in a leaktight container to be treated as waste, or can undergo special treatment to separate the sodium, radioactive products (high activity waste) and steel sheets (low activity waste).

Even though the precise sequence of operations remains to be defined, the Phénix plant's **dis-****mantling** phase can be broadly described. First of all, it is possible to dismantle all the conventional plant facilities not necessitating any particular safety procedures during fuel sub-assembly unloading and removal operations. Facilities such as the turbine hall, as well as the steam generators and the secondary cooling circuits can thus be dismantled. In parallel, removable reactor components (primary coolant pumps, intermediate heat exchangers, control rod mechanisms, etc.) can also be removed, washed, decontaminated, dismantled and evacuated in the form of waste packages, as commonly seen during reactor operations. Curiously enough, this first phase of dismantling is also used to construct several new facil-

ities that are either necessary to continue operations (sodium treatment for example) or useful when it comes to replacing perennial functions for which operational equipment was oversized (demineralised water production, etc.).

The reactor block is the following main unit to be dismantled. Of course, this is the most complex and highly contaminated structure having been irradiated throughout the plant's entire operation. Despite the differences in structure design, size and activation levels, feedback from equivalent operations performed on the SuperPhénix reactor will be very useful. It is interesting to point out that the Phénix plant will make its last contribution to nuclear technologies during this dismantling phase. Observation of the mechanical and metallurgic structures and materials subjected to great stress over the past thirty years of reactor operation will provide precious information for the design of future reactors using a sodium coolant or other high temperature coolants.

Once this phase has been completed, the remaining plant equipment and buildings can be dismantled and demolished just like any other industrial site, not to forget that the amount of nuclear facility dismantling experience already accrued is considerable. Nowadays, it is possible to completely dismantle the Phénix plant in approximately fifteen years.



And after dismantling?



7.8. Phénix's future

Despite the anxiety that many felt faced with the overwhelming volume of renovation, reinforcement, inspection and maintenance work that was done, and after a year 2003 marked by various equipment failures, the Phénix plant soon picked up its "cruising speed". This proved just how well the operator handled all activities and especially how simple it is to operate a fast neutron reactor, as rare was the number of failures caused by operational team errors.

After SuperPhénix was abandoned, which greatly disturbed those implicated in fast neutron reactor technologies, the Phénix plant once again found itself with a solid reason to continue operations thanks to both the "Bataille" law dated December 30, 1991 and the Generation IV Project. However, as the number of years in operation is limited, it is important to optimise them in order to com-

plete the entire programme of irradiation cycles. The period in which Phénix is currently entering therefore requires even more care when it comes to optimising core management, programmed shutdowns and irradiation experiments, etc. Even the imminent dismantling of the Phénix plant represents a new challenge that must be organised now.

What do the years ahead have in store for the Phénix plant? No one can say today, but generally speaking, the operational teams still hold a part of the plant's future in their hands. The current operations both justify and reward the efforts made by the entire team and management over the past few years. To continue in the same direction, it is absolutely necessary that work be carried out with rigour, passion, perseverance, and careful planning, in order to "cultivate" good luck as this small and fragile plant needs to be treated with care, affection, patience and persistence day in day out for it to bloom and thrive.





Alain LAFON

*Director of the Phénix plant
from June 1998 to April 2000*



Up and running after major renovation work

In late May 1998, after several years when the reactor had rarely been in power operation, the plant's reconnection to the grid ran very smoothly, with just a few technical hitches to delay operations. The reactor then ran for several months, giving 77 effective full power days (EFPD). During this period, a few relatively minor problems troubled the smooth running of the reactor. For example, the hydrogen detection system was sometimes difficult to control (owing to ageing of the ion pumps) and the turbine's oil filters clogged up. Problems like these caused brief outages. All in all, the plant's availability was about 70%.

During this period of power operation, we were convinced that despite the long outage that had preceded this new start-up and the major importance of some of the operations carried out (particularly the replacement of pipes in the secondary sodium circuits), the reactor was in good condition for power operation. This was partly due to the care taken in carrying out the work and partly to the way the operating and maintenance teams had kept their skills up to date. Because of this, we paid great attention to the organisation of subsequent works, at the price of considerably strengthening the ownership function (calling in engineers from the CEA's Military Applications department and the EDF's Engineering Centre in Lyons), and involving all the plant's staff in activities during outage, even if that sometimes took them far from their usual habits.

Two types of equipment particularly preoccupied us: the intermediate heat exchangers and the steam generators. It was a leak from an intermediate heat exchanger that caused the reactor outage in November 1998: sodium was escaping from a secondary circuit in the vessel containing the primary sodium. That apparatus and another matching one were replaced by new heat exchangers whose delivery gave rise to bitter debate with the manufacturers. As to the steam generators, having long tried to show that minor defects found on dismantled modules were acceptable in operation, prudence led us to make punctual repairs to the modules still in service.



So, even before undertaking long, complicated and costly works to check the condition of different parts of the facility and ensure that the plant complied with the safety standards in force in 2000, given the information acquired during the 1998 irradiation cycle we knew that subsequent resumption of power operation would be possible under good conditions.



RESULT



Landmarks

5 May 1969	First concrete poured at the Phénix site
31 August 1973	First divergence of the Phénix reactor
13 December 1973	First connection to electrical grid
12 March 1974	Reactor reaches rated capacity
14 July 1974	Phénix plant commissioned
11 December 1974	Tests with reactor at maximum power (597 MWt)
11 July 1976, 3 October 1976 and 31 August 1977	Secondary sodium leaks from intermediate heat exchangers
29 April 1982, 16 December 1982, 15 February 1983 and 20 March 1983	Sodium-water reactions in steam generators
10 August 1982	Net cumulative electrical output: 10 billion kWh
25 August - 13 November 1983	Uninterrupted operation at rated capacity (81 days)
30 October 1988 - 8 January 1989 16 January - 28 March 1989	Successive cycles running uninterrupted (72 EFPD in 72 days each)
6 August 1989, 24 August 1989, 14 September 1989 and 9 September 1990	Negative reactivity trips
11 May - 18 August 1990	Reactor running uninterrupted at 500 MWt (99 days)
22 July 1990	Net cumulative electrical output: 20 billion kWh
7 April 1995	Boitix 9 experimental sub-assembly reaches 144 174 MWd/t
27 October - 23 December 1999	Inspection check on conical shell
15 March - 6 April 2001	Visual inspection of core cover plug and upper internal structures of the reactor
1 September 2003	100,000th hour supplying EDF electricity grid



The Phénix plant has been in operation for thirty years now - and they have been eventful years. The power plant that opened in 1974 was very different in its organisation and technical equipment from the plant of 2004, fully renovated for its final years of power operation. Yet there has been continuity, great enthusiasm and huge successes.

The design of the plant, highly advanced for its time, and the exemplary manner of its construction provided accumulated feedback still unequalled for a fast neutron reactor. It provided an abundant harvest of information on the overall system that makes up a power-generating sodium-cooled fast neutron reactor; on the design of all its components and their behaviour under representative flux and temperature conditions; on the behaviour of the materials of the core and structures; on the qualification of the fuel at increasingly high burn-up; on demonstration of the transmutation of minor actinides and long-lived fission products; etc.

These thirty years of operation have not been trouble-free, but each incident has provided information that has been deciphered and processed. Leaks from the intermediate heat exchangers, sodium-water reactions in the steam generators, sodium leakage, cracking of type 321 steel pipes, automatic trips due to

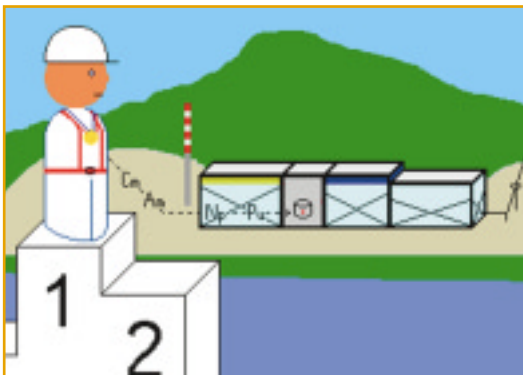
negative reactivity transients, etc. - all these difficulties were successfully overcome in order to continue operation.

Phénix is a **prototype** facility, built to demonstrate the viability of the French sodium-cooled fast neutron reactor project, which has since been put on hold with the government decision to permanently shut down the Superphénix plant at Creys-Malville in 1998, before its due date. Phénix is thus the only plant of its kind in France, and there are few elsewhere in the world. Most of the results and the specific operations carried out there have been, in a way, world firsts. Today it is one of the most fruitful experiments contributing to the technical credibility of fast neutron reactors, whether the coolant used is sodium or any of the other gases or liquids considered for the international research programme on Generation IV reactors.

With a maximum authorised output of 600 MWth (the rated capacity was 563 MWth), Phénix has so far accumulated an energy generation of more than 2 million MWh (equivalent to about 3,900 days in operation at rated capacity) and a net electrical power generation of 21 billion kWh.

The materials and equipment at the plant have given satisfaction. With the modular design of the steam generators and confirmation that the plant could be run with only two of the three secondary circuits functioning, it was possible to keep the plant in operation even during work on one of the steam generators or secondary cooling systems. This characteristic also made it possible to reduce the reactor's operating power but still irradiate experimental objects under a high neutron flux.

Following automatic trips due to negative reactivity transients, the reactor's **safety** was re-examined in detail and its design was not



called into question. The only danger is a reactor shutdown, and the possible causes of these automatic trips cannot lead to a damaging nuclear accident. But it is frustrating not to have identified exactly what caused these phenomena. The safety upgrading concerned a wide range of fields including seismic resistance, pipe cracking, the heat removal systems, leaks and major sodium fires. The work confirmed that the original design was correct, allowing that some improvements were necessary to meet increased safety requirements.

The in-depth renovation work and the checks carried out to enable the reactor's life to be extended in compliance with new safety requirements that were in some cases far removed from the original design, constituted a large-scale operation carried out inside a functioning nuclear power plant. The checks recently conducted on the most critical and heavily-used parts of the reactor block (core cover plug, conical shell) have shown that these structures hold up well over time, despite being subjected to severe stresses. It has been shown to be perfectly possible, although technically difficult, to inspect the internal structures of a sodium-cooled reactor.

Lastly, the use of joint CEA and EDF teams was a definite advantage, not only for the team members, who were introduced to a different corporate culture, but also for the project as a whole, which had the benefit of the two organisations' complementary knowledge, skills and assets. The same applies to the presence of GAAA during the construction phase, and to Novatome alongside the operator for conducting safety studies and renovation work.

The accumulated experience of the fast neutron reactor as such is positive, as regards fuel

density, manufacture of the MOX pellets, the particular core studies, the resistance of the cladding at high burn-up rates, etc. The fuel reaches a specific burn-up of 90,000 MWd/t at the centre of the core and 115,000 MWd/t at the periphery. Experimental pins and sub-assemblies have reached specific burn-up values of about 150,000 MWd/t, three times more than the design values.

Thanks to reprocessing the irradiated pins, the fuel cycle was closed several times, i.e. the plutonium recovered in the reprocessing plant was reintroduced into new sub-assemblies. A significant breeding rate was demonstrated at industrial scale: the plutonium produced amounted to 16% more than the initial quantity.

The flexibility of the reactor was put to extensive use to gradually introduce increasingly numerous and varied irradiation experiments. Thanks to its fast neutron flux, the small impact of the presence of targets in maintaining the chain reaction and the ease of loading and unloading individual sub-assemblies, the Phénix plant has become a valuable experimental tool, particularly for destroying or transforming undesirable radioactive elements. This is a major advantage of fast neutron reactors, as is confirmed by the direction now being taken in the Generation IV forum.

Because **sodium** reacts with both air and water, to use it as a coolant it was necessary to interpose adequate protective arrangements to prevent such reactions or limit their consequences. The integrated reactor design has proven its worth. Combined with the presence of secondary sodium circuits, it gives the system considerable inertia, which is particularly appreciable during transients. It is mainly this design we have to thank for the



low level of the total cumulative dose received by all those who have worked on the Phénix plant over these thirty years: less than 2 man-Sv to date, which is about the amount received by staff at two EDF's pressurised water reactors in a single year.

There have been few sodium leaks: one or two per year in power operation. All in all they have been small leaks of non-radioactive sodium, which have caused no damage to the facility and required only the repair or replacement of the defective part. The few, small sodium fires that have occurred have been quickly brought under control.

More disturbing was the gradual discovery of the behaviour of type 321 steel in the presence of sodium, a problem solved by replacing virtually all 321 steel in the plant. This type of steel, though commonly used in high-temperature industries, proved vulnerable to delayed re-heat cracking under the conditions in which it was used in the Phénix plant. Above all, it was shown to be highly sensitive to the way the welding was done.

A definite drawback is the opacity of sodium, though technical advances are tackling this problem with increasing success. Liquid sodium is above all an excellent heat conductor, therefore a very effective coolant for extracting the high density of energy produced in the core of a fast neutron reactor. Thanks to its high boiling point, it can circulate in non-pressurized circuits. It also has good neutron characteristics and does not become very active during its passage in the core.

Lastly, this type of power plant has a particularly high thermal efficiency. Although technological progress now enables us to build thermal power plants (using gas at very high temperature) with efficiency of over 50%, the average efficiency of the Phénix plant

(40%, despite running at reduced load for a quarter of the time), and its optimum value of 45%, are remarkable for a power plant of its generation.

To sum up, the Phénix plant has fulfilled its original contract to demonstrate the viability of a sodium-cooled fast neutron reactor. The results have even exceeded this objective as regards the fuel, the breeding aspect, the experimental irradiations, the sodium technology and upgradeability while in operation.

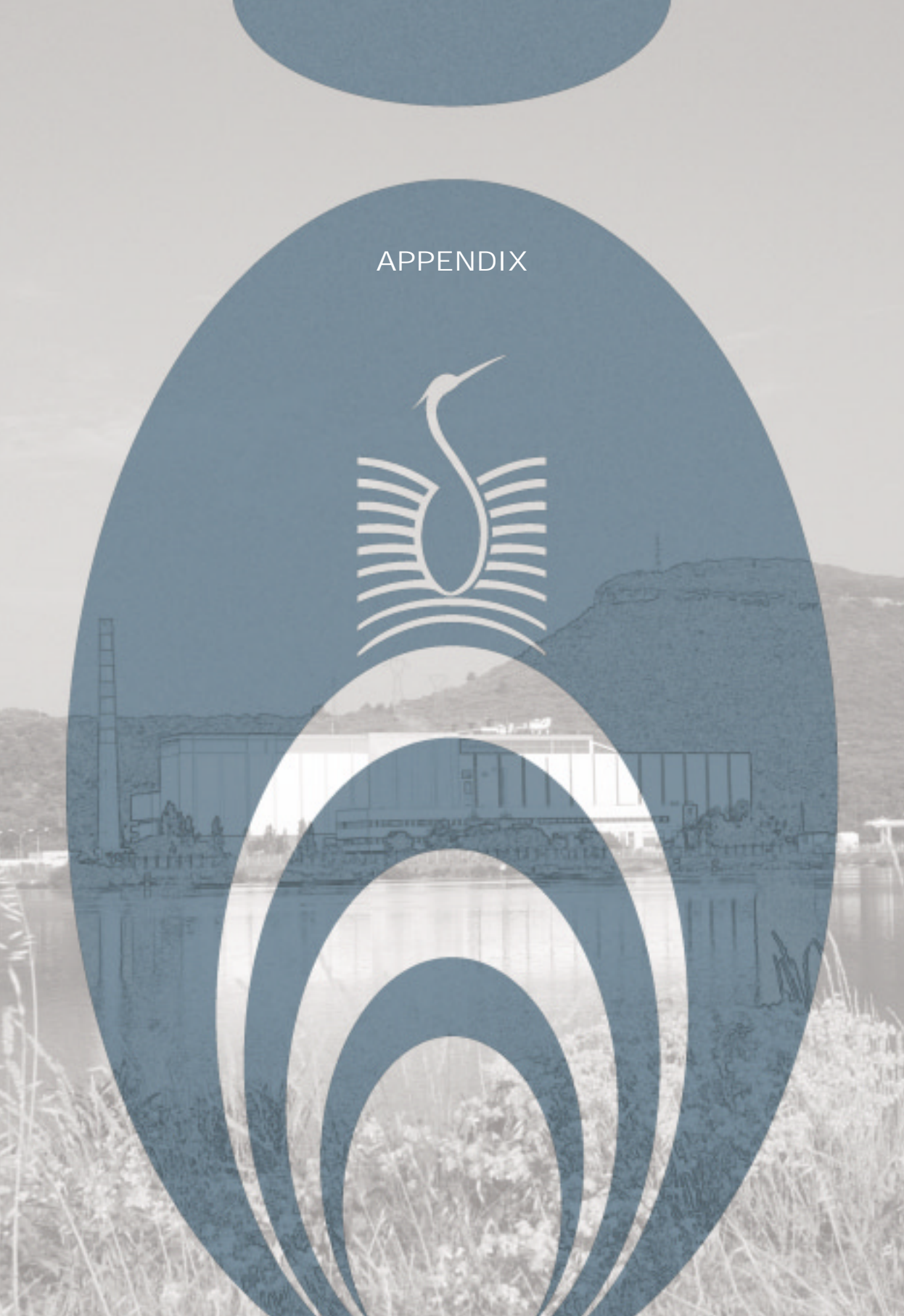
All these results are due to the efforts of the hundreds of engineers and technicians who have invested so much heart and energy as they followed one another in designing, building, operating, checking, repairing and upgrading the plant.

May this accumulated experience benefit future generations.

Marcoule, May 2004



APPENDIX



Located on the banks of the Rhône River on the Marcoule nuclear site in the Gard department, the Phénix plant is a prototype of a fast neutron reactor (or a fast breeder reactor [FBR]) using a sodium coolant and equipped with a turbo-generator set to generate electricity. The Phénix plant is an “integrated” reactor type, which refers to the fact that the core, the primary coolant pumps and the intermediate heat exchangers are all located in the same vessel.

Phénix plant operations are jointly run by the French Atomic Energy Commission (CEA) and Electricité de France (EDF). The CEA contributes 80% of the plant's budget while EDF provides the remaining 20%. The plant personnel is therefore composed of employees from the two organisations. The CEA is in charge of managing this collaboration, as well as being the owner and nuclear operator of the facility.

A.1. Core

The reactor core is composed of fissile fuel from which the greater part of the reactor's power is generated, the fuel itself being enveloped in a breeder “blanket” and a neutron shielding especially designed to limit the activation of the secondary cooling sodium in the intermediate heat exchangers. The core itself – corresponding to the fissile part – barely takes up a cubic meter, whereas the combined fissile + breeder part occupies a volume similar to a 2 meter-long cylinder with a diameter of 2 meters.

The **fissile fuel** is composed of plutonium in the form of a UO_2 – PuO_2 mixed oxide with a central region enriched by approximately 20% and a periphery region enriched by about 25%, which is designed to “level out” the neutron flux and homogenise heating. The fissile fuel is contained in approximately one hundred fuel sub-assemblies, each containing 217 fuel pins formed by a stack of oxide pellets of a diameter of 5.5 mm enclosed in a

stainless steel cladding. A stack of breeder pellets composed of depleted uranium oxide can also be found at the base of each fuel pin, which acts as a lower axial blanket. A spacer wire is spiralled around the fuel pins to guarantee the flow of sodium and optimise heat exchanges. Each fissile column is 85 centimetres long.

Each fuel bundle is placed in a stainless steel wrapper with a hexagonal cross section and also contains upper axial blankets – 37 depleted uranium oxide pins – and a combined stainless steel and boron carbide upper neutron shielding. A spike is located in the lower part of the fuel sub-assembly, which allows its vertical position in the diagrid. This spike is equipped with a gag designed to gauge the sodium flow rate in the assembly, which is generally greater in the centre than at the periphery. A series of “locks” also ensure that the assembly is not placed in a more central position than expected. The head on the upper part of the sub-assembly includes a groove that is used to handle each sub-assembly with a handling gripper. A fuel sub-assembly contains approximately 10 kg of plutonium, is 4.3 m long for a maximum diameter of 15 cm and has a total mass of 226 kg. A fissile fuel pin is 1.8 m long. This represents a total of approximately one tonne of plutonium which generates approximately 600 MWt, with a neutron flux of around 7.10^{15} n/cm².s at the core centre.

A **radial breeder blanket** of depleted uranium oxide is also made of pellets in about one hundred sub-assemblies, with each assembly containing 61 fuel pins. The structural elements of these sub-assemblies – cladding, spikes, heads – are identical to those used in the fissile assemblies, with the sodium being fed through the spikes in the diagrid. A breeder assembly contains 2 kg of plutonium after irradiation, has a mass of 294 kg and retains the same overall dimensions of those of a fissile fuel sub-assembly. The length of a breeder fuel pin is equivalent to 1.8 m.



Experimental sub-assemblies are fissile fuel sub-assemblies and are either:

- Sub-assemblies with a central channel designed to hold an irradiation rig. These sub-assemblies generally contain 180 fuel pins only and no upper axial blanket pins,
- Sub-assemblies differing from driver assemblies by their use of different materials or structural geometry.

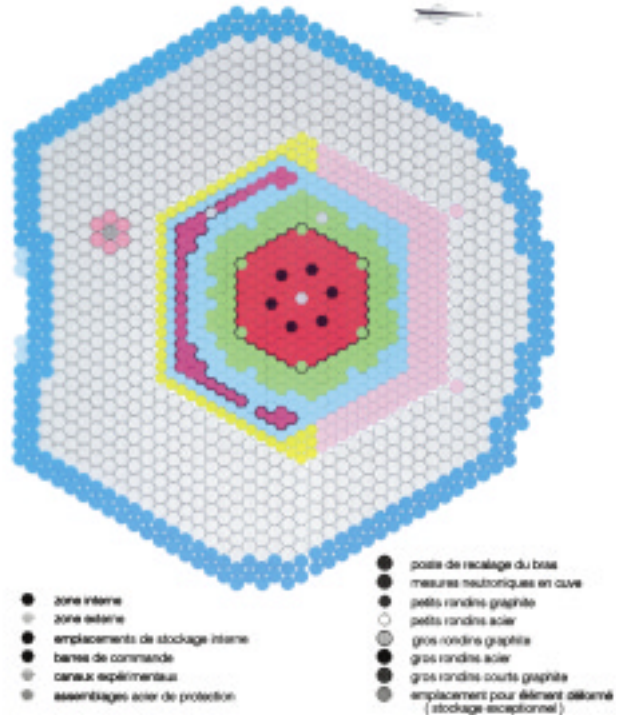
Other sub-assemblies carrying irradiations devices are made of steel and contain a central channel designed to hold an irradiation rig.

Such a sub-assembly is more commonly known as an irradiation and in-core measurement device or a DIMEP^[1].

Absorber sub-assemblies are composed of a sliding control rod in a hexagonal guide tube that is handled in an independent manner. From bottom to top, the control rod of each sub-assembly is composed of: a guiding spike, 7 stainless steel clad fuel pins each containing a stack of boron carbide pellets (in contact with sodium), a flexible rod and a bonding head to guarantee connection with the control mechanism and for handling purposes. The control rod is 4 m long with a mass of approximately 50 kg and a diameter of 12 cm. Absorber fuel pins are 1.2 m long, whereas their guide tube has a mass of 115 kg and is approximately the same size as a fuel sub-assembly sheath. A specific absorber sub-assembly acting as a complementary shutdown system has been implemented in the core centre since 1996. Its driveline assembly is composed of six 0.9 m long absorber fuel pins and is without a guiding spike.

Beginning with the blanket, the **lateral neutron shielding** is composed of:

- rows of stainless steel elements resembling a fuel sub-assembly from the outside (same hexagonal cross-section), placed in the diagrid and cooled by the forced convection of sodium; approximately forty spaces designed to hold irradiated sub-assemblies to be unloaded are also located here,
- rows of stainless steel billets (circular cross-section elements) placed in a structure encircling the diagrid and known as a lateral shielding support; these billets are cooled by the natural convection of sodium.



IMPLANTATION GENERALE DANS LE COEUR

[1] Dispositif d'Irradiation et de Mesure en Pile



A.2. Reactor block

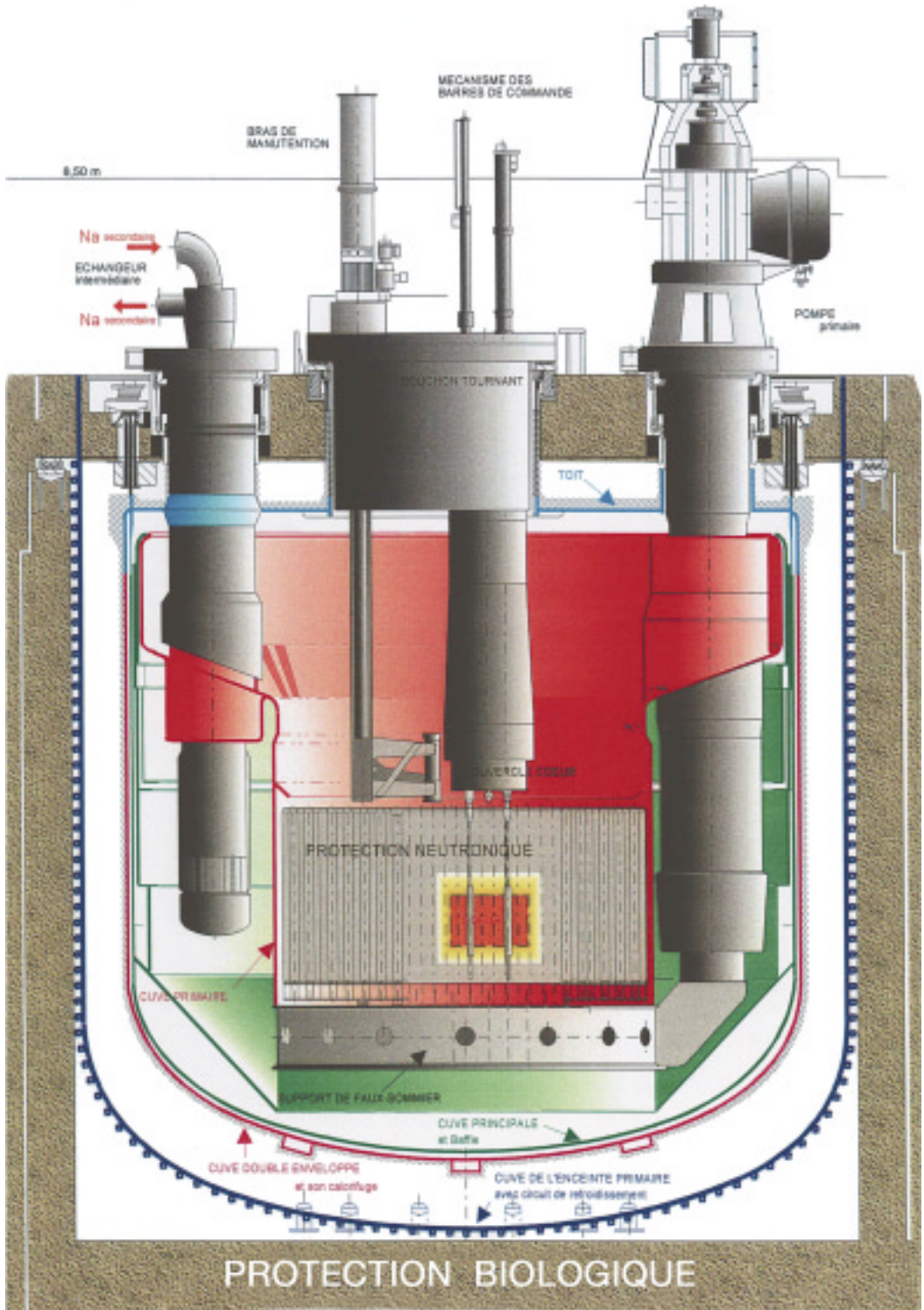
The diagrid includes a sodium distribution box pressurised by the primary coolant pumps, with sodium being supplied through three pipes. The diagrid guarantees the correct distribution of the coolant to the different regions in the core in relation to the thermal power to be removed from each assembly. In order to do this, the shroud tubes in which the core assembly spikes are held are fitted with six openings decreasing in size from the core centre towards the periphery. The diagrid and the lateral shielding support surrounding the core are fixed to matting that is held by the main vessel via a conical shell. This set of components supports the weight of all assemblies which is equivalent to 350 tonnes.

The **main vessel** is 10 metres high and 12 metres in diameter yet the part of the vessel resting in the sodium is nozzle-free so that risks of leakage are reduced to a maximum. The upper reinforced part of the main vessel is extended by 21 hanger rods. The main vessel is therefore supported by the slab that forms the upper part of the reactor block. This slab also bears the weight of the primary coolant pump and intermediate heat exchanger supports, as well as the rotating plug – at the centre of the slab – needed for fuel handling, whose lower part is extended via the core cover plug regrouping all of the core's instrumentation. Leaktightness between the slab and the rotating plug is guaranteed by a fusible metal seal. This seal can be described as a collector filled with a lead-bismuth eutectic alloy that remains solid while the reactor is in operation and can be fused using heat resistors in order to allow the rotating plug to turn during handling operations. The upper part of the main vessel is closed off with a flat roof equipped with pumps and exchangers penetrations. The roof is then connected to the rotating plug shell which centres everything in relation to the slab.

Inside the main vessel, the primary vessel separates the sodium coolant into two pools. The primary vessel is composed of a shell extended by a conical baffle that contains twelve channels equipped with sleeves for the primary pumps, the intermediate heat exchangers and other components. This is followed by a new shell that envelopes the core and is welded to the lateral shielding support. The hot sodium (560 °C) is therefore contained inside this vessel and exits through the intermediate heat exchangers to be pumped cold (400°C) by the main pumps that drive the sodium into the diagrid. The hydraulic baffles are supported by the conical shell and leave space for two concentric rings on the periphery of the main vessel. These two rings are supplied in cold sodium by the diagrid, thus maintaining the main vessel temperature at approximately 400°C during reactor operation.

For safety reasons, a **safety vessel** envelopes the main reactor vessel and is designed to contain any possible sodium leaks without the sodium level falling too low and hindering core cooling. The safety vessel and the main vessel roof are both insulated. A third vessel, the containment vessel, is made from ordinary steel and envelopes the first two vessels. This vessel is welded under the concrete slab and is maintained in a nitrogen atmosphere. This vessel is designed to contain any active products that may be released from the main vessel in the event of an accident. Furthermore, the containment vessel is equipped with a cooling system – more commonly called the emergency cooling system – that is cooled via a heat exchanger initially supplied with water from the Rhône River but was then connected to air coolers in 2002. This cooling system is designed to maintain the reactor pit concrete at ambient temperature when the reactor is in operation, as well as evacuate residual power in the event of the failure of the secondary sodium cooling systems.





A.3. Reactor circuits

Owing to its integrated design, the reactor block contains the **main primary cooling circuit**. This system holds approximately 800 tonnes of sodium, which provides the system with great inertia. Argon acts as an inert blanket above the sodium. The three pumps and the six intermediate heat exchangers are positioned in the annular space between the primary vessel and the main vessel. The primary coolant pumps are vertically submerged in the cold sodium (400 °C) and suspended from the upper part of the slab. They are connected to the diagrid with hinged pipes. These pumps are driven at a continuous variable speed of 150 to 820 revolutions per minute using the main motor or at 100 revolutions per minute using a pony motor. These pipes produce a unit flow of roughly 1,000 kg/s. The intermediate heat exchangers are connected two by two to a secondary cooling circuit and suspended from the slab in a similar way. These straight-tube heat exchangers - over 2,200 pipes forming 19 rings - are fixed onto lower and upper tube plates by expansion and welding techniques^[2]. The primary sodium coolant flows outside these tubes.

Auxiliary circuits are designed to store, refill, drain and purify the primary sodium, as well as control the pressure, the purification process and the release rate of the argon blanket. These circuits are constantly filled with primary sodium coolant and are located in leak-tight cells maintained in a nitrogen atmosphere to avoid active sodium fires. These circuits are equipped with a cold trap and a plugging indicator that are specifically used with sodium and based on the variation in solubility of sodium impurities in relation to the sodium temperature. The sodium is cooled in the cold trap which causes all impurities (oxides and hydrides) to precipitate

and collect on a stainless steel mesh filter (KNIT wad). The temperature of the sodium circulating through the plugging indicator varies according to the predefined cycles. The temperature at which the sodium no longer flows through a calibrated nozzle is measured: this is known as the "plugging temperature". This plugging temperature must be kept as low as possible by purifying sodium while the sodium flowing in the circuit must remain several tens of degrees higher than this plugging temperature.

Three completely and independent **secondary cooling** circuits guarantee the transfer of heat from the intermediate heat exchangers to the steam generators via the sodium. The sodium in these systems is not radioactive thanks to the neutron shielding enveloping the core. At the top of the intermediate heat exchangers, the secondary sodium enters a central tube and flows to the bottom of the distribution box with a convex bottom welded to the lower tube plate. This sodium then goes back up and inside these tubes. Once the sodium has passed through the bundle, it then exits the intermediate heat exchanger via a header equipped with a lateral pipe.

Under normal operating conditions, the sodium exits the intermediate heat exchangers at 550°C and enters at 350°C, flowing at a rate of approximately 800 kg/s and driven by a mechanical pump located in the expansion tank of each system. The main pipes - with a diameter of 500 mm - supply two intermediate heat exchangers and cross through the reactor building to the steam generator building where the secondary cooling system facilities can be found: main pumps, buffer tanks, valves, auxiliary circuits (storage, refilling and purification). The circuits are drained by gravity into storage tanks located on the bottom floor of the building. Each circuit contains approximately 140 tonnes of sodium. All remaining sodium-free circuits and tanks are filled with argon.

[2] "Expansion" refers to the mechanical operation of expanding each tube so as to force the tube wall tight against its hole in the tube plate, so keeping it in place and achieving a preliminary level of tightness. The seal is perfected with a ring of welding.

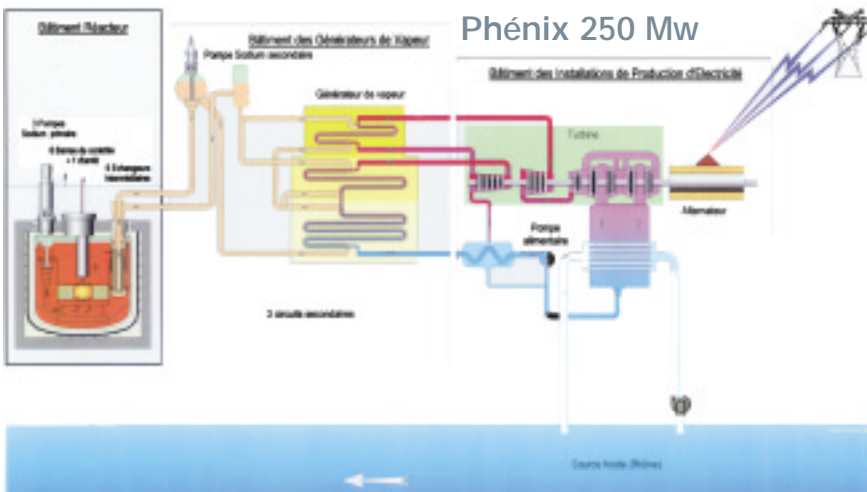


Each secondary cooling circuit is connected to a **steam generator** composed of three parts (evaporator, superheater and reheater) that are each composed of twelve modules. The hot sodium flows through the superheater and the reheater at the same time with steam exiting at a temperature of 512°C and a pressure of 165 and 34 bar respectively. The totality of the sodium then flows through the evaporator that, supplied with water at 246°C, provides slightly overheated steam at 375°C. The evaporator is made from ferritic steel whereas the superheater and reheater are made from austenitic steel. These modules are S-shaped, or two S-shaped modules in the case of the evaporator. Each module is composed of 7 tubes placed inside a shell with water flowing in the tubes and sodium flowing outside the tubes in the opposite direction. Systems detecting water leaks into the sodium protect the facility against sodium-water reactions. The steam generators can also be used as sodium-air heat exchangers to evacuate the reactor's residual power: hatches below and above the steam generator casing are opened and air flows around the modules by natural convection.

Thermal energy generated by the reactor and transported by steam into the steam genera-

tors is transformed into electrical power using a turbo-generator set (250 MWe) at 3,000 revolutions per minute, which is connected to the EDF high-voltage electric grid (225 kV). The turbine is a combined impulse-turbine composed of three cylinders on one unique 39 metre long shaft line. The superheated steam coming from the steam generators takes effect in the high-pressure cylinder. This steam then goes back through the reheater part of the steam generators, before spreading out in the medium-pressure cylinder and the two low-pressure cylinders.

The equipment in this electricity generating system - condenser, low and high-pressure reheaters, water tank, feed-water pumps, steam headers, turbo-generator set, turbine by-pass, buffer tanks and drains, etc. - are similar to those found in a conventional power plant. It is nevertheless important to point out that an evaporator steam/water by-pass circuit connected to each steam generator was added to evacuate the reactor's residual power after a turbine trip. Steam generators, of the forced circulation type, require very pure water whose total volume is filtered through a water treatment station. The condenser is cooled with water drawn from a pumping station on the Rhône River.



A.4. Handling operations

Refuelling is performed when the reactor has been shut down. A rotating arm – fixed in an off-centred fashion to the rotating plug that is itself off-centred in relation to the reactor core – is used to transfer fuel sub-assemblies from their initial position in the core to the intermediate storage area and the removal station. The combination of the rotating arm and the rotating plug makes it possible to reach all positions. After having cooled down in the internal storage area, each element is removed from the reactor using an immersed handling bucket located in the removal station. This sodium-filled handling bucket is manoeuvred by a loading carriage along the primary transfer ramp and placed in the A-framed transfer lock above the slab. This bucket is then tipped and lifted down into the transfer lock and carried down the secondary transfer ramp to a storage drum capable of holding approximately one hundred fuel sub-assemblies. Each sub-assembly remains in a bucket, with all buckets floating in the sodium in the storage drum capable of holding 180 tonnes.

Fresh-fuel sub-assemblies are stored in a storage room upon their arrival on site. After dimensional check and once thermally conditioned, these sub-assemblies are placed in the storage drum in a bucket filled with sodium. During core refuelling, fresh-fuel sub-assemblies go through the opposite procedure to that applied to spent sub-assemblies. These sub-assemblies are transferred from the storage drum to the reactor, where they are directly placed in specific positions defined by neutron calculations. Several fresh-fuel sub-assemblies are stored in the reactor's internal storage area in case a sub-assembly demonstrates cladding failure or thermal abnormality during an irradiation cycle. Faulty sub-assemblies can simply be replaced internally with fresh sub-assemblies.

Spent sub-assemblies are stored in the storage drum for a cooldown period that depends on their residual power. These sub-assemblies are then removed one by one from their bucket using a gripper that transfers them directly into the irradiated elements cell maintained in a nitrogen atmosphere. After the sodium dripping process has been completed, this sodium is destroyed using a wet carbon dioxide gas stream. Next, the sub-assembly is moved to the annex cell^[3] where it is dismantled: the spike and head are sawed off, the hexagonal wrapper is milled opened at the two opposing angles and the fuel bundle is removed using a jack that pushes the bundle into a container. This container is returned to the irradiated elements cell to be sealed in two concentric cases.

These conditioned fuel pins are then transported in a transfer cask to a storage site or fuel reprocessing plant. Sub-assembly structural waste – heads, hexagonal wrapper, spikes – are placed in bins that are transported in transfer casks to a storage unit located on site at Marcoule. Specific operations can also be performed on particular experimental sub-assemblies, such as preparing fuel pins and rigs or reconstructing rigs or sub-assemblies using previously irradiated fuel pins. All such operations are carried out using remote controlled systems by operators behind observation ports more than a metre in thickness.

A specific device known as a neutron radiography reactor is associated with this facility. Using a highly enriched uranyl nitrite solution, this reactor provides bursts of neutrons dedicated to examining the internal structures of irradiated fuel pins. This reactor provides examinations complementing other visual, radiographic and gammagraphic examinations performed on the fuel pins in

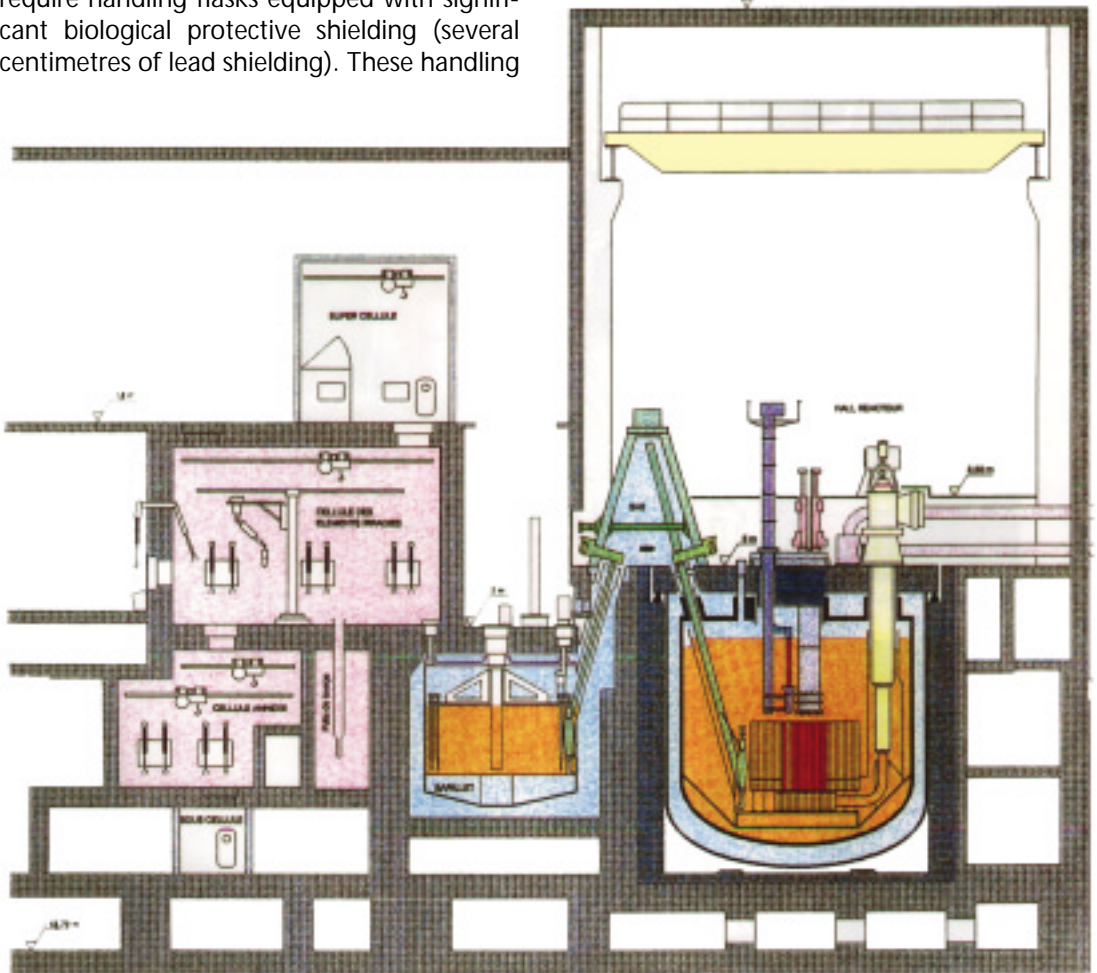
[3] The annex cell was built in 1980. Beforehand, sub-assemblies were dismantled in the irradiated elements cell using a different process to cut the hexagonal wrapper.



the irradiated elements cell. Operations on irradiated sub-assemblies are carried out independently to the reactor state, either while reactor operation or during the reactor shut-down. When such hot cells do not contain irradiated fuel pins, they can be used in an air environment, which simplifies maintenance using remote control systems.

The different **removable components** in the reactor block descending into the vessel through the slab and the rotating plug (primary coolant pumps, intermediate heat exchangers, handling arm, control rod mechanisms, etc.) can be dismantled for maintenance purposes. The lower end of such components being active, handling activities require handling flasks equipped with significant biological protective shielding (several centimetres of lead shielding). These handling

flasks, the heaviest weighing 110 tonnes empty, can be connected up in a leak tight manner using mobile locks equipped with valves on top of the slab. Handling flasks are manipulated in an upright position using the travelling crane, before moving through a lock equipped with a transporter. They are then recuperated in the hall dedicated to special handling activities located in the handling building where their content is emptied into different pits to be cleaned, decontaminated, repaired and stored. Cleaning is carried out by spraying water from fixed ramps into a pit maintained in an inert gaseous environment. Decontamination is carried out through successive diluted acidic washings.



A.5. Instrumentation and Control

The nuclear power of the core is measured by a series of detectors positioned under the reactor vessel. Two detectors can also be found inside the reactor vessel, vertically positioned in the lateral neutron shielding in order to guarantee valid measurements at very low power rates and especially during reactor criticality. The control rods – a total of six – are powered by mechanisms mounted on the rotating plug. All six control rods are designed to guarantee safety functions (automatic shutdown), compensation (reactivity variations) and control operations (power adjustments). The control rod system is composed of two groups of three mechanisms each, with differing designs (pinion and rack drive or travelling nut drive), including independent control systems. Since 1996, a seventh control rod, composed of several articulated components that remain insertable even in the event of core deformation, has a safety function (the complementary shutdown system): positioned at the centre of the core and held in place with an electromagnet, this control rod handles shutdowns and maintains the reactor in a zero power state, as well as re-establishing a thermal state ensuring structural integrity ($T < 450\text{ °C}$) in the event of a rod drop from a rated power. The six control rods and the absorber sub-assembly of the complementary shutdown system are placed at the bottom of their housing during handling activities so that the rotating plug with the sub-assembly transfer arm can operate.

The temperature of the sodium at the outlet of each fuel sub-assembly is monitored by two thermocouples guided through the core cover plug. Other measuring devices have also been implemented to detect abnormal

core operation and monitor the temperature of several structures. Detecting and locating cladding failure helps detect fuel pin leak-tight problems at an early stage, which makes it possible to assess the extent of the problem and decide whether the fuel sub-assembly in question should be unloaded or not. Primary sodium samples are taken on a continuous basis at the intermediate heat exchanger outlets (detection) and above each fuel sub-assembly (location) for scanning. Associated measuring devices detect fission products emitting delayed neutrons that reveal cladding failures when detected in the sodium.

The entire plant is monitored by operators in the control room located on the second floor of the control room and office building. Other than alarm windows and conventional control blocks, the facility is equipped with several independent data processing systems, the last of these systems being added during the plant operation, such as the:

- Rapid Temperature Monitoring System is designed exclusively to monitor sodium temperatures at fuel sub-assembly outlets and triggers safety action (automatic shutdown),
- Central Data Processing System draws on all the other measuring and signalling devices at the plant but does not trigger safety action but alarms only,
- Hydrogen Detection System specifically monitors hydrogen production rates in the secondary sodium and triggers appropriate safety action (rapid shutdown, isolation and decompression of the steam generator in question),



- Reactor Block Temperature Monitoring System draws on reactor block structural temperature measurements without triggering safety action,
- Data Acquisition Back-up System monitors all essential safety parameters as a back-up to both conventional means and the Central Data Processing system,
- Rapid Data Acquisition System rapidly receives and processes a whole set of parameters capable of providing information in the event of a negative reactivity trip.

The computer and electronic rooms are next to the control room and house the main monitoring computers and electronic equipment. The relay circuitry takes up part of the first floor in this building, whereas the medium and low power electrical distribution panels are located on the ground floor. A back-up room located in an underground passageway is used by the shift team to monitor and control the shutdown reactor if

a problem, such as fire or attack, was to prevent them from using the control room.

Other than external electric power sources, including the normal 225 kV power line and the 20 kV back-up power line, electric power can also be distributed by internal sources including the back-up diesel-powered generator sets and batteries. Two diesel-powered generator sets each generate enough power to supply auxiliary equipment that must continue functioning after an automatic shut-down. With each safety upgrading of the plant, various new equipment and components have been added. Therefore, in the event of the total loss of all external electric power sources and the first two generator sets, two other diesel-powered generator sets provide the power required to correctly run the back-up cooling system, decay heat removal circuits and several actuators. Four other diesel-powered generator sets designed to supply specific equipment in the event of a power loss were also implemented.

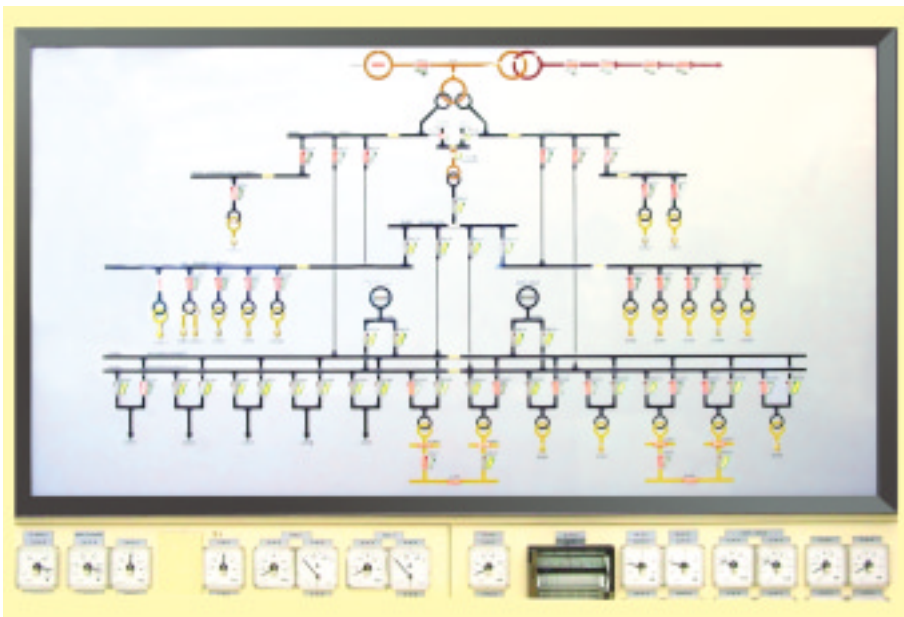


Diagram of the electrical power supply



A.6. Safety Functions

Core reactivity is controlled by manoeuvring the control rods. Fission reactions are maintained at a strictly minimum level required to generate the desired power level by inserting or removing the absorbers. The various different neutron and thermal measurements, for example, make it possible to check that this level is respected. If any parameter oversteps its normal operating range, operators are warned and can reverse the situation using the control rods. If the situation developed too quickly, the control system automatically provokes a shutdown. There are two different automatic shutdown sequences called: rapid shutdown and emergency shutdown. If an incident occurs in a secondary cooling circuits or in the electric power generation facility, the control rods are automatically inserted into the reactor core by approximately thirty centimetres in three and a half minutes. During the first minute, the nuclear power practically reaches a zero-power level. If a reactor-related incident occurs, the control rods and the complementary shutdown system rod gravitationally drop in less than a second, which brings the nuclear chain reaction to a halt. Other than the core (storage drum, irradiated elements cell, fresh fuel storage room), reactivity control is guaranteed by structural arrangements, as separation of the sub-assemblies suffices to prevent a critical scenario from occurring.

The characteristics of fast neutron reactor cores such as the Phénix core represent three main risks that must be carefully considered:

- the void coefficient is positive in the core centre, which means that reactivity increases if the sodium boils,
- the power density is very high, which renders the core sensitive to local or generalised cooling defects,
- the core is not in its most reactive configuration: compaction of the sub-assemblies increases reactivity.

Such phenomena require reinforced core monitoring. Thus, an emergency shutdown can be prompted by a) the medium-level overheating of the core or even the overheating of one sub-assembly only, b) a positive or negative variation in reactivity, c) fuel pin cladding failure, d) an earthquake or e) increase in the sodium temperature at the core inlet.

However, the Phénix core benefits from feedback effects providing the reactor with inherent safety characteristics. The Doppler effect is the first of these feedback effects, which is mainly due to the breeder nuclei: when the temperature rises, the neutron capture rate of uranium-238 also increases, which proportionally decreases the core reactivity. Other temperature effects act indirectly by globally reducing the chain reaction by expansion of the fuel and structures. The constant flow of the coolant guarantees core cooling, even in the event of an accidental situation thanks to the integrated design of the reactor. Furthermore, as xenon and samarium poisoning is non-existent, reactor operations are simplified following a shutdown, which also improves safety levels (only the accumulation of neptunium-239 has a slightly positive effect at the beginning of each cycle).

Decay heat removal in the reactor after shutdown is usually carried out by the secondary cooling circuits, the steam generators and the condenser in the turbine hall. If the condenser happens to be unavailable, the steam generator casing hatches are opened and the air flowing around the modules of only one steam generator suffices to remove residual power. In the event of an accident during which all three independent secondary cooling circuits are



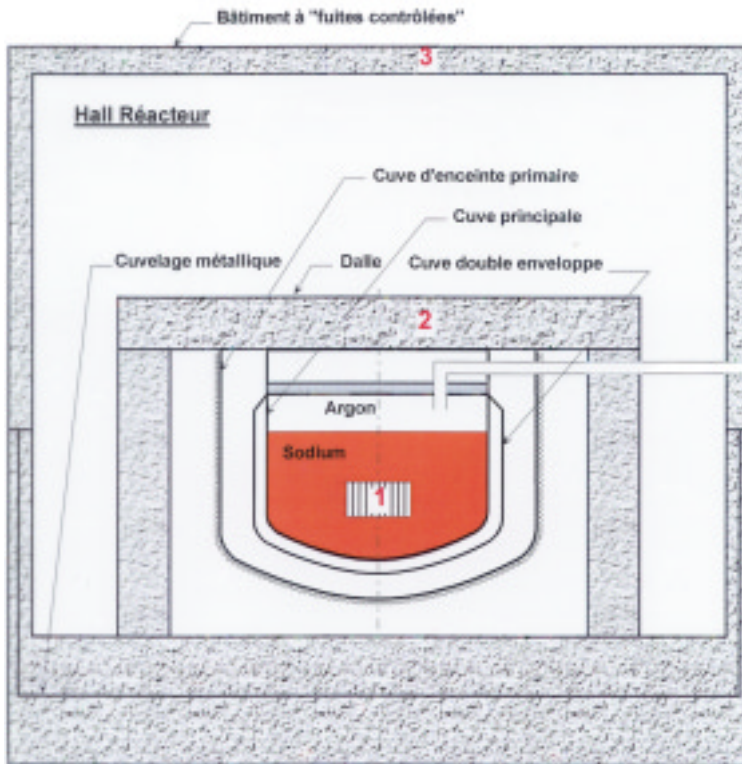
unavailable, the emergency cooling circuit is designed to maintain the reactor block temperature at an acceptable level while absorbing the thermal radiation emitted by the main reactor vessel. This emergency system was considerably renovated between 1999 and 2002 to guarantee its smooth running even after an earthquake of high intensity.

The residual power of sub-assemblies placed in the storage drum is removed via a specific oil-filled cooling system, which is itself cooled by water from the Rhône River. The sub-assemblies are only extracted from the storage drum when their residual power is sufficiently low so that cooling can continue naturally in a nitrogen environment in the irradiated elements cell.

The **containment** is designed to isolate the plant environment and staff from nuclear materials (fuel) and fission product releases, thanks to a series of consecutive barriers. The leak-tight cladding enveloping the fuel pins represents the first barrier. The second barrier is composed of several containment vessels: a) the main vessel doubled by the safety vessel for the main part and closed off in the upper part of the reactor by the roof, b) the containment vessel closed off by the concrete slab in the upper part of the reactor and c) the circuits connected to the primary cooling circuit (primary sodium purification, cladding failure detection, primary argon). The integrated reactor design (the core and all components located in the same vessel) makes it possible to integrate

a second, very compact and therefore very reliable barrier, not to mention the fact that the succession of vessels ensures that the primary sodium constantly cools down the reactor core. The third barrier is composed of a controlled leak-off type reactor building. The infrastructure is integrated into a metal leak-tight liner fitted with a cathodic protection. The superstructure includes a concrete construction composed of a steel framework supporting the roof.

Other than these three barriers, it is also worth mentioning:



Schematic of the containment. 1 : fuel pins clad. 2 : Second containment barrier. 3 : Reactor building



- barriers containing the secondary sodium coolant (pipes, tanks, etc.),
- barriers vis-à-vis handling (A-framed fuel transfer lock, storage drum, irradiated elements cell, etc.).

Radioactive liquid effluents produced at the Phénix plant are collected in two main 20 m³ storage tanks, then inspected and transferred by tank truck of a capacity of 8 m³ to the liquid effluent waste treatment plant (STEL) at Marcoule where such waste is treated with waste coming from other site facilities. Gaseous effluents – mainly argon from the cover gas – are purified and deactivated before being monitored and released through a stack. Solid waste (dismantled sub-assembly structures, operational waste) is monitored and transferred towards ad hoc facilities on site at Marcoule.

Using sodium as a coolant requires taking several precautions owing to its extremely high reactivity when in contact with air and water. In terms of sodium fires, other than the quality of the piping and tanks, free sodium levels are systematically protected by an inert argon or nitrogen blanket. Leak detectors (beaded wires or spark plug leak detectors) and fire detectors (sodium aerosol analysers) equip all plant components and equipment. In terms of the primary sodium coolant, systems outside the reactor vessel pass through rooms maintained in a nitrogen environment. In the event of a leak, the faulty circuits are usually drained, except for the main vessel which is doubled by two other vessels in a nitrogen atmosphere so that sodium levels can always cool down fuel sub-assemblies. Sodium fires are not as fierce as hydrocarbon fires and can be extinguished using the “Marcalina” powder composed of sodium carbonate, lithium and graphite.

The risk of a reaction between sodium and water does exist in the steam generators as only the exchange tubes separate the two fluids. The interposition of the secondary cooling systems is uniquely designed to prevent such a reaction from occurring with the radioactive primary sodium. Hydrogen – generated during a reaction between sodium and water – is used to indicate the presence of a leak in a tube: the hydrogen content in sodium and argon in the secondary circuits is measured constantly. The plant operators are warned in the case of an abnormal generation of hydrogen. The operators shut down the plant, dry the faulty steam generator, isolate the water-steam section and then drain the sodium system. Therefore, the two reactants are no longer in contact. Such actions are executed automatically if the hydrogen content rises rapidly. Last of all, in the event of a severe tube rupture, the increase in pressure in the secondary circuit due to the excessive generation of hydrogen leads to the rupture of the bursting discs, which causes the sodium of the steam generator to empty at an even greater rate.



A.7. Building infrastructure

The Phénix plant is located north of the Marcoule site on a platform covering several hectares. The main buildings are lined up in parallel to the Rhône River and form a series of building of approximately 150 metres long and 42 metres wide. Stretching from south to north, the following buildings can be observed:

The handling building is composed of two main parts. The first is devoted to fuel aspects, and contains the storage drum, the irradiated elements cell and their auxiliary equipment. The second part is devoted to operations on removable reactor block components, including an operation and storage cell, cleaning and decontamination pits and the liquid effluent reception and control system.

The reactor building houses all active primary cooling circuits. The reactor – suspended from the slab – is positioned in a concrete pit. Primary system auxiliary equipment (cold trap, cladding failure detection system, etc.) are mainly found in the two rooms equipped with biological shielding and maintained in a nitrogen environment.

The two above-mentioned buildings form the restricted area with respect to radiation protection issues and are equipped with a special ventilation system that creates a slight vacuum pressure in comparison to the outside atmosphere. The foundations of these buildings extend 11 metres underground and their superstructures – 35 metres high – are high enough to facilitate equipment handling activities.

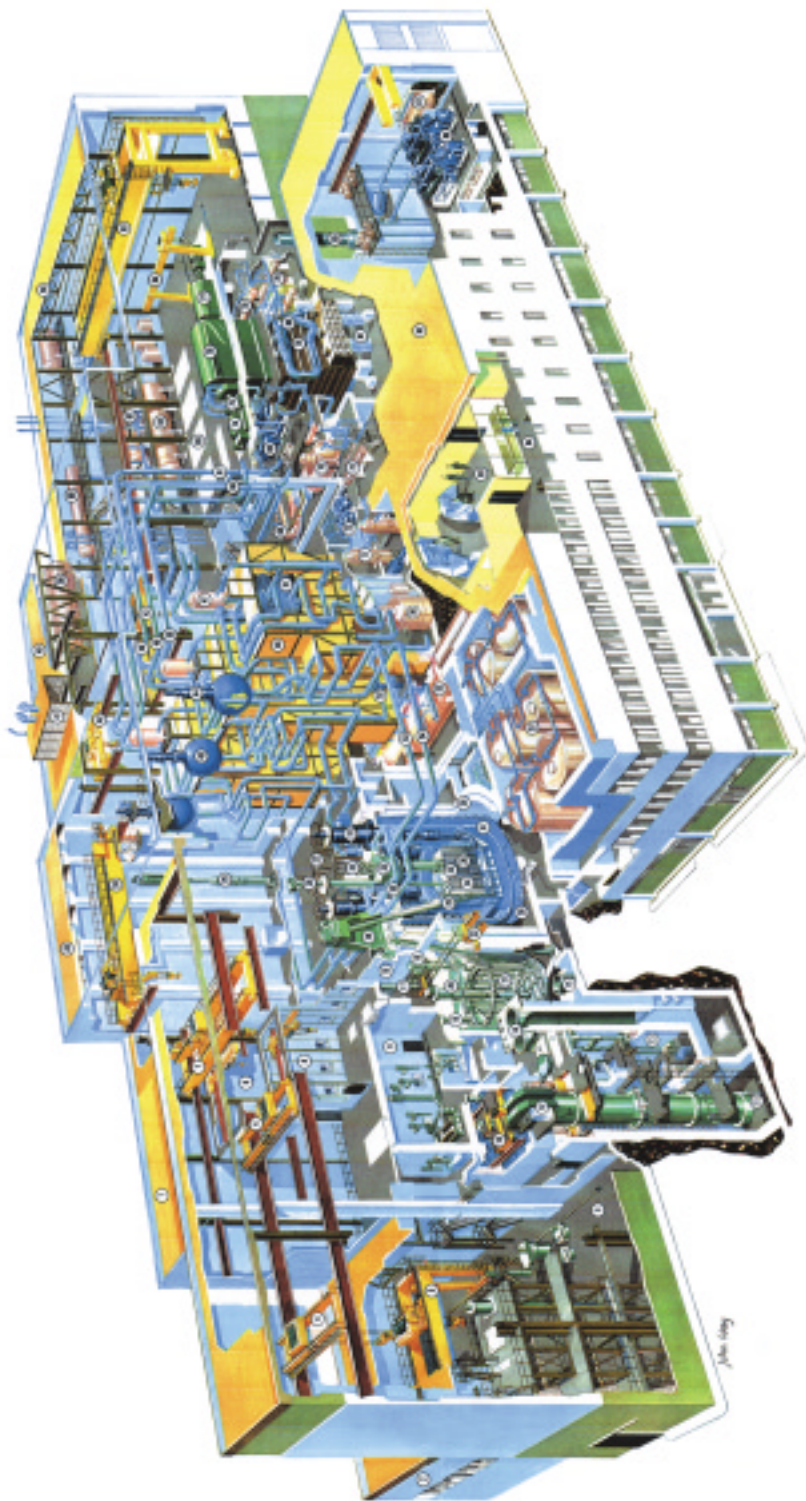
One half of the steam generator building contains the secondary cooling circuits, the draining tanks, the steam generators and their auxiliary equipment, whereas the other half is used for handling steam generator modules.

The turbine hall looks like any other conventional facility containing a turbo-generator set and associated auxiliary equipment (condenser, water tanks, feed water pumps, etc.), with a demineralised water treatment station next to this building. The electrical substation (transformers) is located north next to the turbine hall.

Various different buildings required during reactor operation can be found surrounding these buildings:

- The control room and office building houses two diesel-powered generator sets, electrical and electronic equipment, the control room and offices,
- The annex building houses various different auxiliary reactor equipment such as the ventilation system, cooling systems, nitrogen systems,
- The pumping station is completely underground (as much as 17 metres under) and includes two 200 metre pipes located in the Rhône River. The realise pipe for condenser water can be found a little further downstream.
- The buildings in which the pumps, generator sets and the air-coolers of the new back-up cooling system were built in 1999 on each side of the reactor building,
- The plant stack used for air ventilation and gaseous effluent releases,
- The visitors reception building including the physics laboratories and the maintenance building are used as offices, as are various other small buildings.





A.8. Plant operation

To perform a **start-up** when the facility is in a 250°C isothermal state, the plant operator raises the complementary shutdown system rod, before simultaneously raising the control rods in a curtain-like movement to a predetermined level. The control rods are then progressively raised to a level and criticality occurs when the last rod has been raised. In parallel, the condenser and the turbine are preheated thanks to steam generated by the steam supply facility on site at Marcoule. Reactor power is increased to 5 MWt, whereas its temperature is maintained at 250°C and the steam generators are progressively supplied with water. The power generated by the fuel is used to heat the plant (primary sodium coolant, secondary sodium coolant, steam generators, electric power production facility) while making sure that all thermal gradients and regulatory heating rates are respected. When a global temperature of 360°C is reached, the steam generators are switched to the steam phase. The turbine is activated when the steam meets design characteristics of 140 bar and 400°C. The generator is therefore connected to the grid seeing that the reactor is already generating several tens of MWt and all excess energy is removed by the turbine by-pass. The power increase takes place over several hours and is obtained by altering the level of the control rods as well as the primary and secondary coolant pump speeds.

The Phénix plant is not operated in relation to EDF's needs in electricity, nor does it play a part in the frequency control of the electric power grid. Phénix's power is generally operated at the maximum power tolerated by reactor equipment. The control rods are gradually raised once or twice per shift^[4] to compensate for fuel burn-ups, while retaining approximately the "curtain" configuration. The plant can also be freely operated at two thirds of its rated

power, using two of the three primary pumps, or two of the three secondary circuits. It is also possible to shut down one of the three primary coolant pumps during reactor operation. However, it is necessary to shut down the reactor when moving from three secondary systems to two secondary systems and vice-versa.

Shutdown of the Phénix plant is carried out gradually by inserting the control rods in the core. When reactor power is sufficiently low enough, the generator is disconnected from the grid and the steam is directed to the condenser. The reactor's temperature progressively decreases and the control rods continue to be inserted until the reactor is completely shut down. Automatic shutdowns (rapid or emergency shutdowns, cf. § A.6) only take a few seconds to occur. It is worth pointing out that the plant is not equipped with a house load operator control system. A turbine trip and an automatic shutdown are initiated when a defect of a few seconds occurs in the EDF high voltage power line to which the plant supplies electricity. Once the chain reaction has been stopped, the residual power is removed via the condenser before exiting through the steam generator hatches.

Under normal shutdown conditions, particularly when refuelling, the reactor temperature is maintained at 250°C. The control rods are placed in a low position so that the absorber rods are completely inserted in the reactor core. The core is composed of fresh and spent fuel sub-assemblies and does not need to be removed as spent fuel sub-assemblies are replaced with fresh fuel ones. For component handling operations or in the event of

[4] A shift, just like on a ship, is defined as the period of time during which the same team remains at work. At the Phénix plant and other EDF power plants, each shift lasts eight hours, including the overlapping times between the two teams. Six shift teams are required to operate the Phénix plant.



a long shutdown, the reactor is cooled down to 180°C. If core residual power is insufficient to sustain this temperature, energy from the primary and secondary pumps – when the secondary circuits are operating – is used to stabilise the reactor temperature. The electric heater from the primary sodium auxiliary system can also be used for the same reasons.

Fresh fuel sub-assemblies were produced by the Cogema fuel fabrication workshop at Cadarache until its closure in 2001. These fuel sub-assemblies were transported to Marcoule site and stored at the Phénix fuel storage room in line with strict safety regulations. When these fresh fuel sub-assemblies need to be used, they are once again inspected and thermally conditioned before being loaded into the stor-

age drum. These sub-assemblies are then transferred into the reactor during a refuelling campaign. After several irradiation cycles, during which the fuel sub-assemblies are usually moved around to optimise core management, the irradiated fuel sub-assemblies are transferred to the internal storage area located on the core periphery behind three rows of stainless steel sub-assemblies designed to reduce irradiation. When the residual power of these spent sub-assemblies has fallen below 10 kW, they are then transferred – usually during the next refuelling campaign – into the storage drum to continue their cooldown. Several months later, when these spent fuel sub-assemblies have cooled down so that their residual power is below 6 kW, they are then transferred into the irradiated elements cell to be dismantled.



NOMINAL TECHNICAL CHARACTERISTICS

Thermal power:563 MWt
Gross electrical output:250 MWe
Neutron flux at core centre	$.7 \cdot 10^{15}$ n/ cm ² .s
Fraction of delayed neutrons360 pcm
Active volume in the core1.4 m ³
Maximum temperature at pellet centre2,300 °C
Average power density1,200 kW/ dm ³
Maximum linear power rating of fuel pins450 W/ cm
Maximum temperature of cladding700 °C
Temperature coefficient	- 2.7 pcm / °C
Power coefficient	- 0.5 pcm /MW
Doppler effect in the fissile region at 1,500 °K	- 0.3 pcm / °C
Sodium temperature at core inlet:400 °C
Sodium temperature at core outlet:560 °C
Primary sodium flow in the core:2,800 kg/s
Sodium temperature at intermediary heat exchanger inlets:350 °C
Sodium temperature at intermediary heat exchanger outlets:550 °C
Secondary sodium flow in each system:740 kg/s
Water temperature at steam generator inlets:246 °C
Water temperature at steam generator outlets:512 °C
Steam pressure at superheater outlet:165 bars
Steam pressure at reheater outlet:34 bars
Water flow in each steam generator:210 kg/s



Joël GUIDEZ

*Director of the Phénix plant
since November 2002*

A reactor that's easy to live with



Pressurised water reactor specialists are always surprised how easy it is to run a fast reactor: no pressure, no neutron poisons like boron, no xenon effect, no compensatory movements of the rods, etc. Simply, when one raises the rods, there is divergence and the power increases. Regulating the level of the rods stabilises the reactor at the desired power. The very strong thermal inertia of the whole unit allows plenty of time for the corresponding temperature changes. If one does nothing, the power will gradually decrease as the fuel ages, and from time to time one will have to raise the rods again to maintain constant power. It all reminds one of a good honest cart-horse rather than a highly-strung race horse.

Similarly, the supposed drawbacks of sodium often turn out in practice to be advantages. For example, the sodium leaks (about thirty so far since the plant first started up) create electrical contacts and produce smoke, which means they can be detected very quickly. Again, the fact that sodium is solid at ambient temperature simplifies many operations on the circuits. More generally, because of the chemical properties of sodium, the plant is designed to keep it rigorously confined, including during handling. During operation, all this provides a much greater "dosimetric convenience" than conventional reactors. In particular, a very large part of the plant is completely accessible to staff whatever power the reactor is at, and the dose levels are very low.

Because of the very high neutron flux (more than ten times as high as with water reactors), there is great demand for experiments. These experiments are performed using either rigs inside carrier sub-assemblies or using special experimental sub-assemblies with particular characteristics. All experiments are run and monitored in the core like the other sub-assemblies. Since the origin Phénix irradiated around 1000 sub-assemblies, on which 200 were experimental sub-assemblies. It is true that the Phénix is not as flexible as an experimental water reactor, in which targets can easily be handled and moved. But, with a minimum of preparation - which is necessary anyway for reasons of safety and quality - numerous parameters such as flux, spectrum and duration can be adjusted to the needs of each experiment.

Furthermore, the reactor was designed by modest people who thought in advance of everything that would be needed for intervention on the plant: modular steam generators, washing pits, component handling casks etc. All of which has been very useful and has made possible numerous operations and modifications in every domain. All this has meant that a prototype reactor built in the early 1970s is still operational in 2004, and will continue so for several years yet.



GLOSSARY OF TECHNICAL TERMS
SPECIFIC TO THE PHÉNIX PLANT

Availability factor: The ratio between the electrical energy produced by the power plant over a given period of time and the product of its rated capacity (250 MWe) and the length of time concerned (= load factor), to which is added the time devoted to normal handling of the sub-assemblies (chance factors excluded) and R&D tests (handling of experimental sub-assemblies) and production time lost due to external causes (e.g. electricity grid failure). The availability factor expresses the power plant's ability to operate at the maximum of its potential for a sustained period.

Breeder: Breeder nuclides (usually uranium 238) are nuclides that can be directly or indirectly transformed into fissile nuclides (e.g. plutonium 239) by neutron capture. The term is also used to describe material containing one or more breeder nuclides, and by extension to the sub-assembly containing this material.

Breeding rate: Ratio of the number of fissile nuclei produced in the reactor core from fertile nuclei to the number of fissile nuclei destroyed during a given period of time. It is higher than 1 (otherwise the term used is burning).

Burn-up: Ratio of the number of atomic nuclei of a given element (or a given set of elements) that disappear by nuclear combustion to the initial number of nuclei.

Cladding failure or clad failure: Appearance of a defect in the cladding of a fuel pin, through which fission products can escape. A distinction is made between cladding failures that release only gaseous fission products and open ruptures which bring the fuel pellets into contact with the primary sodium

Controlled leak containment: The containment formed by the building walls, designed to confine the radioactive materials and kept permanently at lower pressure than the atmosphere outside, by extractor ventilators. This arrangement prevents the transfer of any contamination in the buildings to the outside except via the flow of extracted air, and this air is collected, filtered and checked before release through the stack.

Cover gas: In the reactor block, the space above the free surface of the primary sodium, in between the main vessel and the roof, is permanently filled with argon to prevent oxidation of the sodium. Fresh argon is injected via the slab penetration seals. The cover argon circulates in an monitoring and purification circuit

Decontamination: After washing a component, decontamination consists of removing contamination deposited on the component (products of erosion or corrosion of reactor structures, especially the fuel pin cladding). Decontamination is carried out in special pits using several stages of diluted acid baths. After decontamination, the component is dried and can then



be worked on (stripped down, repaired, dismantled etc.) in virtually normal conditions for a nuclear area. **Démantèlement (d'un assemblage, d'un composant)** : C'est l'opération de découpage d'un assemblage irradié ou d'un composant usé du réacteur. Elle permet de séparer les déchets de différentes catégories tout en les réduisant à une taille acceptable par les châteaux de transport et les installations d'entreposage provisoire ou de stockage définitif de déchets radioactifs. Dans le cas des assemblages combustibles, fissiles et fertiles, cette opération permet de récupérer les aiguilles afin de les expédier dans une usine de retraitement.

Delayed neutrons: Neutrons emitted by nuclei in an excited state formed during beta decay of fission products. The neutron emission itself is instantaneous; the observed delay is due to the preceding beta emission or emissions. The delayed neutron fraction (i.e. the ratio of the average number of fission events caused by delayed neutrons to the total number of fission events caused by prompt and delayed neutrons together) is essential to ensure control of a nuclear reactor. In the core of the Phénix plant, this number is 0.325%, or $\beta = 325$ pcm. This is the value which, by convention, sets the value of the "dollar" (\$), i.e. the level of reactivity required to make the reactor critical on prompt neutrons alone ("prompt critical").

Delayed re-heat cracking: This is a defect triggered in the root pass of a weld in the immediate neighbourhood of the contact zone, and which spreads radially between the grains when under stress in service. This only occurs with certain materials such as 321 type stainless steel, as it is caused by hardening of the steel due to fine precipitations of titanium carbide inside the crystalline structure. The hardening causes the plastic deformation capacity to be transferred to the periphery of the grains. For this to happen, the following conditions have to be met:

- a high operating temperature (> 475°C for 321 steel),
- a geometrical discontinuity at the weld root,
- strain hardening at the weld root showing significant shrinkage,
- heavy local load, which may be due to welding stresses,
- a defect in the weld root (e.g. a small shrinkage crack).

Dismantling: (of a sub-assembly or a component): Cutting up an irradiated sub-assembly or used reactor component. The different categories of waste can then be separated and reduced to an acceptable size for removal in transport casks and storage in provisional or definitive radioactive waste storage facilities. In the case of fissile and breeder fuel sub-assemblies, dismantling enables the operator to recover the fuel pins and send them to the reprocessing facility.

Dosimetry: Estimation, from individual measuring devices, of the dose absorbed by an individual or group of individuals. It is expressed in Sieverts (Sv) or milliSieverts (mSv) and, with collective doses, man-Sieverts (man-Sv). Until the early 1980s, the unit of measurement used was the rem (1 rem = 0.01 Sv = 10 mSv).



Dummy heat exchanger: A device used to replace an intermediate heat exchanger on a secondary cooling circuit that is not in use in power operation. It is basically an intermediate heat exchanger with no tube bundle. Its functions are to plug the slab penetration so as to ensure biological protection, and to provide an argon seal between the hot and cold pools.

Effective full power days or Equivalent full power days, EFPD: Ratio of heat energy produced in the reactor core (expressed in MWd) to the reactor's rated capacity (563 MWt). The number of EFPDs expresses a duration of irradiation of the sub-assemblies present in the core, regardless of their position.

Fast neutrons: The neutrons released on fission of a nucleus are emitted with high energy (around a million of electron-volts) and hence at high speed (about 20,000 kilometres per second). Using these fast neutrons without slowing them down (unlike slow neutron reactors or thermal reactors) requires a material with a high concentration of fissile nuclei (e.g. uranium 235 or plutonium 239) to offset the lower probability of fast neutrons causing fissions. Whence the absence of a moderator (e.g. hydrogen or carbon) in fast neutron reactors and the need for a coolant fluid that does not slow down the neutrons (e.g. sodium, helium, mercury or lead).

Incineration: Cf. transmutation.

Irradiation cycle: Time period between two fuel replacements. However, the core can be rearranged in the course of an irradiation cycle (in which case one speaks of successive loading plans). During the first years in operation, the average duration of an irradiation cycle at the Phénix plant was extended from fifty days at first to about ninety days. For the final irradiation cycles it has been set at 120 effective full power days (EFPD).

Irradiation experiment: Experiment in which selected objects or materials are irradiated in the reactor core for a defined period (a few months to several years) generally expressed in effective full power days (EFPD). The devices used are either experimental sub-assemblies, or rigs housed in carrier sub-assemblies.

Lagging: Insulation material placed around pipes and tanks containing a high-temperature fluid such as sodium, argon, nitrogen, water or steam, to prevent or limit heat loss. The materials used are poor conductors of heat, such as glass fibre and asbestos. Lagging may be partly or entirely removed from a pipe or tank to give access for inspection, repair etc.

Linear power rating: Thermal power produced per unit of active length of a fuel element. It is expressed in Watts per metre (W/m) or, more commonly, Watts per centimetre (W/cm).



Load factor: Ratio of gross electrical energy produced by the power plant (and of the equivalent energy supplied to the Marcoule facility in the form of steam) during a given period of time, to the product of the rated capacity (250 MWe) and the length of time concerned.

Negative reactivity trip (A.U.R.N.): Reactor shutdown automatically triggered by the three power range neutron measuring channels that monitor the core's reactivity when two of these channels measure reactivity below a value set at -10 pcm. This automatic response protects the reactor from accidents caused by largely insufficient cooling of the core, such as instantaneous breach of the connection between primary pump and diagrid.

Reactivity: In the reactor core where the chain reaction takes place, reactivity is the parameter reflecting departure from the critical state. Positive reactivity values reflect supercriticality, negative values sub-criticality. It is expressed in pcm (parts per hundred thousand) or as fractions of a dollar (\$).

Safety authority: The French nuclear safety authority ASN was originally the central department for safety of Nuclear Installations (SCSIN), formed in 1973 as part of the Ministry for Industry and receiving technical assistance from

- the Institute for Nuclear Safety and Protection (IPSN) (part of the CEA),
- the NSSS control Office (BCCN),
- the regional Directions for Industry and Research (DRIR) (the bodies responsible for industry and research in each Region).

The Phénix plant came under the DRIRs for Languedoc-Roussillon (Mining Services) and Provence-Alpes-Côte d'Azur for the nuclear part.

In May 1991, the SCSIN became the Nuclear Installation Safety Directorate (DSIN) while the DRIRs took on responsibility for environmental matters and took the name Regional Directions for Industry, Research and Environment (DRIRE).

In February 2002, the DSIN became the General Directorate of Nuclear Safety and Radioprotection (DGSNR), absorbing the body formerly responsible for radiation protection (Office for Protection against Ionising Radiations or OPRI), which in July 1994 had taken over from the Central Service for Protection against Ionising Radiations (SCPRI). Alongside this, the IPSN was entirely detached from the CEA and became the Institute for Radioprotection and Nuclear Safety (IRSN).

Sodium: An alkaline metal element, atomic number $Z = 11$, symbol Na. It is the seventh most abundant element in the Earth's crust. At atmospheric pressure, it has a melting point of 97.5°C and a boiling point of 883°C . Its density is 0.97. It oxidises spontaneously in contact with air (in the form of a high-temperature fire) and reacts violently with water. It has a fairly low neutron absorption cross section. Under irradiation, two isotopes are created: ^{22}Na , a β^+ and γ emitting isotope with a half-life of 2.6 years, and ^{24}Na , a β^- and γ emitting isotope with a half-life of 15 hours.



Sodium aerosols: Fine particles of various compounds of sodium (oxides, carbonates etc.) resulting from the combustion of hot sodium in air and dispersed in the form of opaque white smoke and deposited on surrounding surfaces. Sodium aerosols can also be formed by oxidation of hot sodium by traces of oxygen in the neutral gases used as cover gas in the sodium tanks (argon and nitrogen).

Specific burn-up: Total energy released by nuclear transformation of atoms when a reactor is operating (nuclear burn-up), per unit of fuel mass. Usually expressed in megawatt-days per metric ton (MWd/t).

Total loss of decay heat removal circuits (D.C.N.E.P.): This is a hypothetical accident in which all three independent secondary sodium circuits suddenly and simultaneously go out of action. In this case the residual power in the core would be removed mainly by convection through the reactor vessels to the emergency cooling system. The temperature of the reactor block would rise until the power removed by thermal radiation offsets the residual power, which would decrease over time. To ensure that all the reactor structures remain undamaged, this temperature must be below 720°C.

Transmutation: Transformation of one atomic nucleus into another by nuclear reaction. This may result in a different chemical element, or simply a different isotope of the initial element. This type of reaction provides a way of transforming long-lived radioactive isotopes into short-lived or stable isotopes in order to reduce the long-term radiotoxicity inventory of radioactive waste.

Washing: This operation is carried out in pits especially designed either for the irradiated sub-assemblies, or for the extractable components of the reactor block (intermediate heat exchangers, primary pumps, control rod mechanisms etc.). Washing consists of eliminating the metallic sodium by transforming it into soda or sodium carbonate. This is done by circulating first a wet inert gas, then water.



Fast Breeder Reactors in the world



Fast Breeder Reactors in the world

Country	Reactor	Thermal power	Electrical power	First criticality	Definitive shutdown
United States	Clementine	25 kWt	-	1946	1953
	EBR 1	1.4 MWt	200 kWe	1951	1963
	EBR 2	62.5 MWt	20 MWe	1961	1994
	Fermi	200 MWt	61 MWe	1963	1972
	FFTF	400 MWt	-	1980	1992
USSR	BR 2	100 kWt	-	1956	(1958)
	BR 5	5 MWt	-	1958	(1971)
	BR 10	8 MWt	-	1971	2003
	BOR 60	55 MWt	12 MWe	1968	
	BN 350	1000 MWt	130 MWe	1972	1999
	BN 600	1470 MWt	600 MWe	1980	
United Kingdom	DFR	60 MWt	15 MWe	1959	1977
	PFR	650 MWt	250 MWe	1974	1994
France	Rapsodie	40 MWt	-	1967	1983
	Phénix	563 MWt	250 MWe	1973	
	Superphénix	3000 MWt	1200 MWe	1985	1998
Germany	KNK II	58 MWt	20 MWe	1972	1991
Japan	Joyo	140 MWt	-	1977	
	Monju	714 MWt	280 MWe	1995	
India	FBTR	40 MWt	13 MWe	1985	
China	CEFR	65 MWt	23 MWe	2005	



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