Core physics

Controlling the core of a sodium-cooled fast reactor is very simple compared to pressurized water reactors. The power, due to thermal feedback coefficients, is stable for any given position of the control rods. When the power drops, the rods simply have to be raised to compensate for the fuel burn-up. Similarly at start-up, there is no adjustment of poison (boron in water reactors) to be made, nor vaporization and pressurization of the coolant. In the event of unscheduled shutdowns, there is no xenon effect and hot restarts are easier. Lastly, the high thermal inertia of the mass of primary sodium is an advantage.

This chapter outlines the composition of the Phénix core, its management, and monitoring and control mode.

Periodic or exceptional tests, performed from the beginning are also recalled. They show that results could be reproduced over time and that a good understanding of the core's physical phenomena and neutronic parameters could be achieved. These tests helped contribute to the validation of many calculation codes, as they evolved.

A great number of data were obtained for this type of technology in terms of monitoring, instrumentation and control.

Most of this experience of core follow-up, monitoring and control was successfully carried over to Superphénix, and will apply to future reactors.

Introduction

A lot of the criteria to be met in a nuclear facility, with respect to operation and safety, concern the reactor core as it is the hub of energy-controlled production.

Core physics, in its broadest sense, covers various factors such as its composition, operation, control, change over time not to mention the related instrumentation that ensures its maintenance in specific areas and which guarantees a predetermined safety level. The tests performed at start-up and during its life, also form part of the core physics, management and monitoring.

Core components

The core itself contained fissile fuel, surrounded by a fertile blanket, then neutron steel shielding limiting damage to the core's peripherals and the activation of secondary sodium at the intermediate heat exchangers. The fissile part for Phénix occupied a volume slightly greater than 1 cubic metre. Its structure was as follows:

a) Fissile element

The fissile element was plutonium in the form of mixed oxide (U, Pu) O_2.

This fuel was contained in about one hundred sub-assemblies each having 217 pins, themselves consisting of a pile of oxide pellets enclosed in stainless leaktight pins.

Each of the bundles of pins was placed in a stainless steel housing of hexagonal cross-section (TH) making up the sub-assembly. Each pin was surrounded by a wire helicoidal forming a space, with respect to its neighbours, for the circulation and stirring of sodium.

The sub-assembly consisted of a spike for its mechanical and hydraulic connection to the diagrid. The latter, supplied with sodium by the primary pumps, performed as a distribution box to enable suitable cooling of each sub-assembly.

Fuel sub-assembly

b) Fertile element

The fertile element was depleted natural uranium. It was also implemented in the form of oxide pellets housed in about one hundred sub-assemblies, each comprising 61 pins.

The fertile sub-assembly had the same outer dimensional characteristics as the fissile sub-assembly.

c) Lateral neutron shielding

The lateral neutron shielding consisted of several rows of steel sub-assemblies having the external shape of the fuel sub-assemblies. These sub-assemblies were plugged into the diagrid and cooled by forced circulation of sodium. Several rows of steel billets outside the diagrid were added, cooled by natural convection.

The core also included six control rods in the fissile zone made of boron carbide neutron absorber and a safety rod (Complementary Shutdown System: CSS) located in the centre of the core.

Furthermore, in the diagrid area representing the lateral neutron shielding, 41 cavities were dedicated to storage, to decrease the residual power of irradiated fuel elements.

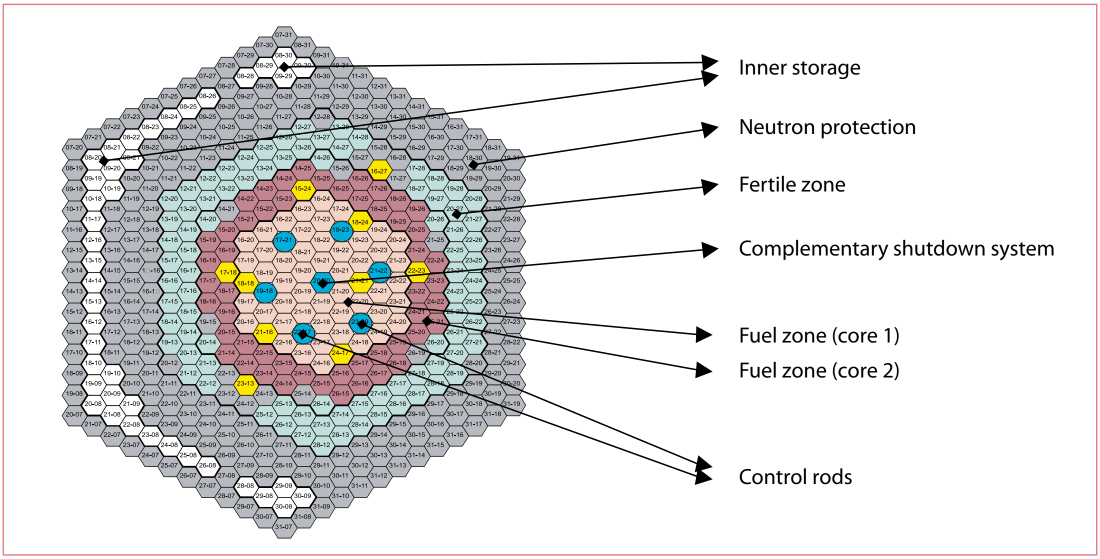


Figure 1: Plan of Phénix's core

Note that for specific irradiation needs, the core could include some experimental sub-assemblies (incore irradiation and measuring device / DIMEP), different from the standard sub-assemblies described above through a central channel that can accommodate a rig.

The neutron calculations performed, as forecasting or monitoring, obviously took all the core components into account (spatial distribution of the various elements' concentrations).

Core operating conditions

They were, of course, related to the operating conditions of the power plant. It should be remembered that after 1993 Phénix operated at 2/3 of its rated capacity, i.e. a reduction from 563 MWth to 350 MWth. The Phénix reactor was always operated as a base-load plant, which means that there was no automatic control of the power level upon the grid request.

The optimization of the core's thermal efficiency led to designing two different zones of Pu enrichment as well as 16 flow areas.

Core-1 was the lowest enrichment zone, core-2 the highest enrichment zone (see Chapter on the fuel element). A core-1 sub-assembly contained about 7 kg of Pu 239 compared to 9 kg a core-2 sub-assembly (for equivalent Pu 239 masses).

Contributions from the two core zones to the total power were roughly equivalent.

At 2/3 power for example:

- 181 MW for core-1
- 143 MW for core-2
- 24 MW for the fertile blanket.

As for the flow areas, they allowed high average heating of the core (150°C) while levelling and limiting the maximum temperature of sodium in the hottest channels of each area.

Overall, the criteria chosen to carry out the optimization of the core were:

- maximum linear power for the pins of 450 W/cm,
- maximum clad temperature of 700°C,
- The quantity of clads exceeding 650°C

and 670°C were limited to 5% and 0.1% respectively of the total number of clads.

The main neutron characteristics of this core were as follows:

• The neutron flux at the centre of the core, at the power of 350 MWth, was 4.5×10^{15} n/cm^2/s (7.1×10^{15} n/cm^2/s at 563 MWth).

• 82% of the neutrons had an energy greater than 0.040 MeV, 35% an energy greater than 0.5 MeV and 15% an energy greater than 1.35 MeV.

• For 1,000 neutrons produced by fission in the entire core, 342 generated a new fission, 615 were captured and 43 left the core.

• During a cycle of 120 EFPD, the core consumed 83 kg of U8, 2 kg of U5 and produced about 15 kg of Pu. The production and amount of Pu were highly dependent on the choices made in terms of blankets.

The reactor could even be a burner if necessary.

• The six control rods represented a negative reactivity of about $20 which helped compensate $7 representing fuel wear during a cycle, $3 representing feedback effects between 0 and 350 MWth and always have a safety margin of $10 with respect to handling error.

Each control rod contained 6.7 kg of B_4C enriched by 48% in ^{10}B.

The dollar is the fraction of delayed neutrons (β). Reactivity is expressed in pcm. The value of the dollar, in terms of pcm, depends on the type of fuel. In a fast reactor, a dollar represents about 325 pcm.

Core monitoring

Monitoring of the core included systems dedicated to instrumentation and control and/or safety.

The core's main monitoring systems included:

a) Nuclear measurements

The set of neutron channels ensured the operation and safety of the power plant from the shutdown level, all rods down, beyond the rated power. The dynamics of the control extended over 11 decades and required three types of detectors, the operating diagrams of which overlapped. There were 2 pre-start neutron measuring channels located in the vessel, and located under the vessel in line with the core, 3 intermediate channels and 3 power range channels.

These channels were completed with 2 dedicated control channels, a reference channel and a gamma measuring channel, all located under the vessel.

The electronics associated with these channels delivered values such as power, reactivity or neutrons doubling time to which safety thresholds were related (alarm and scram).

b) Processing core temperatures

The processing of core temperatures was based on measurements of the sodium temperature at the outlet of the core fissile sub-assemblies and at the core inlet. The purpose was to detect an abnormal rise in the temperature of each sub-assembly, average heating of the core, clad temperature or core inlet temperature. For this purpose, each sub-assembly had 2 thermocouples at its outlet located in a thimble (placed 75 mm above the sub-assembly outlet). Three pairs of thermocouples measured the temperature of the sodium at the core inlet (1 pair per primary pump). Thermocouple pairs were used for safety system redundancy. The scanning rate of each sub-assembly was every 1.5 s.

The use of sodium/stainless steel thermocouples helped to follow the temperature noise, if necessary, with considerably lower time constants and cut-off frequencies.

Safety actions included:

- **plugging detection**

When the heating of a sub-assembly exceeded a high threshold SH, above the nominal value, an alarm was generated, and a scram was initiated if it exceeded a very high threshold STH (SH = 6°C, STH = 12°C).

- **core heating**

When the average heating of fissile sub-assemblies exceeded the nominal value by 3% an alarm was generated and a scram initiated if it exceeded the nominal value by 10%.

- **clad temperature**

The TRTC (core temperature processing system) computed an estimate of the clad temperature Tg, using parameters related to the loading in progress and core input temperature. An alarm was generated if Tg exceeded 650°C for more than a minute.

- **clad hot spot temperature**

Similarly to the previous case, the TRTC also calculated an estimate of the clad hot spot for each cycle.

An alarm was generated if it exceeded 700°C for more than one minute, followed by a scram if the duration exceeded 6 minutes.

- **core inlet temperature Te**

A rapid trip (motor-driven shutdown as opposed to the scram by control rod drop) was generated when the max. Te threshold was exceeded.

The TRTC accuracy and reliability led to setting up monitoring of the heating of sub-assemblies during irradiation cycles, which helped highlight small changes in flow rates related to pin swelling. These measurements confirmed the steel 316 swelling phenomenon, later characterized by means of post-irradiation examinations (see chapter on fuel performance).

c) Clad rupture detection systems (DRG)

Two types of measurements were carried out to reveal the deterioration of a clad:

- detection of fission products that emit delayed neutrons in sodium samples,

- measurement of cover gas activation by argon sampling.

The sodium samples were taken at the inlet of the six intermediate heat exchangers and circulated in six measuring blocks fitted with helium 3 metres.

The count rates were compared with two thresholds representing the generation of an alarm and tripping of a scram respectively.

The argon samples were analysed by several ionization chambers and the measurement results recorded without any safety action.

d) Failed fuel localization systems (LRG)

This stage occurred after the detection of a clad failure and could require returning the reactor to a limited power level after a possible scram initiated by the DRG.

The LRG system allowed selective sampling of sodium at the outlet of sub-assemblies and the analysis focused directly on the sodium sampled as well as the dissolved gases it contained, the latter being very useful because it often allowed early detection of clad failures before they developed into open fractures and released fission products. These analyses ultimately led to identifying a group of three suspect sub-assemblies.

Most of the time the history of these sub-assemblies helped in finding the faulty sub-assembly with a clad failure. If not a few handling operations followed by a partial power build-up might have been needed to identify and unload the faulty sub-assembly.

15 clad failures occured during Phénix life, including 8 on experimental sub-assemblies. This figure is to be compared with the number of pins to have spent time in the core i.e. approximately 180,000. It is also worth noting that all clad failures were detected and localized without difficulty by the DRG/LRG systems (see Chapter on experience feedback from fuel elements).

Changes to the core

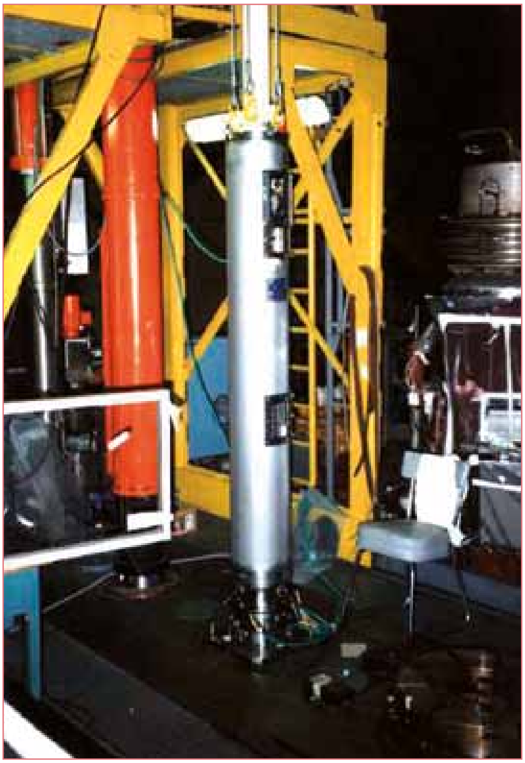
The inherent flexibility of fast reactors led to several significant changes to the core configuration.

The most significant was linked to replacing the central fuel sub-assembly with a Complementary Shutdown System (CSS).

This absorber sub-assembly, placed in normal operation above the fissile zone, was designed to drop even in a very distorted core configuration. The negative reactivity introduced was enough to stop the chain reaction and keep the reactor shutdown condition at a temperature below 450°C.

Other changes aimed at either optimizing core management or specific experiments on a large core area. Among these, there was the PAVIX experiment which consisted in introducing 6 sub-assemblies with a fertile axial slice to study the concept of heterogeneous axial cores.

In terms of core management, it should be noted that when operating at 2/3 power, the transition to a so-called small core configuration consisted of replacing 18 corner sub-assemblies of the sixth ring with steel sub-assemblies. The purpose was to maintain representative irradiation characteristics when operating at reduced power.



Complementary shutdown system

Core management

To compensate for fuel burn-up, the core sub-assemblies were renewed at a rate of one sixth per cycle of 120 EFPD (Equivalent Full Power Days, i.e. 563 MWth).

During a sub-assembly's irradiation, which lasted about 720 EFPD, it was moved several times according to the Power Plant's fuel management code. It had specific modules adapted to the management of the core allowing it to perform all the neutronic, thermal and hydraulic calculations for core loading in progress and forecast the characteristics of subsequent loading to meet the operation and safety criteria.

Please note that during the life of the power plant, the time that fissile sub-assemblies spent in the reactor went from 6 x 56 EFPD to 6 x 120 EFPD. The concept of specific burn-up used in the early life of the power plant was completed by the concept of damage received by the sub-assembly's structures (clads and hexagonal tubes). The latter quantity expresses the average number of displacements per atom. It depended on the fluence, neutron spectrum characteristics and material involved.

Taking this parameter into account in sub-assembly management increased the specific burn-up of sub-assemblies from 50,000 MWd/t to 90,000 MWd/t at the centre of the core and 115,000 MWd/t at the periphery. In addition the number of sub-assemblies (increase of the core size) and the enrichment of zone 1 were increased.

The burn-up fraction expresses thermal energy in MW.day extracted from a tonne of fuel. It characterizes the degree of fuel use. In a pressurized water reactor, this rate is around 45,000 MWd/t. As an order of magnitude, a burn-up fraction of 75,000 MWd/t corresponds to a burn-up fraction of all fissile atoms of around 9%.

As far as the management of fertile sub-assemblies is concerned the main parameter limiting their time spent in the reactor was the bending of the TH as a result of irradiation. This time depended on the location of the sub-assembly and could vary from 670 EFPD for a sub-assembly located in the 6th ring to 1,700 EFPD for an assembly located in the 9th ring or beyond.

Note that the maximum value was increased from 1,160 to 1,700 EFPD following the transition of steel 316 cold-worked to 316 Ti (see Chapter on the fuel element).

Core control

In addition to the conventional operations : divergence, power build-up and possible power level adjustment, core control helped offset the loss of reactivity due to burn-up (approximately 20 pcm per EFPD), loss of reactivity due to the transformation of plutonium 241, fissile, into americium 241, less fissile (about 0.3 pcm per calendar day).

The control also had to take into account the neptunium effect (about 70 pcm at 350 MWth) which impacted the divergence height and full reactivity between cold shutdown and a given power level. To understand this phenomenon, it should be remembered that a fraction of the neutrons is captured by uranium-238 which turns into uranium 239. It decays by β- disintegration β in neptunium 239 with a period of 23 minutes, which in turn gives plutonium 239 byβ- disintegration β with a period of 2.4 days.
In balanced operating conditions, the stock of neptunium 239 in the core was constant at its saturation level and for each neutron captured, on average one plutonium 239 nucleus was created.

This continuous creation of fissile matter was included in the burn-up. On the other hand, upon restart after a sufficiently long shutdown (a few neptunium 239 decay periods), all the neptunium was transformed into plutonium 239 constituting an injection of reactivity that had to be taken into account for the provisional calculation of the divergence hight (lower hight).
Then during the first days of operation, the captured neutrons replenished the stock of neptunium to its saturation level without there being an equivalent simultaneous creation of plutonium-239. During this transitional phase, the burn-up was several times higher than the balanced burn-up.

The control also included actions such as those designed to meet specific tests (see below) or "quash" a hot spot by shifting a control rod.

Finally, it is worth remembering that the stability of the reactor and the safety of its control came from three factors:

• existence of delayed neutron fraction (β = 325 pcm) which, for the reactivity insertions involved in control or test purposes, gave the reactor a builduptime greater than 2 minutes, making its control easier.

• DOPPLER counter-reaction which prevented any increase in temperature of fuel and therefore power.

• thermal counter-reactions of structures that were generally negative.

Physical tests performed

Many tests were carried out when the facility was started up and throughout its life to ensure compliance with specific safety criteria, confirm design choices or improve knowledge. The most common neutron tests included:

• **Reactivity testing of the absorbent control rods**

Different methods (testing per period, by compensation, inversion of kinetic equations, source multiplication with or without correction of spatial effects) were widely used and met the requirements of normal operation.

They were supplemented by the REACTIVIX test programme in 1995 based on measurements by control rod drop to obtain an absolute reactivity standard to be used as a reference. This type of test was also extended to the CSS when it was set up in 1997 and showed that the CSS drop introduced a negative reactivity of 3.5 $, ensuring compliance with the new safety criterion.

The improvement in accuracy obtained helped efficiently to compare neutron calculation tools with measurements and therefore be very useful for the qualification of codes. From the every day point of view, it turned out that the swing method offered a bias lower than 10% compared the reference method and given the simplicity of its implementation, it continued to be used regularly by the operator. Measuring tests by movements of control rods were carried over for the new configuration in the power plant's final tests.

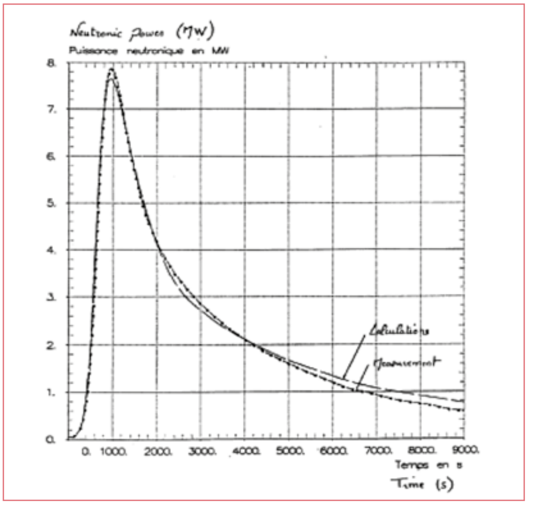
• **Self-stabilization tests**

They involve injecting low reactivity (0.1$) in a zero-power critical core by control rod withdrawal to study its natural behaviour and the overall thermal counter-reactions of loading in progress. The power reached 8 MW in 15 minutes, and then the thermal counter-reactions introduced negative reactivity which brought it back down to a balanced level of 50 kW in less than 3 hours.

• **Measuring tests of coefficients k, g and h**

These coefficients express the reactivity sensitivity to the three parameters characterizing an operating point of the reactor (core inlet temperature, average heating of the core and core power). To measure them, a series of primary flow (-10%), secondary flow (-10%) and power (-10%) small transients were carried out. The 3 operating conditions obtained provided a system of 3 equations with 3 unknowns, which allowed coefficients k, g and h to be calculated. They characterized the thermal counter-reactions.

All three of these coefficients were negative, although sensitive to the power level and loading pattern, with a remarkably consistent behaviour throughout the reactor's life.



Self-stabilization test

The following orders of magnitude were obtained:

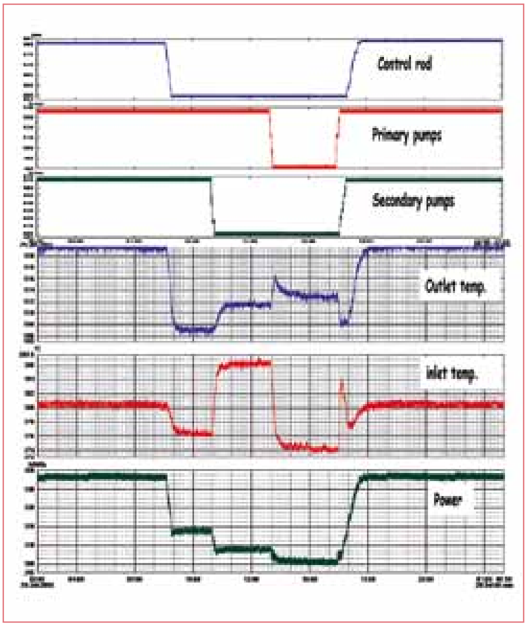
k # -2.5 pcm/°C,
g # -1.8 pcm/°C,
h # -0.5 pcm/MW.

- **Doppler coefficient isothermal measuring tests**

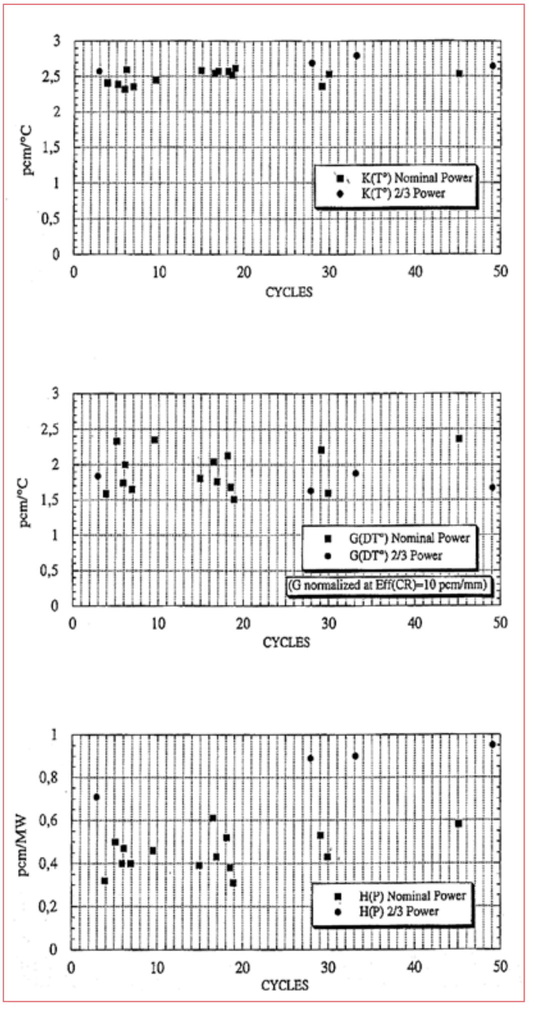
These tests involved measuring changes in reactivity due to the control rods in order to keep the critical state of an isothermal core with different temperature levels.

The kiso zero power isothermal counter reaction coefficient was thus obtained. It incorporated two physical phenomena relating to the modification of macroscopic cross-sections (core expansion and DOPPLER effect). Several kiso measurements could separate these two effects and deduce the corresponding k1 and k2 coefficients from them.

k1 # - 1.7 pcm ; k2 # - 826 pcm ; kiso (300°C) # - 3.14 pcm/°C



Measurement of k, g and h coefficients



Follow-up of k, g and h

- **Residual power measuring tests**

This test consisted of measuring the decrease in residual power (as a function of time) by an accurate plant-wide calorimetric balance, and comparing it to the forecasted calculations (see Chapter on residual power removal).

- **Monitoring reactivity balance over time**

This monitoring helped, in identical operating conditions, to monitor reactivity over time or, for different operating levels, to measure the full reactivity effect of thermal counter-reactions.

This enabled any drift of parameters k, g and h to be highlighted and to schedule the tests required for their reassessment.

The power coefficient h, in particular, was sensitive to fuel restructuring occurring in the new sub-assemblies.

- **Measuring the effect of neptunium and reactivity loss during the cycle (burn-up)**

The gradient of the curve in the figure below gives the reactivity loss due to burn-up, about 20 pcm/EFPD. The breaks visible in the figure correspond to the neptunium effect occurring during various restarts (gain in reactivity due to the transformation of neptunium 239 into plutonium 239 during the shutdown). The amplitude of each jump depended on the steady state obtained at the time of the previous shutdown and the duration of the shutdown.

- **Measuring the reactivity effect of fuel ageing**

Given its low value (0.3 pcm / calendar day) and the decay time of plutonium 241 (13 years), it was difficult to isolate this effect during reactor operation.

On the other hand, during a long-term shutdown of the reactor (several months), it was possible to measure this effect through the change in critical hight.

- **Final tests**

Many neutron measurements were taken during final testing, to round off the knowledge in this field. They are presented in the relevant chapter.

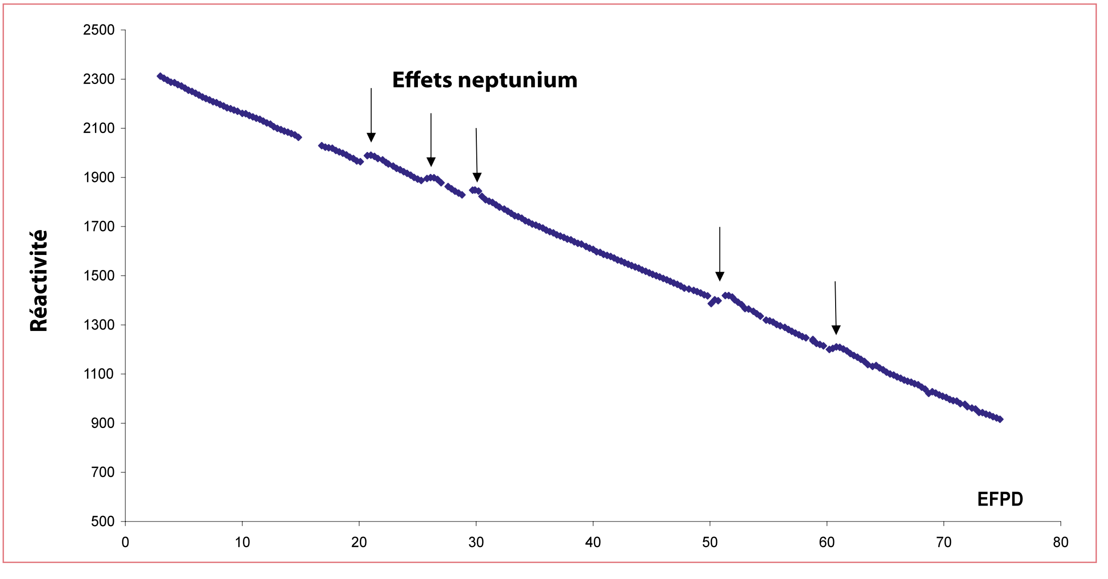
Gains for the reactor system

Experience feedback from Phénix includes technology advances that will be reused in the future development of fast neutron reactor.

These mainly include:

a) Neutron control

The use of neutron chambers located under the safety vessel was validated with respect to operation and safety, however it was noted that the sodium layer located between the core and the bottom of the main vessel could affect the fine measurements made for experimental purposes.



Measurement of burn-up and neptunium effect

The attenuation of the neutron flux due to sodium absorption varied according to the temperature and cast additional uncertainty on these measurements.

The solution, already implemented at Superphénix, was to provide for neutron guides to channel neutrons through the sodium.

In addition, the use of incore pre-start-up chambers helped validate high temperature chamber concepts.

b) Processing core temperatures

The very principle of TRTC operation required regular campaigns of thermocouple inter-calibration. They also allowed monitoring of these sensors to anticipate failures and provide for their replacement.

Moreover, it was shown that specific types of incidents (untimely control rod withdrawal) could be detected first by the TRTC, which will motivate, in the future, an improvement of analysis frequencies and a reduction of the time constant for thermocouples (around 15 seconds at Phénix).

c) Core management

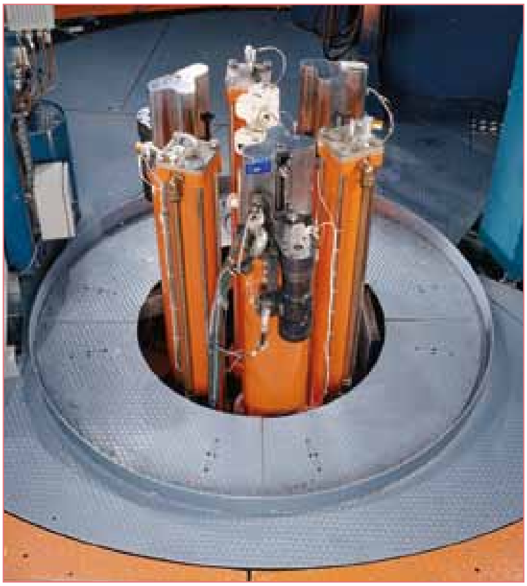
The measuring and calculation equipment available enabled individual management of sub-assemblies and authorized management of the core adapted to increasing the performance of materials.

d) Core control

The control and related test equipment showed high reliability and were never found at fault, either in daily operations or during specific tests.

e) Physical tests

The physical tests performed showed the flexibility of fast cores and excellent reproducibility of the results obtained in similar situations.



Control rods

Several tests were similar to laboratory experiments in terms of the accuracy of measurements (0.5°C on calorimetric balances or a few pcm on neutron balances). These performances related to the design of fast reactors, helped consistently improve modelling and calculation tools dedicated to this reactor type.

Conclusion for the future

The Phénix reactor's operation since 1973 confirmed that the management and control of a fast reactor core would not cause any particular difficulties and that this type of reactor would be suitable for flexible use. The substantial changes made during its life, to the geometry of the core, loading patterns and operating conditions were managed perfectly well in terms of predictive calculation tools as well as operating and testing equipment. It should be remembered that about a third of the sub-assemblies were experimental ones.

The operation of the reactor in its various configurations showed high stability of the operating parameters. As such, the measurements made confirmed the influence and magnitude of thermal counter-reaction coefficients which, by taking into account any increase in sodium temperature or reactor power, provides fast reactors with self-stabilization. Similarly, the absence of xenon effect, and the fact that there was no adjustment of a poisonning required in the coolant (boron for PWRs) made it easier to take into account transients and hot restarts.

Finally, this operation validated the core management and monitoring methods, the main aspects of which were successfully carried over to Superphénix.

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