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ONE YEAR OF PHENIX OPERATION *

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SUMMARY

This report describes chronologically the operations carried out since the first coupling of the Puenix Station to the grid on Dec. 13, 1973. It describes briefly the last of the tests before putting into service, interspersed with periods of work and which were terminated July 14, 1974. From this latter date, the system was in use with developments taking place without major incident as of the date of editing of this report. This chronological description is accompanied by diagrams for times up until the end of March 1975, production statistics, main operating variables at different states, and a summary of the incidents that have occurred during use. From these elements charge rates and availability of 76% and 81% with a gross output greater than 45% are found.

The chief subassemblies of the station are reviewed with details of possible incidents, modifications made, and partial conclusions that can be drawn. In particular, problems of fuel, block reactor structures, sodium pumps, clad rupture detection, steam generators, sodium circuits, and energy production installations are enumerated. In general, performance of the installation and values predicted have been in good agreement.

Fuel handling was the objective of a special study, rather brief because of the few incidents encountered. The handling techniques used are described chronologically and the modifications introduced are reviewed. Dismantling and transfer in the processing plant of the first irradiated fuels are indicated.

The conclusion makes it possible to emphasize the contribution of the Phenix station to the furtherance of the French program on fast-neutron reactors.

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I. INTRODUCTION

The Phenix power station is located beside the Rhone, near the CEA Center at Marcoule, in the Department of Gard. This station has a fast-neutron reactor, is cooled with sodium, has a predicted power of 563 MW-th, and a standard turbo-alternator unit of 250 MW. Construction started at the end of 1968. The first divergence was on Aug. 31, 1973, at least 5 years after the first breaking of ground.

A number of operation tests on apparatus, sub-assemblies, assemblies, and finally the whole plant were made during the construction and up until final start-up time on July 14, 1974. The total construction, tests, and operation followed the program that had been planned, and the station is now connected with the French Electricity Grid the same way as a standard energy station.

The present report indicates the considerable viability of the group of equipment, the great availability of the station, and the validity of the experiments on which the program was based.

II. CHRONOLOGICAL DEVELOPMENT

1. Final Tests

The station was first coupled to the grid on December 13, 1973. The 20 MW-e power was maintained for about 20 hr. Immediately after that and up until the end of January, a series of tests was made with special emphasis on natural convection, voltage deficiencies, and various thermo-hydraulic experiments. The installation was then brought to nominal operating temperatures, and 60% of the projected power was maintained for about 4 days.

An operating incident with the control rods interrupted the tests for 15 days and required dismantling, modifications, and new tests on the rods involved.

On Feb. 14, after the 84th divergence and the 18th coupling, the installation was again shut down,because of a small sodium leak in supplementary piping, for about a week. Startup was made again on the 25th, and the power was brought to 60%.

On March 12 the nominal power was reached and even slightly exceeded, i.e. 568 MW-th and 251 MW-e.

Over a period of about a month and a half a series of neutron tests were made to obtain more detailed physics data on the core. These experiments required three handling experiments. During this time some operation was carried out, i.e., certain programs and the last of the construction. Other accomplishments, equally important, were modifications required as a result of the preceding tests, in particular on the turning plug of the reactor, the water-steam circuits, and the accessory diesels.

From mid-May to mid-June that last series of power tests was made on the operation---dynamic tests, turbo-alternator tests, unloading tests, tests on operation with perturbance or nonequilibrium. This period was ended with a 2-week duration test at nominal power. The test period was ended June 16.

An operating period of one month led to a definitive design for operation of the whole station. Several supplementary tests were made, in particular an almost complete remaking of the hydraulic controls of the steam generator valves. A final handling experiment restored the power core.

The station was declared to be in final use on July 14, 1974.

2. Startup Operation

Operation began by a period of 2 weeks at nominal power to complete the first irradiation cycle. The operation of the station was divided into cycles of irradiation of 56 JEPP (Jours Equivalent a la Puissance de Projet, 563 MW = equivalent days at project power, 563 MW), followed by a period of fuel handling for replacement of 1/6 of the core. After the second handling series, lasting 5 days, irradiation cycle No. 2 was started August 3. Several minor incidents around August 15 somewhat disturbed this cycle, which was interrupted Sept. 20 by a sodium leak on a secondary circuit. The secondary circuit involved was isolated, and the station started up again two days later on two loops and therefore two steam generators. This operation was completely stable and very satisfactory at 2/3 of the power until Oct. 13. On this date cycle 2 was stopped, and handling experiment No. 3 was carried out in preparation for start of cycle 3 on Oct. 18.

This stoppage for the handling program was used for repair, putting into order, and maintenance on the secondary cycle in which the preceding leak had been detected. Cycle 3 was then brought to nominal power under normal conditions. Several incidents the first of December and the last week in December disturbed only this operation, which was ended Jan. 5 at the end of the cycle which had been extended to 71 JEPP, carried out in 77 calendar days.

Mention should be made of a test on increase in maximum power made in December, permitting reaching a power of 591 MW-th and 267 ME-e.

After handling series No. 4, irradiation cycle No. 4 was started Jan. 13. Operation with a single untimely operation on Feb. 10 until the end of the cycle Feb. 23 was carried out. This shortened cycle resulted in reestablishing the average of 56 JEPP per cycle since the beginning.

After the handling procedures, No. 5, irradiation cycle No. 5 started up again Feb. 28. This cycle was disturbed March 14 by a sodium leak similar to that of September 1974. Operation of the station was ensured for 9 days with two secondary loops, reestablishment of the situation being made March 24 and the usual power reestablished until the end of March, when this report was prepared.

III. STATISTICS

In this chapter are grouped the statistics and diagrams which can be used to give concrete indications from the preceding chapter and to judge objectively the operation of the Phenix station since the beginning and since the final putting into operation.

1. Diagram of Progress (Fig. 1)

The chief stages are shown on the diagram. After the first coupling, a series of tests was made up to nominal temperatures, then the series ending with nominal power in mid-March 1974, and finally a month at nominal power for final tests and duration. After one month of putting back into order and of experiment, the actual operation was started. There was evidence that operation would be with a minimum of disturbance, permitting reaching the excellent conditions given in detail in the following chapter.

2. Production Statistics (Fig. 2)

The following data were obtained on March 31, 1975, at midnight:

Production since start

Thermal power produced by reactor = 139,282 MWd

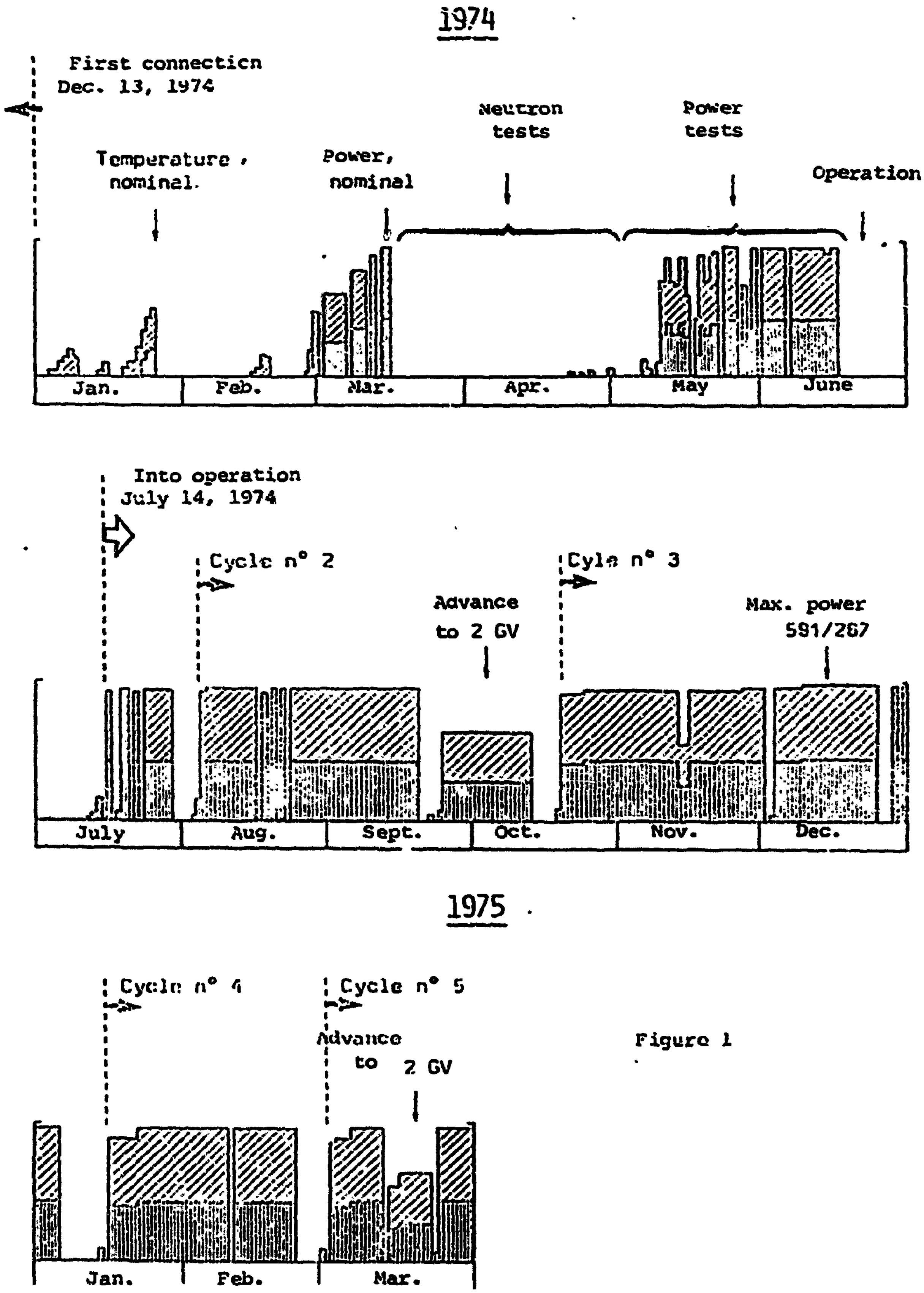
1974
First connection
Dec. 13, 1974
Temperature, nominal.
Power, nominal
Neutron tests
Power tests
Operation
Jan.
Feb.
Mar.
Apr.
May
June
Into operation
July 14, 1974
Cycle n° 2
Advance to 2 GV
Cyle n° 3
Max. power
591/267
July
Aug.
Sept.
Oct.
Nov.
Dec.
1975
Cycle n° 4
Cycle n° 5
Advance to 2 GV
Jan.
Feb.
Mar.

Figure 1

Electric power, gross = 1,437,010 MWh; net = 1,337,887 MWh

Reactor at power greater than 10 MW = 7,753 hr

Turbo-alternator group coupled for 6,246 hr

Secondary circuits in sodium for 16,000 hr

Fissile burn-up rate, max = 5.13%; average = 3.65%

Mass burn-up, max = 42,000 MWd/t; average = 29,700 MWd/t

Production since July 14, 1974

Thermal power produced by reactor = 112,981 MWd

Electric power, gross = 1,196,400 MWh; net = 1,115,303 MWh

Charge rate = 76.40%; disposal rate = 81.76%

3. Operating parameters

Table 1

Operating parameters		Planned	Reached	Max
Thermal power	MW	563	562	591
Electric power gross	MW	250	254	267
Electric power net	MW	233	238	250
Yield gross	%	44,4	45,20	45,18
Yield net	%	41,4	42,35	42,30
Max. clad temperature	°C	700	670	690
Sodium temperature at core inlet	°C	400	396	395
Sodium temperature at core outlet	°C	560	554	563
Primary sodium flow	kg/s	2.760	3.000	3.000
Sodium inlet temperature GV	°C	550	540	550
Sodium outlet temperature GV	°C	350	341	342
Secondary sodium flow	kg/s	2.210	2.400	2.400
Water inlet temperature GV	°C	246	237	240
Temperature of superheated steam	°C	512	509	516
Temperature of re-superheated steam	°C	512	515	523
Water flow	t/h	751	752	788

Table 1 gives the chief operating values from the initial project and during the progress to corresponding power, thus permitting a valid comparison. Also shown are the same values during the progress to maximum power made in December 1974 for test.

4. Stops, Untimely or Because of Incidents

Since the number of unplanned shutdowns of a station is a basis for evaluation, we have compiled (Table 2) a list of the 27 unscheduled shutdowns from start-up time to the end of March 1975. We find that almost half these stoppages occurred during startup of the station and that ten of them were in the steam installation. The number of disengagements per automatic shutdown was 22.

Table 2

Causes of stoppage	Total No.	of which No. at startup
Control		
- Calculator relay	2	
- Calculator breakdown.............	1	
- Sodium pump disengagement	4	1
- Miscellaneous	3	1
Steam		
- Leaks in components	5	3
- Regulation problems	5	4
Sodium leak	2	
Lattice distortions......................	1	
Handling problems	4	3
Total	27	

IV. GENERAL OPERATION

Nine months have passed since the startup of the Phenix station. Despite the short time, we can already make some interesting statements and reasonably expect confirmation in the coming months.

The whole plantwas shown to have an almost perfect stability in both the reactor and the power-producing sections. Operator intervention was seldom necessary and consisted essentially in several daily adjustments of the main variables.

Startup and rise in power were rather touchy operations in the first tests, the main reason for this being the behavior of the reactor block as a result of the temperature gradients supported by these structures. Experience permitted progressively releasing an improved and somewhat traumatic procedure, while decreasing the total length of the operation. Actually, cold startup from the holding state---the sodium at 250°C--- requires 24 hr to reach full power. It seemed possible to us that this time could be decreased by several hours (see Fig. 2). In the case of an unscheduled shutdown followed by an immediate re-startup, without significant cooling, full power is generally reached 5 to 6 hr after shutdown (see Fig. 3).

Programmed shutdown from full power was, like startup, rather touchy and the initial procedure consisted in a power drop to 60%, then a rapid shutdown with simultaneous disengagement of the turbo-alternator group. The procedures put into operation progressively permit [word not in dictionary] and a guided shutdown to zero reactor power, with turbine disengagement at 20 MW-e without causing disturbances in the reactor. An example of this procedure and of the general flexibility of the installation was given on March 23 1975. The station was operating at this time with two steam generators, and shutdown had been progressive with the reactor remaining critical. The secondary loop No. 2 had been filled and put into service with the reactor being kept at about 10 MW. Restartup was then undertaken, the operation after shutdown having lasted a total of 30 hr.

In the case of a normal planned shutdown from full power, it is possible to reach the sodium temperature of 250°C necessary for fuel handling in 6 hr, with the steam generators remaining in water. This shutdown takes 12 hr, the steam generators being drained but cooled by natural air convection.

On the subject of steam generators, the extreme convenience of cooling with natural convection, caused by opening the upper and lower hatches of the caissons. Cooling of the core is then handled with precision, and maintenance of the temperature required for the shutdown period is easy. Further, this possibility of cooling permits completely disconnecting the reactor and the secondary circuits of the steam installation, which makes possible the always numerous interventions on this installation.

On the other hand, these same steam generators impose use regulations that are very constraining on the temperatures. During fillings, the temperature differences between the two fluids should not be greater than

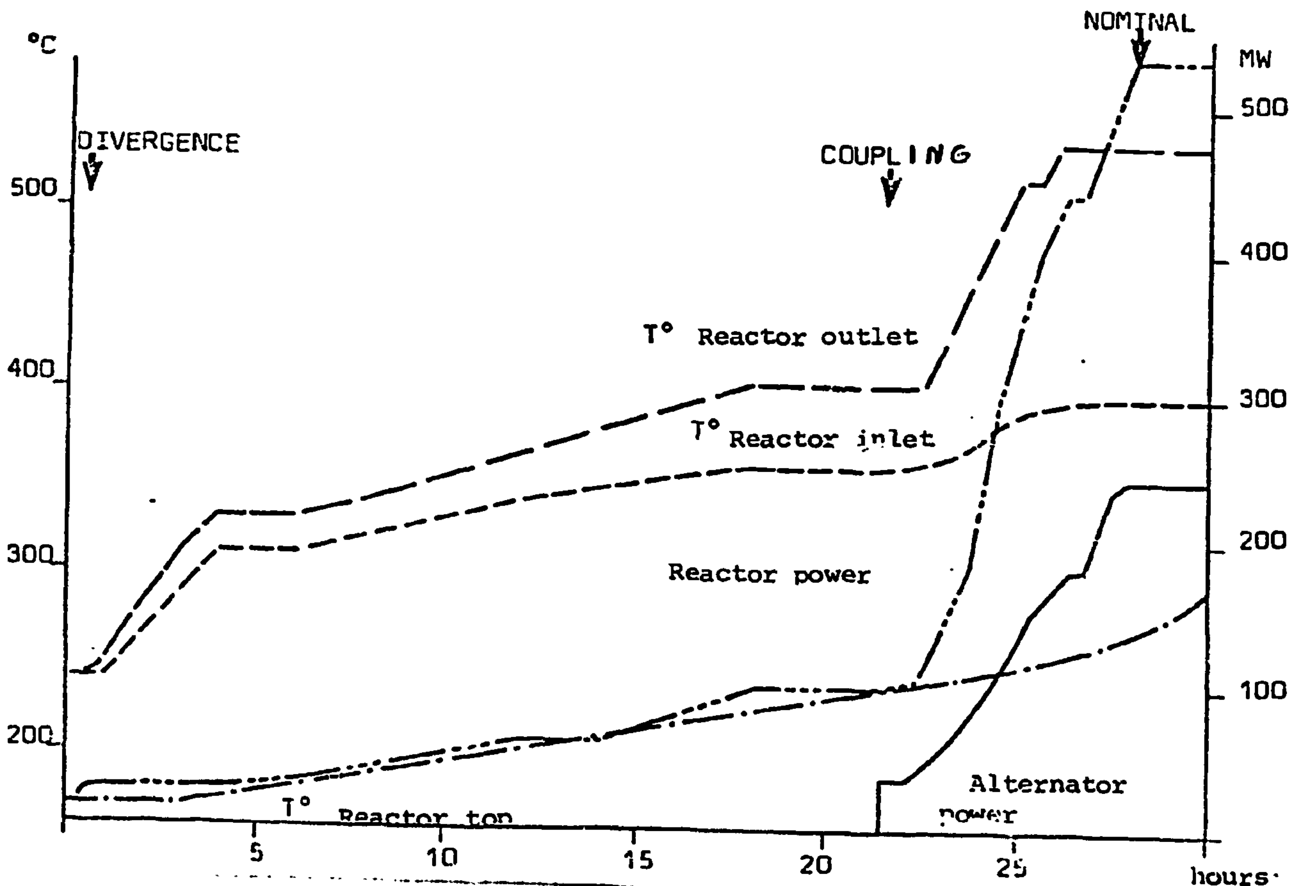

Fig. 2. Cold startup after holding, October 20, 1974

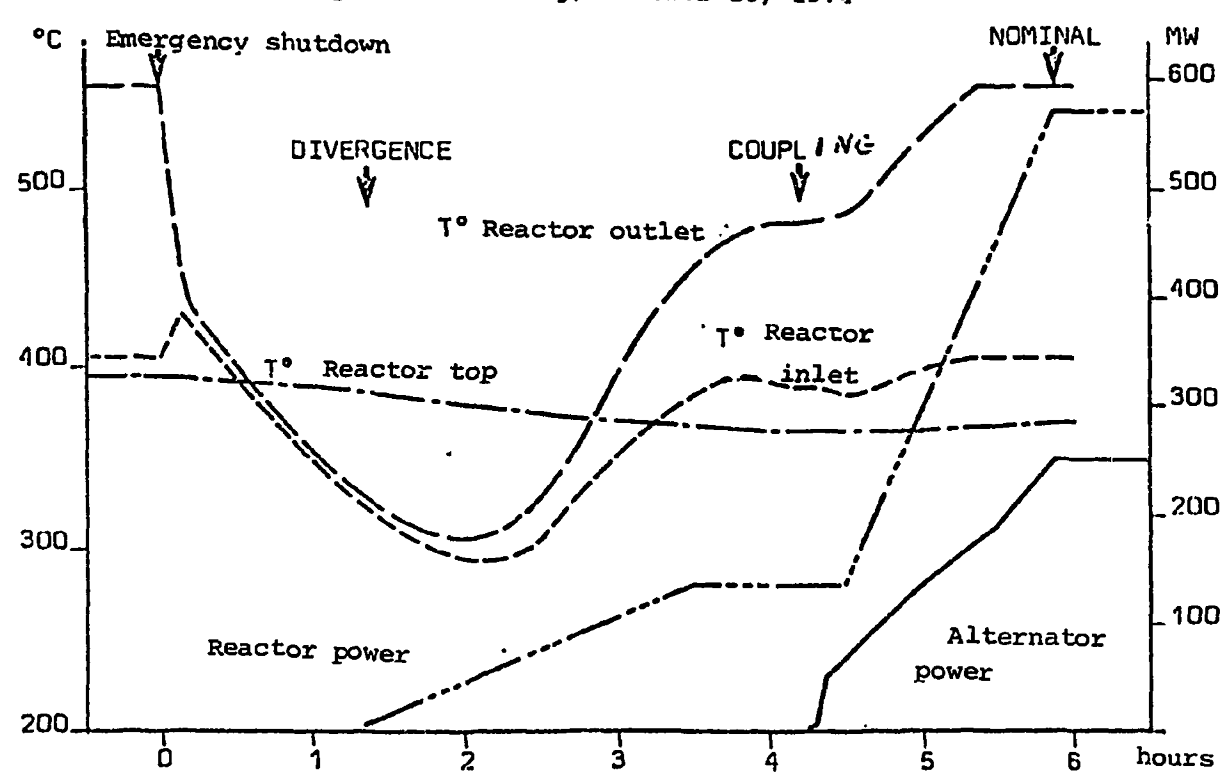

Fig. 3. Startup after unscheduled shutdown, August 21, 1974

70°C. In case of an unscheduled shutdown followed by an immediate re-startup and drying on the water side, this imposes a lowering of the sodium temperature to 240°C, with a significant reactor cooling time, and then a rise in power slowed down by structure constraints. We may say that each unplanned drying out costs the station a day of production.

In December 1974 tests were made with a rise in power allowed by under-use of the core whose temperatures were much lower than the nominally characteristic ones predicted. A thermal power of 591 MW was reached, with a gross electric power of 267 MW by maintaining, in particular, the maximum clad temperature on this side of its project value, 700°C, the primary and secondary sodium flows being unchanged. The only supports found on this occasion came from regulations of the energy-producing installations. It was possible starting with the next stoppage to modify the points of constriction, thus making possible a normal operation at the power reached in the tests each time that the core geometry perittted. Probably the turbo-alternators could reach the 270 MW-e planned by the constructor.

We will not say much about the general control because this part of the installation showed good reliability. Only the sodium pump feed control gave any difficulty, which was indicated in other ways. Emission relaying was satisfactory, and interventions were few. The calculators, after several failures at the start during the tests, operated well. Maintenance was especially concerned with the outlet peripheries which had to be replaced. The core-control calculated was the cause of two unscheduled shutdowns in July 1974.

V. REACTOR BLOCK

1. Fuel - Core Management

At the end of March 1975, the fuel assemblies of the first core reached the performance values

Maximum burnup rate	5.13%
Maximum mass burnup	42,000 MWd/t

No clad ruptures were detected. Numerous fuel handling operations never caused difficulties at introduction or withdrawal. We might reasonably expect that the geometry of these assemblies has remained the same. Two test assemblies were examined after 1 and 3 irradiation cycles. The results were satisfactory.

The control rods,being permanently activated mechanically, should have a geometry that can accept some deformation in the neutron flux and especially a possible expansion of the steel. The actual position of use is caution, and the rods are changed regularly for examination. By means of several progressive arrangements, we hope to obtain reasonably a correct behavior for ten irradiation cycles. A more important modification is to the study to increase this time.

Good management of the fuel in the core has allowed us to reach powers clearly greater than those projected. The calculational code used, and permanently improved, gives the heatings, the linear powers, and the temperatures at the hot point of each assembly with precision. Taking the technical limitations of the fuel and safety into account, the maximum power hoped for for each cycle is determined to about 2%.

The reactivity of each charge is estimated to at least about 100 per thousand. The development since the first cycle has been 21 parts per thousand per JEPP, about 5% greater than projected.

The isotopic analysis of the first irradiated and reprocessed assemblies will permit, in the next few months, improving still further the constants taken into account in the calculational codes and also knowing the possible regeneration rate.

3. Reactor Block Structures

The geometry of the main tank supporting the reactor block necessitates a precise surveillance on the strains, and thus the temperatures, especially during the transition periods of startup or shutdown. The sensitive part is located in the upper part of the walls of the tank bearing the connection with the roof, the slings, and the double enclosure. The sodium level in the tank, which varies with the flow, disturbs the temperature distribution on the tank as a function of its own temperature.

The maximum gradient permitted during a startup, i.e., during the temperature rise, is 600°C per min. The same gradient during decrease in power or shutdown, i.e., during the lowering of the temperature, is 300°C. A maximum temperature difference of 140°C is imposed between the roof and the cold sodium at the reactor inlet. These values depend on the rapidity of reheating of the top, i.e., on the temperature of the core outlet sodium, and the levels in the tank, i.e., on the flow in the primary pumps. Thus the variables must be optimized. Experiment permitted us to determine the best procedures in each case and the subsequent operations of startup or shutdown permitted maintaining the gradient at the tho pf the tank between 400 and 500°C/min during startup and 150-200°C/min in shutdown.

In order to minimize the gradients and therefore the strains, the top of the tank has an outside electric reheater. It is regulated to maintain a temperature difference of 16 to 20°C between the roof and that expected in the tank. Its operation has been without problems so far.

The integrity of the core support structures is controlled by the altitude position of the support beam. Measurements were made at each shutdown with a precision of about 1 mm. No change in these measurements has been found since the loading of the reactor. Measurements made during handling of the fuel by means of handling rods have confirmed the geometric steadiness of the beam.

3. Sodium Pumps

The three sodium pumps of the primary circuit and the three of the secondary circuit are similar. They were subjected to very careful testing before being installed in the reactor. Since their operation in use, we have noted the following incidents with the primary pumps:

A seizing of the mechanical tightness filling; a new design of the parts, without change in the rubbing parts, eliminated the trouble.

Poor functioning of the valve, on the discharge side; the docking of the valve support, which determines the hydraulic forces on the mobile assembly, had to be modified.

An erroneous measurement of the sodium temperature at the pump inlet was taken care of by a more correct positioning of the thermocouple

The modifications on the valves and thermocouples required removal, modification, and reinstallation of the three primary pumps. These operations, which necessitated complete cleaning, dismantling, and reinstallation, were an interesting experience.

Since these modifications, the equipment has operated more than 12,000 hours for the primary pumps and 15,000 hours for the secondary without incident.

While the pumps have shown satisfactory mechanical operation, the electric system (Scherbius system) has given us more trouble. There have been frequent minor incidents, but the number caused more disengagements during the first month of tests. Relay modifications and component replacements led to a reasonable reliability which should be good in the nearfuture. Since startup, there have been four shutdowns attributable to this equipment.

4. Detection and Localization of Clad Rupture (DRG-LRG)

Clad ruptures may be detected by three different methods: total removal of the argon from the top of the pile, removal of sodium at the inlet of the six exchangers, permitting a spatial prelocalization, and removal of sodium at the outlet of each assembly by mechanical selection. The detectors associated with these procedures are of the type appropriate for the fluid removed and may involve safety actions. Very complete tests, chiefly on calibrations with standard gas, were made during the preliminary test period.

Since the start of reactor operation, there has been no significant release of gas in the DRG argon. Continuous external pollution by fissile materials from several U-235 assemblies has permitted repeating the tests and confirming the good operation of the DRG and LRG.

During the first operating cycle, an exchange of two assemblies corresponding to a movement of the pollution from about 1 cm^2 of rupture on the same exchanger permitted detection by the "Prelocalization by 1/6-th of the Exchanger" of the DRG sodium. [sic]

In the second cycle of operation with two steam generators, this same operation showed a hydraulic dissymmetry corresponding to the zone of the insulated exchangers.

At the end of the fourth operating cycle, the removal of two assemblies decreased the counts due to pollution 20%.

5. Sodium Control

The sodium of the primary circuit is purified in an auxiliary circuit external to the reactor block. A flow of 10 m^3/hr comes out of the reactor and regenerates the return of the DRG. It is reinjected after passage through a cold trap. Stoppage indicators permit control of the sodium quality. Removals are made several times per cycle either directly in the tank or on the purification circuit at the end of the analysis. Sodium activation is normal:

Na-24: 6 mCi/g, saturation value at nominal power
Na-22: 0.1 uCi/g, at the end of the fourth cycle

The traces of fission products detected come from pollution of the manufacture of the assemblies and are at the limits of analytical detection. The carbon content is less than 30 ppM, which is the limit of analysis.

VI. SECONDARY CIRCUITS

1. Steam Generator

The steam generator tests made before startup of the station permitted testing natural convection, and the safety features in case of a sodium-water reaction. Operation has been generally satisfactory, the only incidents to be noted being the following:

A small sodium leak on a regulating valve near a weld; after replacement of the valve, the leak reappeared in 4 months. A temporary repair permitted re-startup, but a local repair of the piping was necessary.

A water leak to the outside at a closure plug of the module caused by a defective weld.

Incomplete sealing of the valves regulating the water flow, caused by wear of the inner parts.

The most significant event was operation with two secondary loops, the first time for 20 days in September and October, then for 9 days in March 1975. The operating periods took place under ideal conditions of homogeneity and regularity, the power extracted being exactly 2/3 of the normal power with three loops.

2. Sodium Piping

During the first fillings of the secondary loops and the first temperature rises, abnormal displacements were detected on the piping. The trouble was corrected, and, after replacement of the collars and controls of the supports, the anomalies disappeared. Since being put into service, the piping has functioned well and there have been no new problems.

A sodium leak occurred twice at the same place, near the regulating valve at the generator 2 superheater inlet. A crack of several centimeters was produced near the valve-pipe joining weld. The geometry of the valve sleeve partly explains the trouble. The valve was replaced, but the same leak appeared again several months later. A temporary repair was made. This repeated incident was due to the local geometry of the piping and possibly to the welding method. Studies are in progress.

3. Detection of Hydrogen in the Steam Generators

Possible water leaks in the steam generators were detected by measuring the hydrogen diffusing across nickel tubes 0.3 mm thick. The steam generators did not have a free level, and the detection was made directly in the sodium. A mechanical selection made it possible to localize the module of the steam generator possibly at fault. The temperature of the sodium arriving is kept at a temperature greater than 300°C. Measurements are corrected by the control calculator (T.C.I.)

Tests made during the startup permit us to control and regulate the installation. Calibration by injection of hydrogen was done at different sodium temperatures.

The installation has been operational since the end of 1973. The hydrogen diffusion rate across the water tubes has remained stable at around 500 mg/hr per steam generator during the two first cycles of operation and then decreased regularly. It is of the order of 350 mg/hr. The result is a residual hydrogen concentration in the sodium of about 0.090 ppM when the cold trap operates at 110°C. It is possible to detect a concentration variation of 0.005 ppM. It is estimated that the minimum detectable leak permitting finding the module at fault is 0.03 g/sec.

VII. POWER PRODUCTION PLANT

1. Turbo-alternator Group

This group, like similar installations in the EDF power station, has operated satisfactorily so far. However, an inner spectacular deterioration should be indicated, although it has not had a significant effect. This is a rupture of two large draw-off pipes. The result was a power loss of several megawatts for two weeks and then, after simple stoppage of the draw-off, a return to normal power. These pipes were eventually redone.

The turbo-alternator assembly has profited, like most others, from an authorization to increase the power to 270 MW. The maximum reached as of now was 267 MW and only external regulatory problems have limited this power. We can hope for regular operation greater than 260 MW with the reactor core at equilibrium and loaded entirely with plutonium.

2. Components of Water-Vapor Circuit

The assembly of components of the circuit has been satisfactory except for the valves. This seems to be the usual situation at the startup of a station. But the generally very good quality of the nuclear components shows the deficiencies of more ordinary installations which are very easily served by a more average quality of material. The use of water has passed 20 tons/hr because of the poor life of the valves.

We profited from this state of things to revise some circuits and eliminate purely and simply whole pieces with their valves that cause incidents. The installation was planned with a high possibility of configurations that are useless.

The number of valves that had to be considered is interesting. Two hundred and 260 clamp-packed valves, made 50 repairs of various types, replaced 20 valves completely, and, above all, removed 75 of them. We hope that the situation is improved but we again regred that this part of the station is too often treated without proper care.

3. Production and Treatment of Water

The production of water is ensured by two methods, the normal one giving 30 tons/hr. Use is usually around 10 tons/hr. The minimum has been 6 and the maximum 22 tons/hr. This is to be compared with the flow of water in the steam generators, which is treated completely at each passage and amounts to 750 tons/hr.

The successive treatments are filtration, demineralization on mixed resin beds, and finally treatment to adjust the pH. The treatment values correspond to record book values of the charges,

Fe	< 10	$\mu g/l$
SiO_2	< 10	$\mu g/l$
Conductivity	$< 0{,}1$	$\mu S/cm$
pH	9,2 – 9,4	

The first startups were the source of great difficulty for the water processing. The large iron content caused delays in the rise to power. By a radical change in the type of filter bed the problem was solved. The results immediately became satisfactory. The filtration has been used systematically at each startup because work at shutdown disturbs the circuits.

VIII. FUEL HANDLING

1. Development of Campaigns

With definitive startup of the station, the fuel handling installation had already functioned in an operational way. The first loadings and the numerous neutron experiments at low power necessitated operation of the installation corresponding to several years of normal operation.

Since July 14, 1974, four normal handling campaigns have been made. Campaign No. 2 of July 29 to Aug. 3 1974 used 18 assemblies of the first core. It lasted 5 days, from the shutdown of the reactor to the reaching of divergence. Several incidents in rotation of the turning plug slightly lengthened the predicted delay.

Campaign No. 3 from 14 to 18 October 1974 lasted 4 days in all and used 11 assemblies. Again, some difficulties in rotation of the turning plug delayed the development. Complete examination of the plug and the sealing was done.

Campaign No. 4 from 6 to 12 January 1975 used 42 assemblies. Set up visus No. 2. The turning plug operated well. (NOTE:"visus" defined below).

Campaign No. 5 from 23 to 28 February 1975 used 17 assemblies. Turning plug operated well, as did visus.

Movements of the assemblies indicated were in internal transfer and removal or new chargings. All told during this period, there were 88 removal operations and the same number of introductions; 35 assemblies were removed and the same number charged. There were no notable incidents.

Three irradiated assemblies were dismantled in the cell after one and three cycles of irradiation. They were subjected to a large number of examinations which were not all completed. One of these assemblies was sent to the plant for reprocessing and underwent the complete operation for test.

The cell for dismantling was used for numerous tests on false assemblies, the installation being operational and in particular in a nitrogen atmosphere. The information obtained in these tests made it possible to make several improvements in the cell machines. It has been active since February 1975.

2. Principal Components

Turning Plug. This component, the most important, caused several disturbances in the handling operations during the first campaigns. The efforts for rotation became harder and harder, and a pure and simple stoppage was feared. It was finally found that incidents arose from the kinematic entrainment chain and the motorization and not from the rotation efforts which were within normal limits. The motorization underwent several modifications and the plug rotation offered no further difficulties.

The tigntness was completely controlled by liquid metal on the lifting jacks and the internal surfaces. All was in good order.

Transfer Arm. The only incidents were those caused by rotation difficulties. Considerable effort resulting from the strains undergone by the turning due to differential dilatations whose importance had been underevaluated. The other components did not cause any notable incident.

3. Visus

Each movement of the transfer arm in the reactor is preceded by a control charge with a detector for a possible obstacle. This system of Visualization in the Sodium by Ultra-sound (VISUS) permits, by sweeping the space above the core, guaranteeing freedom of the maneuver.

During the first handling campaigns for the fuel for the first reactor loading, this system showed "phantoms" of obstacles. Partial shielding of the reactor sodium at the assembly heads was considered necessary. This was an operation of great practical interest and the only one of this type. The "phantoms" suppressed by electronics, the visus remained operational and did the service for which it was intended. The only problem was that it was necessary to remove it before each startup and put it in the pile at each shutdown because it did not bear the hydraulic force of the sodium current at full flow without vibrating. Visus No. 2 was constructed and replace in January. The new one is retractable and can thus remain in place permanently, permitting a gain in availability of 2 days per campaign. Its operation is also satisfactory, like that of its predecessor.

IX. CONCLUSION

It is interesting to confirm first the validity of the large operations for Phenix and the operation. We first consider the solution of the integrated reactor for which some disadvantages were expected. Their final effect on the operation was shown to be insignificant. For example, the strains in use caused by the structures had to be considered. We were able to determine that experience has allowed us, if not to remove these strains, at least to take care of them by a relative increase in the startup delay.

The advantages of the solution, on the other hand, have retained all their value. It is sufficient to review the principal ones. The inertia due to the large mass of sodium facilitates all use operations and, further causes a significant intrinsic safety factor. Natural convection in case of a stopping of the flow of coolant in the primary circuit has been demonstrated. The only sodium leaving the reacor is the secondary sodium whose activity is negligible. The results of the surveillance of personnel and of the environment show the absence of any problem. The advantages mentioned are chiefly those of use. It is others, especially safety, which confirm clearly the validity of the operation.

The use of sodium as cooling fluid is shown to be convenient for the procedure when the necessary precautions are taken. The absence of pressure in the circuits then becomes significant. Leaks are easy to detect.

The system of three secondary loops showed an astonishing flexibility from the point of view of control of cooling the reactor at shutdown as well as the possibility of operating with a loop out of service. It is despite all at this point that we should note one of the most unfortunate restrictions for use, determined by the limitation of the temperatures during filling of a loop.

The fuel which approaches its normal lifetime without rupture of the clad seems able to satisfy the project requirements.

The yield which surpasses 45% is at the top of values usually found in standard thermal power plants of the most modern design.

The principal components generally give satisfaction.

The feasibility and general availability are especially satisfactory and have been mentioned in this paper, e.g. specific advantages just enumerated.

From the economic standpoint, the results of use are also satisfactory because of the exceptional charge factor and in always superior to those predicted for the first year of operation. The economics will be studied after several years of use when the complete fuel cycle will have been run through. This is still unknown and its effect on the economic calculations will be basic.

It thus appears that the choices made fot the Phenix station have been good. The large options confirm their value and permit reasonable assurance of a similar success for the eventual power reactors of the French program and the use of fast neutron reactors.