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FUEL MANAGEMENT CODES FOR FAST REACTORS

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*** EDF

The CAPHE code is used for managing and following up fuel subassemblies in the Phenix fast neutron reactor; the principal experimental results obtained since this reactor was commissioned are analysed with this code. They are mainly concerned with following up fuel subassembly powers and core reactivity variations observed up to the beginning of the fifth Phenix working cycle (3/75). Characteristics of Phenix irradiated fuel subassemblies calculated by the CAPHE code are detailed as at April 1, 1975 (burn-up steel damage).

I. INTRODUCTION

Since the Phenix reactor was commissioned on July 14, 1974, the calculation code used for the fuel management of this reactor has made it possible to correctly choose the successive core load plans; in addition, the neutronic, thermal and hydraulic characteristic of the fuel subassemblies have been foreseen by this code with sufficient accuracy for safe operation.

In future, this code, called CAPHE, will enable Phenix reactor fuel subassembly management to be optimized under good conditions; this will be possible when the technological criteria defining maximum stay of subassemblies in the reactor have been improved; the technological criteria currently adopted for the fuel are those of the "project" studies for the reactor, viz :

1) nominal linear rating, below 430 W/cm
2) nominal clad temperature, below 650°C
3) maximum clad temperature, below 700°C
4) nominal burn-up, below 50,000 megawatt-days per ton; in the Phenix core center this corresponds to a displacement number below 72 per atom of steel.

After a brief description of the principles called upon in working out the CAPHE code, we indicate the principal calculated results obtained, since reactor start-up, regarding following up the reactivity and subassembly power distribution; these results are compared with the experimental values measured on the reactor. Lastly, we indicate the characteristics, calculated by the CAPHE code, of Phenix irradiated subassemblies as at April 1, 1975 (burn-up, steel damage).

II. DESCRIPTION OF THE CAPHE CODE

II.1. Aims

A fuel management code must, in the first place, make it possible to know, at any time when the reactor is working, all the irradiation conditions (neutronic, thermal and hydraulic) of all the subassemblies in the core; secondly, it must make it possible to choose the subsequent core load for the following working cycle of the reactor or to modify the reactor load very quickly according to operating constraints.

The CAPHE code carries out all the neutronic, thermal and hydraulic calculations for a core load (at the beginning and at the end of the working cycle) and makes it possible to foresee the characteristics of a subsequent load plan (at the beginning and at the end of the working cycle). For a given core load, this code furnishes the following calculated characteristics for each subassembly :

1. total power
2. sodium flow
3. sodium temperature increase
4. maximum nominal clad temperature calculated with a sodium mixture between the channels
5. maximum hot point temperature of a clad
6. maximum linear power reached in the fuel
7. maximum fission burn-up attained by the fuel
8. average fission burn-up attained by the fuel
9. total fluence attained (maximum and average on the fuel)
10. number of displacements attained per atom of steel (maximum and average on the clads)
11. the balance in heavy atoms for the fuel and the lower and upper axial blankets.

Lastly, the reactivity balance of the subsequent load in relation to the initial load is calculated.

II. -2. Principles adopted

The two basic principles that served for the conception of the CAPHE code are the following :

1) the calculation - experiment divergencies observed during neutronic tests at reactor start-up must be able to be simply integrated in this operating code;

2) the code must be inexpensive to use (very short calculation time).

From the neutronic point of view, the two possible ways of preparing this kind of code are as follows :

1) utilize the standard multi-dimension neutron codes, with diffusion approximation; the problem then, for a desired precision, is to reduce calculation costs, by working with few energy groups and with a wide spatial mesh; this way only complies very imperfectly with the two principles set out above :

- the calculation - experiment divergencies observed at start-up are difficult to insert in this kind of code

- the programming is more complicated

- costly to operate.

2) Knowing the calculated neutron flux distribution in the start-up core, determine the neutron flux distribution of a given load from the calculated effects established separately for each envisageable perturbation of the start-up core load. The distribution of the initial neutron flux (and powers) and the perturbation sub-programs are established once and for all using all the standard neutron codes (multidimensional, hexagonal, mesh 3.5 cm, 6 energy groups).

The second method was chosen for the CAPHE code, bearing in mind all the neutron calculation tests carried out on the interactions of the different reactivity perturbations (and therefore the neutron flux perturbations). There is obviously a large number of such perturbations; they concern :

1. position of the control rod curtain
2. individual interaction of each control rod with the others
3. placing diluents in any position in the core
4. placing experimental subassemblies in any position in the core
5. any modification of the interface between the two core zones
6. any modification of the outer contour of the core
7. replacement of uranium 235 subassemblies (transient outer core) by standard outer core subassemblies (plutonium subassemblies)
8. fuel wear in the different subassemblies (core 1 plutonium, core 2 plutonium, core 3 uranium, experimental subassemblies)
9. plutonium formation in the axial blankets of each subassembly
10. plutonium formation in the radial blanket subassemblies.

This way made it possible to integrate in the start-up core calculations all the calculation - experiment divergencies observed during neutron tests on start-up (power distribution, influence of control rods, reactivity value of the subassemblies). Furthermore, it enabled us to work out an inexpensive, simple to use code for the reactor operator.

The only drawback to this kind of code is that all load perturbations have to be thought of and foreseen right from the time the code begins to be employed; this is the contrary of the other possible way (conventional neutron codes).

II. -3. Simplified flow chart of the CAPHE code

The code has the following data available :

1) detailed neutronic results on the start-up core and results of all envisageable perturbations

2) fuel fabrication data, geometrical and hydraulic data for each subassembly

3) successive load plans

4) operating conditions of the reactor and duration of each of the two working cycles.

All the neutronic, thermal and hydraulic calculations fit in together as shown in Table 1.

These calculations can be effected four times in the CAPHE code :

1) for the load called "in hand" : at the beginning and at the end of the cycle called "in hand"

2) for the subsequent load called "estimated load" : at the beginning and end of the "estimated" cycle.

Table 2 shows the operating diagram utilized conventionally and the points of calculation effected.

III. RESULTS OBTAINED ON PHENIX

The results of calculation - experiment comparisons obtained on Phenix can be separated into two parts :

1. those concerning following the power of each subassembly
2. those concerning following the reactivity.

We shall assume that all the data concerning the working of Phenix since its commissioning are known (1).

III. -1. Results concerning following the power of subassemblies

III. - 1.1. Core load estimates

Before the end of the "in hand" cycle, the CAPHE code furnishes the calculated characteristics of each subassembly of the "estimated" cycle.

Before the beginning of the "estimated" cycle, the characteristics calculated in this way are used to control the numerical values necessary for the two computers that survey the sodium temperature rise ΔT^i in each subassembly i from the safety point of view ("fouling" surveillance).

During the first power increase effected at the beginning of the "estimated" cycle, both computers constantly make calculation - experiment comparisons of the values of these temperature rises ΔT^i ; both computers set off an alarm (then a reactor scram) if, for one subassembly, the calculation - experiment divergence obtained on both computers exceeds the threshold fixed for the alarm (then for the reactor scram).

Thresholds are as follows :

1) for the alarm : about 3% of ΔT^i
2) for the scram : about 7% of ΔT^i .

The ΔT^i values calculated by the CAPHE code therefore condition the proper working of the reactor power increase.

To take into account the calculation - experiment divergencies already observed during the "in hand" cycle, the ΔT^i calculated values are corrected by these divergencies for those subassemblies which have not changed position during the handling period separating the "in hand" cycle from the "estimated" cycle.

We assume that the calculation - experiment divergencies are conserved during the cycles, i. e. that they arise from causes inherent in the subassembly, due to fabrication tolerances (hydraulic, plutonium content, etc). On the other hand, for new subassemblies entering the core for the first time, no correction is made to the ΔT^i calculated values supplied by the CAPHE code and used to control the numerical values necessary for both computers.

The histograms giving the number of subassemblies in relation to the calculation - experiment divergence on the ΔT^i observed on starting up the 2nd, 3rd, 4th and 5th cycles are indicated in tables 3, 4, 5 and 6; as seen previously, we have separated the new subassemblies from used subassemblies that have not changed position.

We can see that the number of subassemblies whose calculation - experiment divergence on ΔT^i is above 3% is very small, which enabled the first power increases of each of the Phenix working cycles to be made in perfect safety.

This is true both for subassemblies which have not changed position and for new subassemblies. However, the calculation - experiment divergencies of the latter come within a wider range, which reveals the uncertainties inherent in subassembly fabrication tolerances.

III. - 1.2. Subassembly power evolution

Neutron flux being constant, the power of Phenix subassemblies diminishes by about 1.3% per irradiation cycle (56 days).

To follow ΔT^i calculation - experiment divergencies concerning this power variation, we concern ourselves solely with subassemblies that have not changed position in the reactor from the first cycle to the n^{th} working cycle. In addition, we correct each of the gross calculation - experiment ΔT^i divergencies by the calculation - experiment divergencies observed at the beginning of the first cycle, so as to eliminate uncertainties inherent in fabrication tolerances; let ε^i be the calculation - experiment divergence (in %) thus obtained for subassembly i at the beginning of a cycle.

Tables 7, 8, 9 and 10 show the histograms giving the number of subassemblies i in relation to the calculation - experiment divergence ε^i observed at the beginning of the 2nd, 3rd, 4th and 5th Phenix working cycle respectively.

Since we are only concerned here with subassemblies that have not changed position from the first cycle to the n^{th} cycle, the total number N of subassemblies concerned diminishes from Table No. 7 to Table No. 10, i.e. from the beginning of the 2nd cycle to the beginning of the 5th cycle :

N = 103 at the beginning of the 2nd cycle
N = 102 at the beginning of the 3rd cycle
N = 83 at the beginning of the 4th cycle
N = 67 at the beginning of the 5th cycle.

We can see that the successive histograms deteriorate slightly between the beginning of the 2nd cycle and the beginning of the 5th cycle; however, the maximum value of the 5th cycle; however, the maximum value of the calculation - experiment divergence ε^{i} never exceeds 4%.

There are two kinds of cause for the slight deterioration of these histograms :

1) a slight deterioration of the working conditions of the subassemblies as they are irradiated in the core. This seems to be excluded for the moment, bearing in mind the radiation rates attained so far (cf. section IV);

2) a slight deterioration of the calculation results furnished by the CAPHE code as changes are made to the core load in relation to the start-up core load :

- withdrawal of new diluents
- reduction of core size
- insertion of special irradiation subassemblies in the reactor
- fuel evolution in the subassemblies
- plutonium formation in the blanket subassemblies
- etc.

In any case, it seems that in future we shall be able to predict subassembly power evolution by calculation with sufficient accuracy to be able to detect any serious subassembly working anomaly (progressive loss of flow due to steel swelling for very high radiation rates).

III. - 2. Results concerning following the reactivity

These concern :

1) the anti-reactivity value of the control rods at each working cycle;

2) the "cold" critical dimension figure of the control rod curtain during the divergence effected at the beginning of each working cycle;

3) the daily fall in reactivity due to fuel wear and the formation of fission products.

III. 2. 1. Control rod anti-reactivity value

The calculated anti-reactivity values of each of the six control rods of the reactor are indicated in Table 11 for each of the five loads corresponding to the first five Phenix working cycles.

The neutronic calculations are effected in 2-dimensionally (hexagonal), in diffusion approximation, with 6 energy groups and utilizing Cadarache CARNAVAL III formula.

The anti-reactivities thus calculated are corrected by two effects :

1) The B_4C heterogeneity effect not taken into account by the hexagonal calculations (-8. 5%)

2) correction due to analysis of the program of neutronic experiments carried out on Masurca and Sneak (-6%).

The individual value of the rods varies from 1304 pcm (BC 5 at the 3rd cycle) to 1400 pcm (BC 2 at the 4th cycle), i. e. a 7. 4% variation, whereas the total anti-reactivity of the six control rods varies in lesser proportions : 4. 5% between the 3rd and 1st cycles.

These variations are due to two main reasons.

1) Local neutron flux perturbations. In the first cycle, for instance, the presence of the three internal diluents against the three control rods BC 1, BC 3 and BC 6 depresses the neutron flux and therefore brings about a fall in the anti-reactivity of those rods (1365 pcm) in relation to that of the other three (1390 pcm).

2) The successive withdrawal of 3 diluents at the end of the first three cycles reduces the neutronic importance of the six control rods and so causes their total anti-reactivity to fall from 8875 pcm at the first

cycle to 8485 pcm at the third cycle.
At the beginning of the fourth cycle, reduction of the external size of the core by 3 subassemblies brings that value up again to 8859 pcm for the same reason.

Anti-reactivity measurements of the 6 Phenix control rods were effected at the beginning of the first, fourth and fifth cycle; the calculation - experiment comparisons concerning the anti-reactivity of the rods are shown in Table 12.

Experimental uncertainties can be divided into two :

- uncertainties due to the experimental procedure : $\pm$ 3%.

- uncertainties attaching to the calculated value of the dollar, estimated at $\pm$ 5%.

Because of replacing uranium subassemblies by plutonium subassemblies in the fourth and fifth cycles, the dollar values utilized were as follows :

β = 442 p.c.m. for the first cycle

β = 435 p.c.m. for the fourth cycle

β = 424 p.c.m. for the fifth cycle.

III.2.2. Divergence dimensions

It will be recalled that the first Phenix reactor divergence was effected on August 31, 1973 (2) and that the experimental critical mass (87.5 subassemblies) was situated right within the calculation forecasts (81 $\pm$ 7 subassemblies).

Since then, a calculation - experiment comparison has been made systematically at the beginning of each working cycle.
Calculated forecasts are based on :

1) the experimental reactivity available at the end of the preceding cycle ;

2) the experimental counter-reaction effects (temperature and power) observed at the beginning

of the preceding cycle.

3) the calculated reactivity variations due to load changes made during reactor shut-down;

4) the calculated control rod anti-reactivity variations from one load to the next.

After the divergence, the estimated calculations are corrected to bring them in line with the experimental conditions; these corrections concern :

1) the sodium temperature (T_{Na});

2) the mean temperature (T_M) along the part of the control rod mechanisms situated "out of the sodium" (expansion effects);

3) the actual duration of the preceding cycle (fuel wear correction (N JEPP);

4) the neptunium effect which depends on the number of days the reactor is shut down (NA).

The experimental conditions observed during the divergencies effected at the beginning of the second, third, fourth and fifth Phenix working cycles are shown in Table 13.

The calculation - experiment comparisons on the reactivity available during each of these divergencies are given in Table 14. So far, the calculated estimates are always within about 100 p.c.m. of the experimental results.

III.2.3. Loss of reactivity due to fuel wear

The experimental values necessary to determine reactivity experimental variations due to fuel wear are read every other hour during normal reactor working; these values are as follows.

1) Total thermal power of the reactor (P).
2) Sodium temperature on entering the core (T_E).
3) Dimension of the control rod curtain (Z).
4) Average sodium temperature rise in the subassemblies (Δ Tc)

5) Average temperature of the part of the control rod mechanisms situated "out of the sodium" (T_M).

The experimental reactivity available in the control rods is then deduced from these measurements :

1) by bringing them to what are known as "normal" working conditions; these conditions are :

$P = 563$ MWth	$T_E = 389°C$
$\Delta T_c = 180°C$	$T_M = 145°C$

2) by utilizing the measured values giving the effectiveness of the control rod curtain in relation to the dimension figure of its insertion in the core (the anti-self-shielding factor utilized is the calculated factor).

Experimental uncertainties concerning these experimental results depend very much on :

1) knowledge of all the counter-reaction effects (input temperature, power, temperature rise in the fissile subassemblies, etc.);

2) knowledge of the effectiveness of the control rod curtain in relation to its insertion in the core.

The experimental results thus obtained are shown in figures 1, 2 and 3 giving, for the 2nd, 3rd and 4th working cycles respectively, the variations of the reactivity available in the control rods in terms of the equivalent number of days the reactor has been working at full power (1 JEPP = 563 MWth in one day).

We can see that :

1) the loss of reactivity due to fuel wear is constant right through a working cycle provided that the neptunium effect which appears at the beginning of the cycle and at each reactor shut-down is eliminated; this experimental reactivity loss is :

21.2 pcm per day during the 2nd cycle
20.9 pcm per day during the 3rd cycle
20.8 pcm per day during the 4th cycle.

The experimental uncertainties on these values are around 10%.

2) the maximum neptunium effect is around (130 $\pm$ 20) pcm.

The calculated values of reactivity losses due to fuel wear and neptunium effect are currently determines by concentional neutronic codes and not by the CAPHE code. The results obtained are as follows :

19.2 pcm per day during the 2nd cycle
18.4 pcm per day during the 3rd cycle
18.1 pcm per day during the 4th cycle.

The calculated neptunium effect is 115 pcm.

It seems, therefore, in the present state of interpretation, that the calculation under-estimates the loss of reactivity due to fuel wear by about 10 to 15% (+ 10%) ~~the loss of reactivity due to fuel wear by about 10 to 15% ($\pm$ 10%)~~ and that the neptunium effect is well calculated.

IV. STATE OF IRRADIATION OF THE SUBASSEMBLIES IN THE PHENIX CORE AT APRIL 1, 1975

Tables 15 and 16 show, as at April 1, 1975, the CAPHE code calculated values of the state of irradiation of the subassemblies in the Phenix core during the fifth working cycle (values utilized in (3).

These values, presented in histogram form, concern :

1) Table No. 15 : the maximum fission burn-up attained by the fuel of each subassembly.

2) Table No. 16 : the maximum number of displacement per steel atom attained by the clads of each subassembly.

These histograms shows the number of subassemblies having attained a given fission burn-up and number of displacements per atom as at April 1, 1975; they separate the subassemblies according to the nature of the fuel.

1) Plutonium core 1 internal
2) Plutonium core 2 external
3) Uranium core 3 external.

The maximum fission burn-up and displacement per atom values are given in the following table, for the three Phenix cores :

Maximum values attained April 1, 1975

	Core 1 Pu	Core 2 Pu	Core 3 U^5
Max. fission burn-up (%)	4.6	4.7	5.1
Max. displacement per atom	57	46	45

For core 1 Pu this corresponds to a burn-up of about 40,000 MWD/T whereas the burn-up aimed at for Phenix is 50,000 MWD/T; the particularly satisfactory working of Phenix during the first 5 cycles augurs well for reaching the burnup aimed at. In particular, no slug burst has been revealed to date.

During Phenix start-up and then working tests up to the fifth working cycle, the calculated results furnished by the CAPHE fuel management code were tested against the principal experimental reactor results (power, reactivity); the calculation - experiment comparisons effected show good agreement and this code can be considered qualified for Phenix fuel management.

The only important point remaining to be tested in 1975 and 1976 is the fuel and fertile evolution problem (composition of each subassembly in transuranians and fission product).

We do not yet have any experimental results concerning this point but a big experimental program in hand will enable us to test the fuel evolution data and sub-program shortly to be incorporated in the CAPHE code.

The first neutron studies concerning the future fuel management code for the Super Phenix reactor will be begun in the forthcoming months. They will make it possible to choose the type of code to be utilized for a big 1200 MWe reactor (either conventional neutron codes or a CAPHE type code).

- o -

REFERENCES

(1) - F. CONTE

Phénix après un an de fonctionnement

Cette conférence - Session B 1 - Communication n° 10

(2) - MM. CLAUZON, COULON, GOURDON, GOURIOU, HUMBERT, MARTIN, MOUGNIOT, SICARD

Proceedings of the International Conference organized by the British Nuclear Society on 11-14 March, 1974 at the institution of Civil Engineers

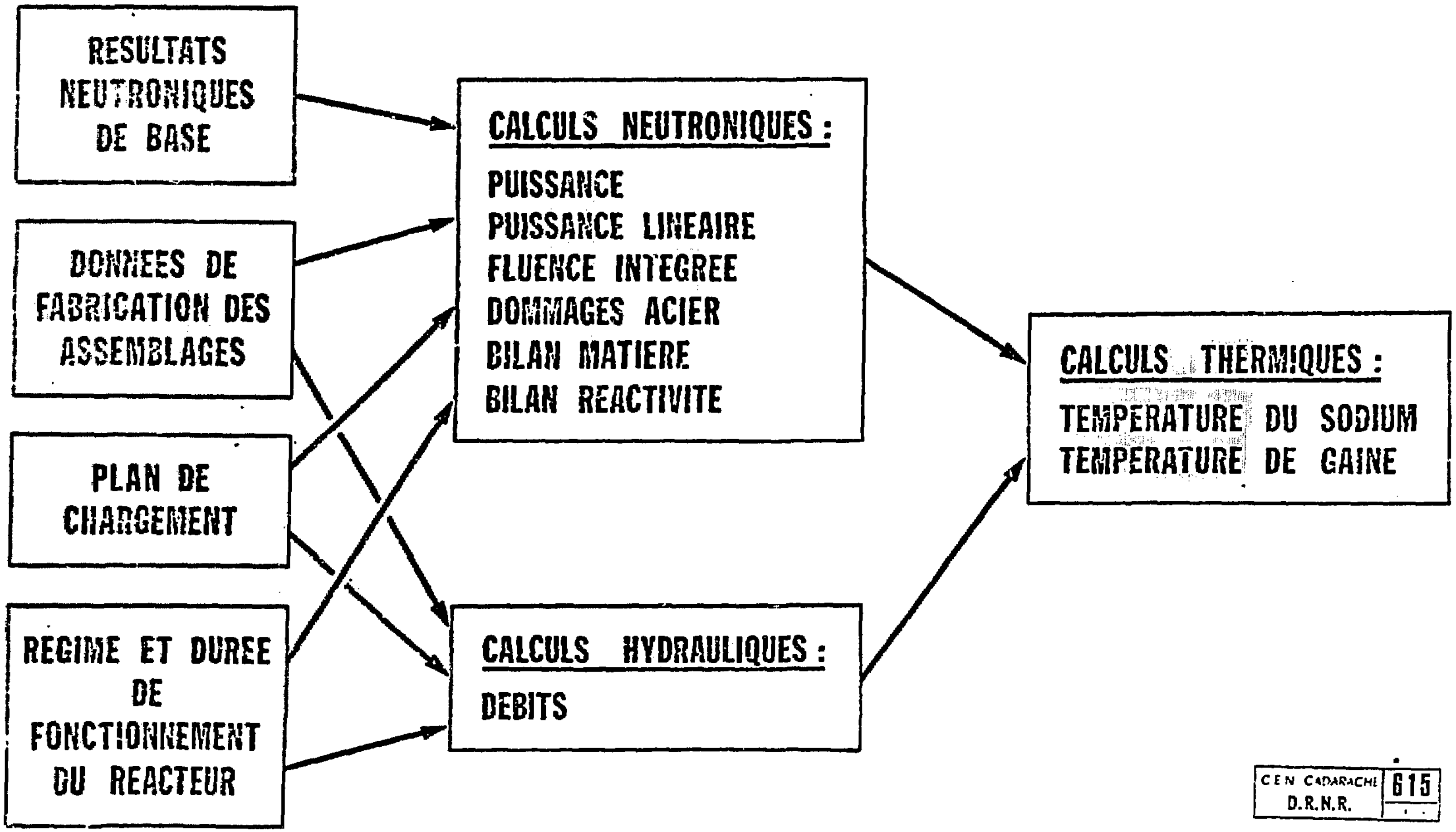
LONDON - Specialist Session 2 - Nuclear Performance

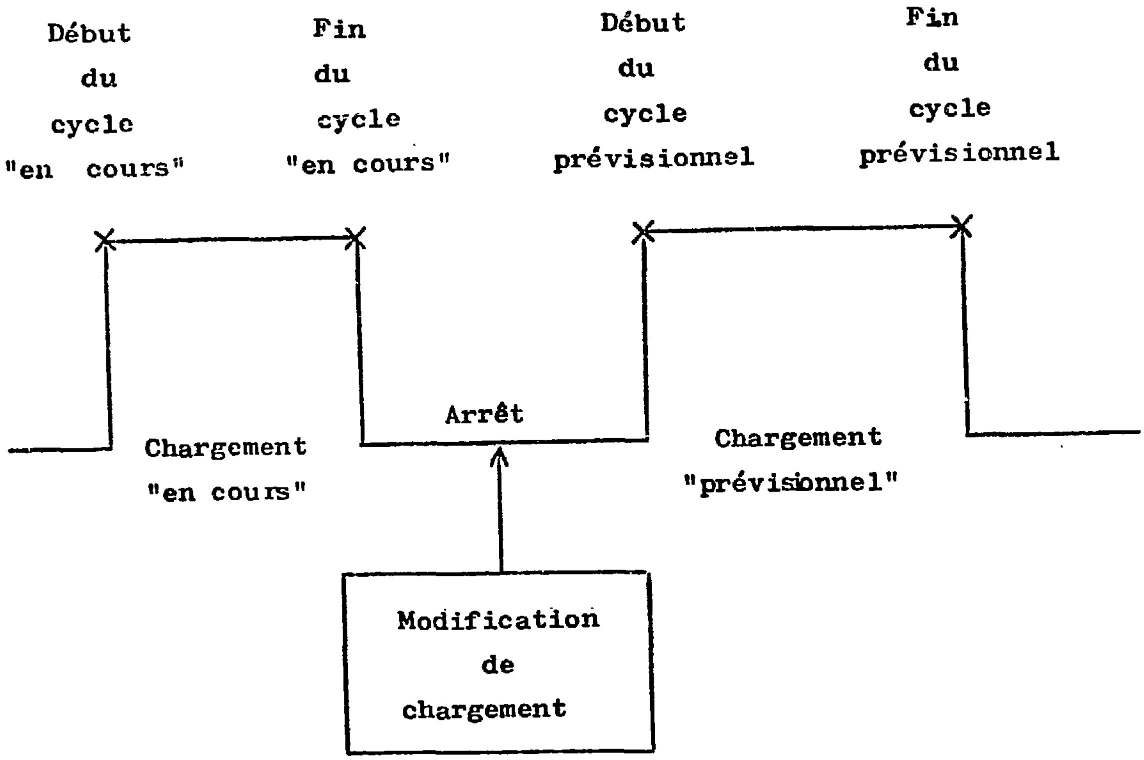
(3) - MM. J. LECLERE - J.P. MARCON

Choix des caractéristiques de l'élément combustible de Phénix. Premiers résultats de leur comportement en pile.

Cette conférence - Session C 5 - Communication n° 4

PHENIX _ TABLEAU : 1



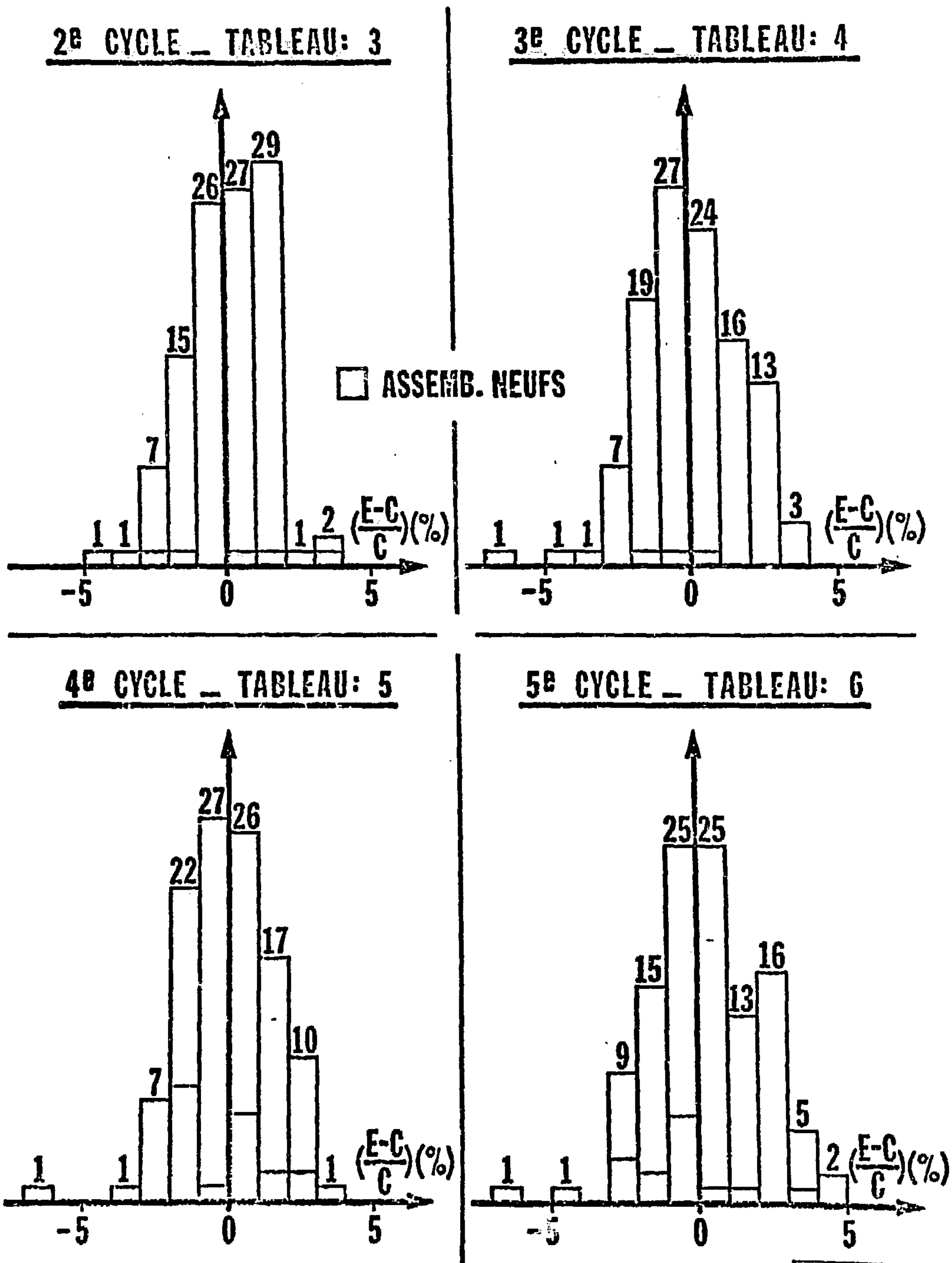


X Calculs effectués par le code CAPHE

TABLEAU N° 2

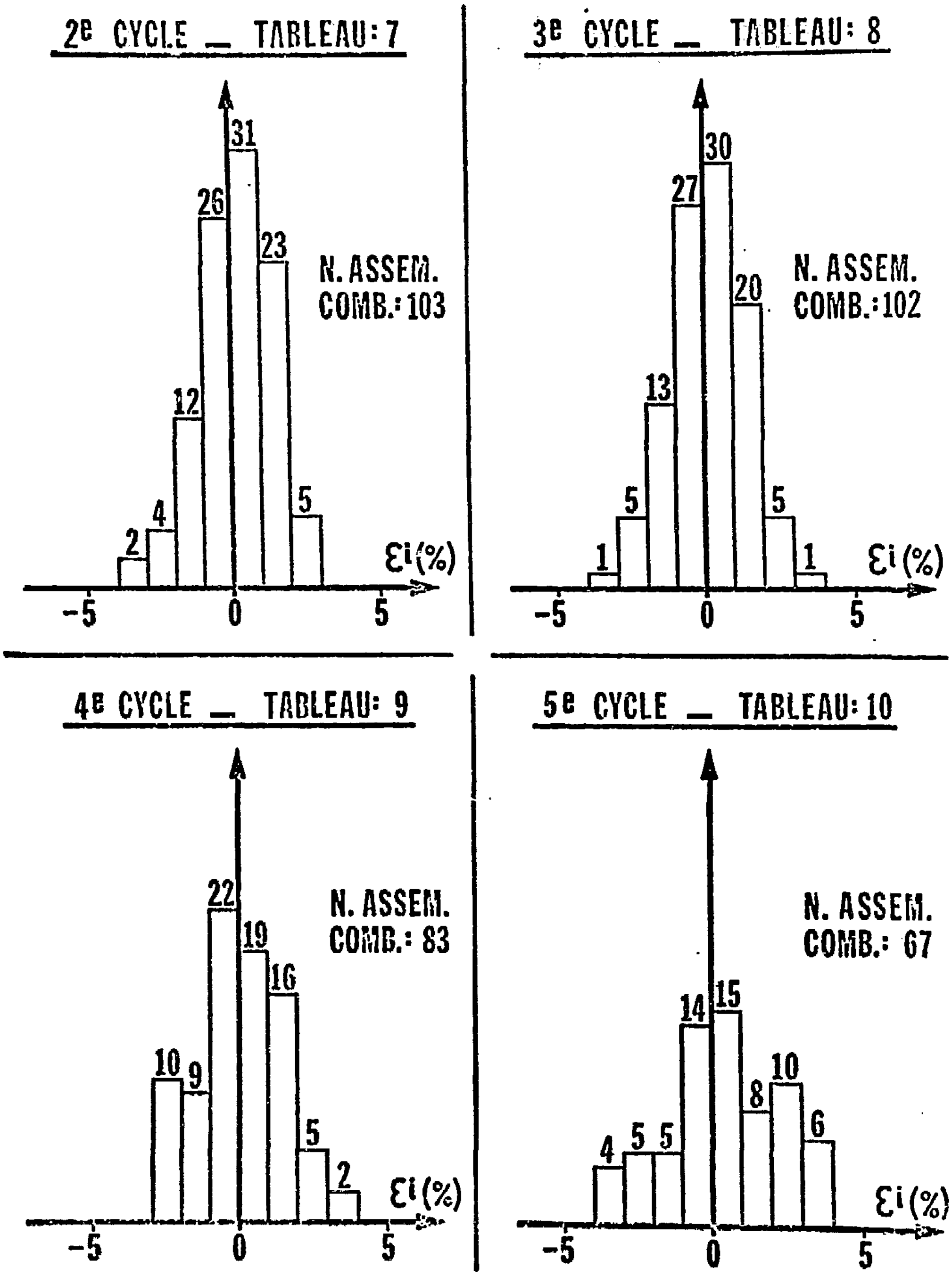
PHENIX

HISTOGRAMME DU NOMBRE D'ASSEM. COMBUST. EN FONCTION DE L'ECART CALCUL-EXPERIENCE SUR ΔT^i AU DEBUT DU:



PHENIX

HISTOGRAMME DU NOMBRE D'ASSEM. COMBUST. N'AYANT PAS CHANGE DE POSITION EN FONCTION DE ε^i AU DEBUT DU:



ANTIREACTIVITE TOTALE CALCULEE DES
BARRES DE COMMANDE AU COURS DES CINQ
PREMIERS CYCLES DE FONCTIONNEMENT DE PHENIX

N° de la barre Emplacement	BC 1 19/18	BC 2 22/17	BC 3 23/19	BC 4 21/22	BC 5 17/21	BC 6 18/23
Premier cycle	1365	1390	1365	1390	1390	1365
Deuxième cycle	1354	1351	1354	1351	1351	1354
Troisième cycle	1319	1362	1338	1343	1304	1286
Quatrième cycle	1379	1400	1385	1372	1314	1317
Cinquième cycle	1374	1391	1351	1331	1327	1308

TABLEAU N° 11

ANTIREACTIVITE TOTALE DES BARRES DE COMMANDE

COMPARAISONS "CALCUL-EXPERIENCE"

$$\left(\frac{\text{Exp-Cal}}{\text{Cal}}\right)(\%)$$

Nº de la Barre Emplacement	BC 1 19/18	BC 2 22/17	BC 3 23/19	BC 4 21/22	BC 5 17/21	BC 6 18/23
Premier cycle	- 0,7	+ 0,6	- 1,1	+ 0,1	+ 0,2	- 0,1
Quatrième cycle	- 0,1	- 1,4	- 1,6	- 0,1	+ 2,2	+ 1,9
Cinquième cycle	- 0,8	- 2,7	- 1,0	+ 3,5	+ 4,2	+ 6,8

TABLEAU Nº 12

CONDITIONS EXPERIMENTALES DES DIVERGENCES

Conditions expérimentales	Début du 2ème cycle 3/8/74	Début du 3ème cycle 18/10/74	Début du 4ème cycle 13/1/75	Début du 5ème cycle 28/2/75
Cote moyenne des 6 barres (mm)	499,8	503,5	498,5	464,1
Température du Sodium (TNa)(°C)	246	231	245	209
Température moyenne des mécanismes de barres de contrôle (TM)(°C)	197	186	169	183
Nombre de jours d'arrêt (NA) jusqu'à la divergence	4,9	5,0	7,8	4,8

TABLEAU N° 13

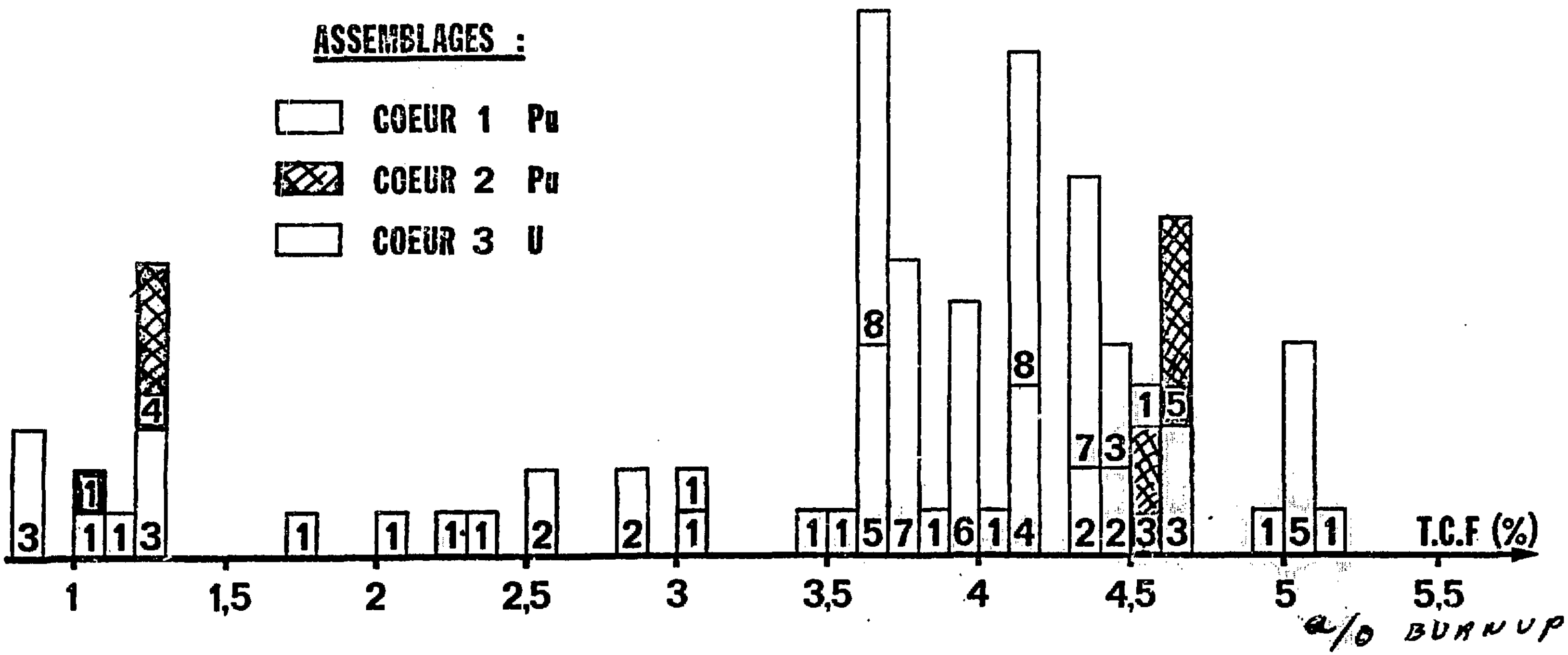
ECARTS CALCUL-EXPERIENCE (en p.c.m.)
OBSERVES LORS DES DIVERGENCES

Divergences	Ecarts (en p.c.m.)
Début du 2ème cycle	$- 70 \pm 140$
Début du 3ème cycle	$- 33 \pm 140$
Début du 4ème cycle	$+ 80 \pm 190$
Début du 5ème cycle	$- 22 \pm 140$

TABLEAU N° 14

PHENIX – HISTOGRAMME TAUX DE COMBUSTION EN FISSION (T.C.F.) AU 1.4.75

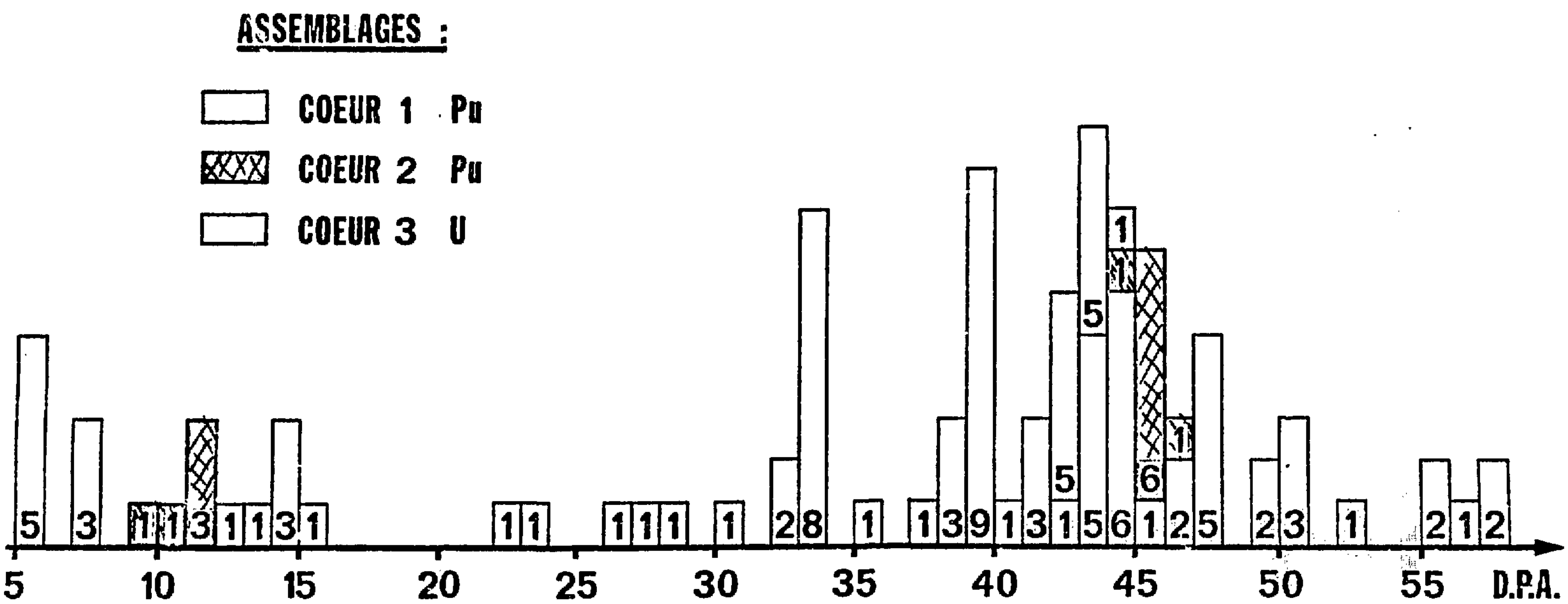
TABLEAU : 15



CEN CADARACHE D.R.N.R. 619

PHENIX – HISTOGRAMME DU NOMBRE DE DEPLACEMENTS PAR ATOME D'ACIER (D.P.A.) AU 1-4-75

TABLEAU 16



CEN CADARACHE D.R.N.R. 618

FIG. 1 PHENIX

PERTE EXPERIMENTALE
DE REACTIVITE DUE A L'USURE
DU COMBUSTIBLE AU COURS DU
2e CYCLE DE FONCTIONNEMENT

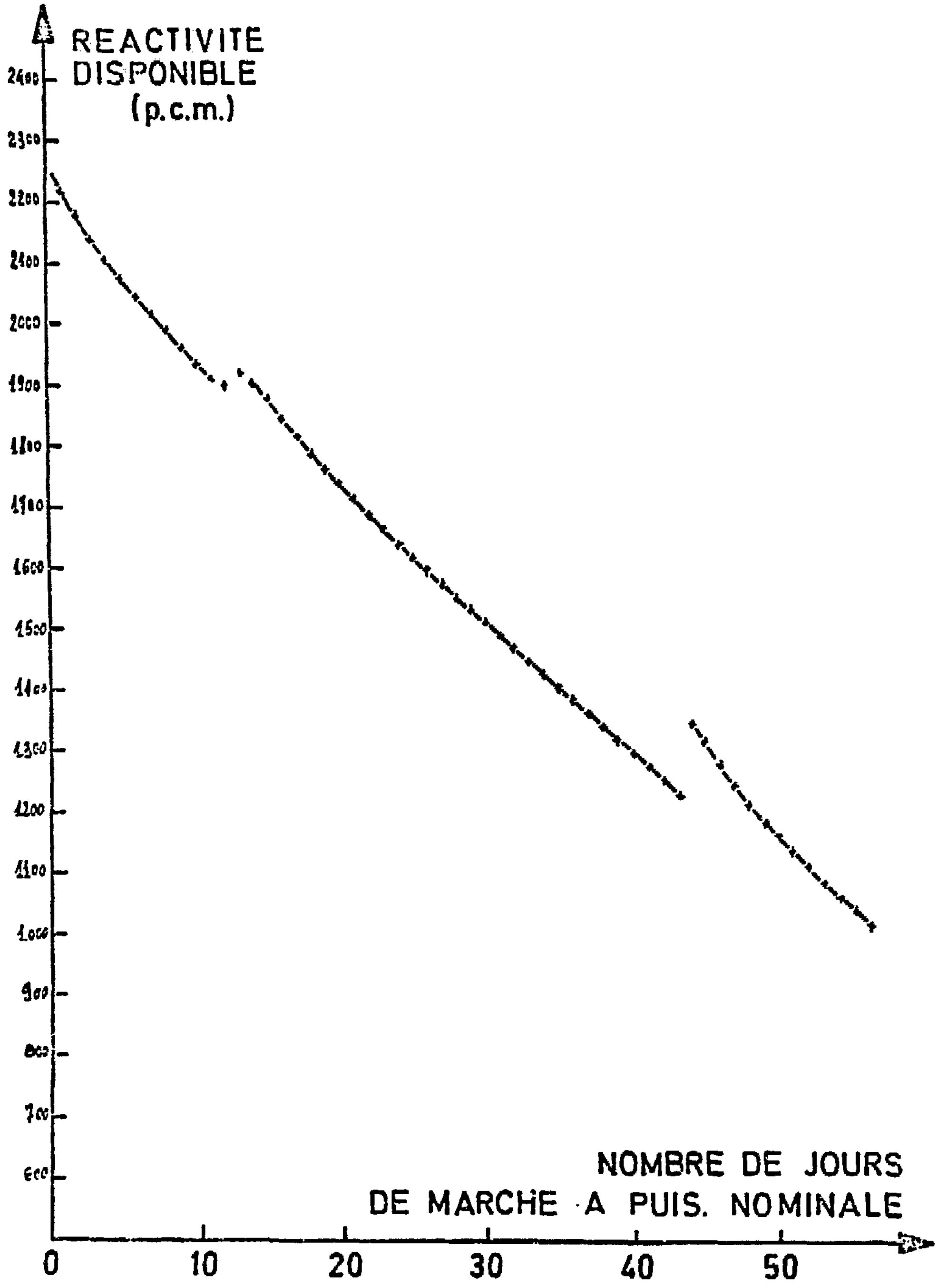
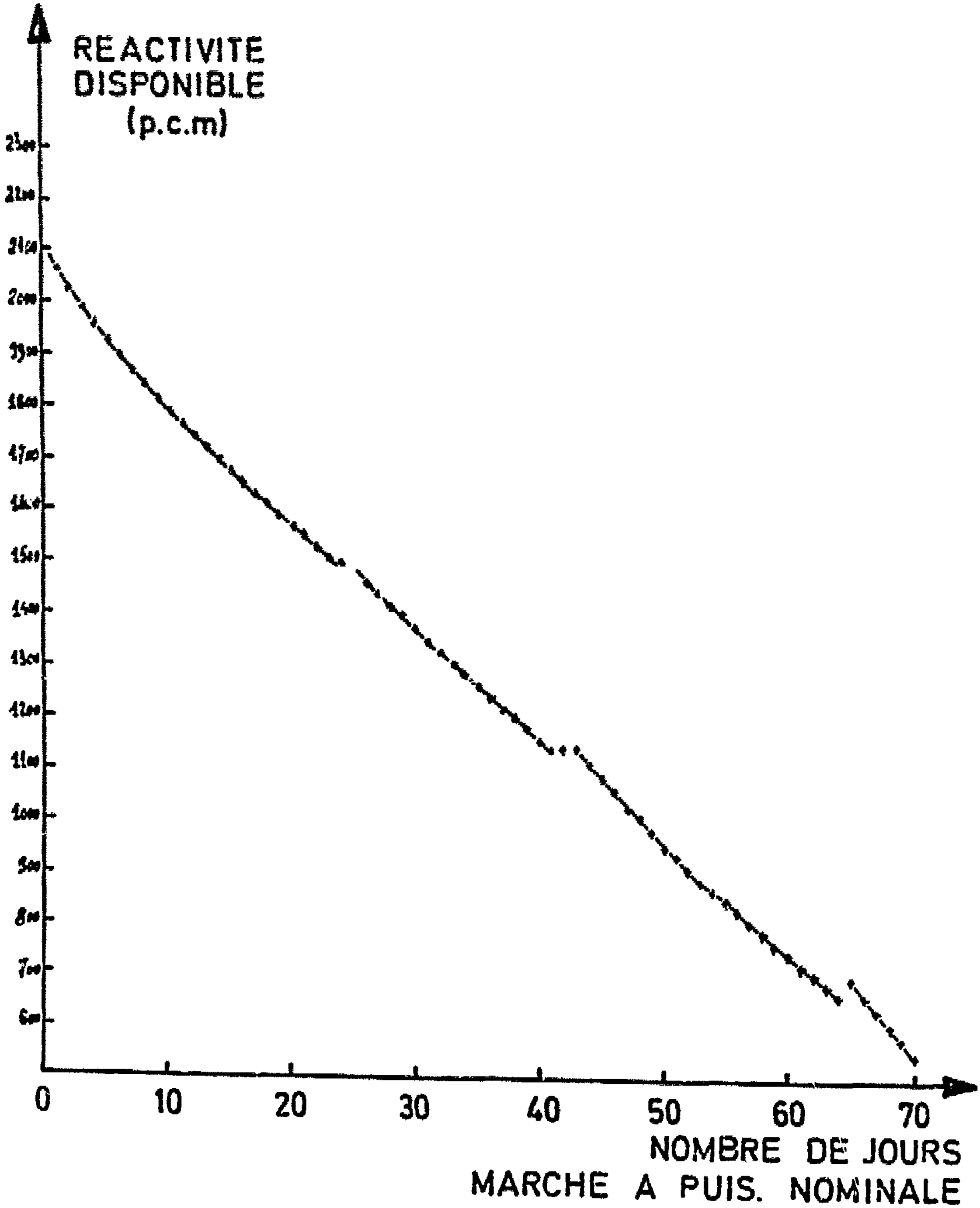
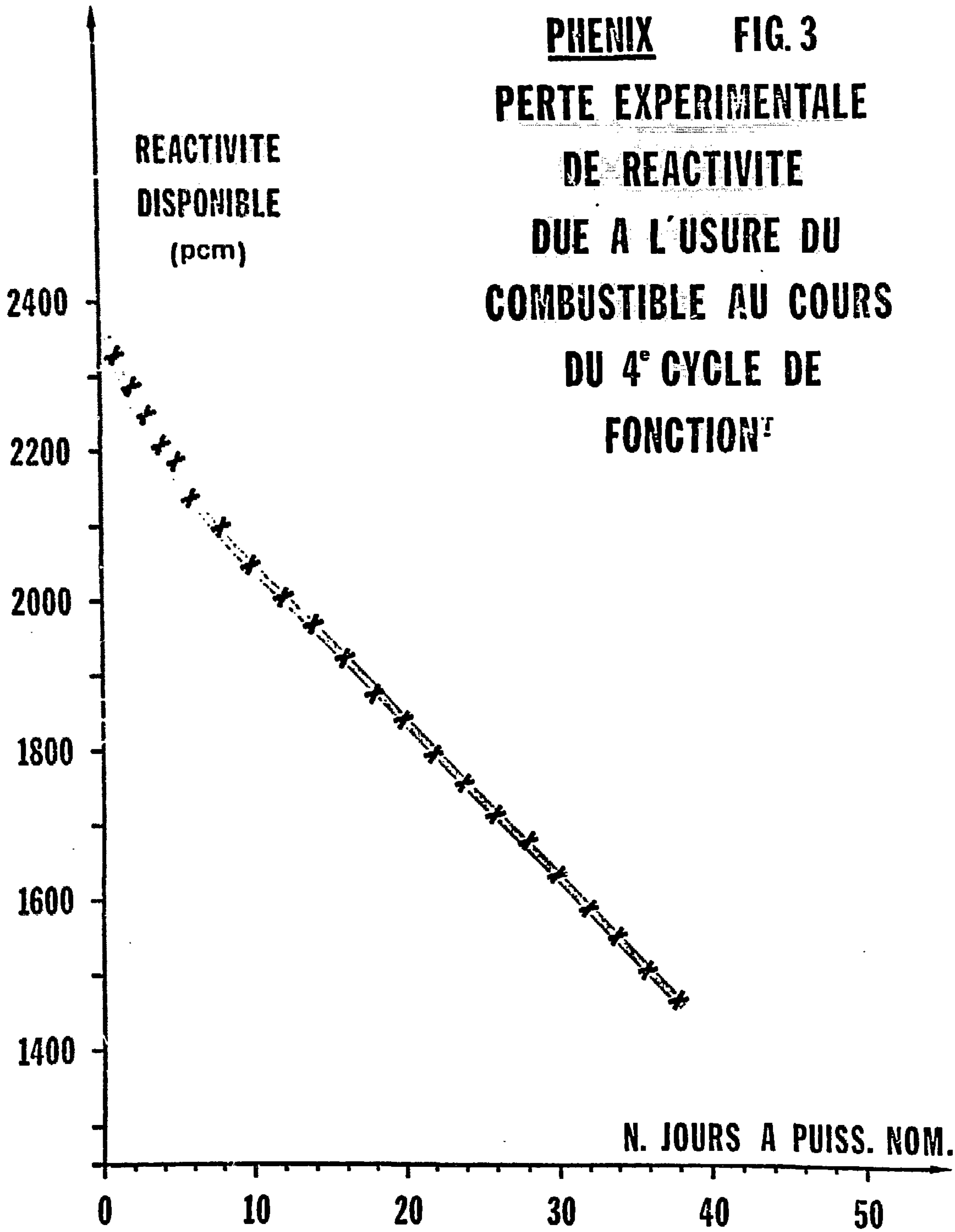


FIG. 2 PHENIX

PERTE EXPERIMENTALE
DE REACTIVITE DUE A L'USURE
DU COMBUSTIBLE AU COURS DU
3e CYCLE DE FONCTIONNEMENT





PHENIX **FIG. 3**

PERTE EXPERIMENTALE DE REACTIVITE DUE A L'USURE DU COMBUSTIBLE AU COURS DU 4e CYCLE DE FONCTIONT

C.E.N. CADARACHE 613
D.R.N.R.