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GENERAL OUTLINE OF THE OPERATION AND UTILIZATION OF THE BR2 REACTOR.

from 1969 to 1973

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SUMMARY

The reactor has operated at power levels ranging from 58 to 88 MW. The highest value of thermal neutron flux attained is $9 \times 10^{14} n.cm^{-2}.s^{-1}$, and the fast neutron flux (E > 0.1 MeV) reached a level of $6 \times 10^{14} n.cm^{-2}.s^{-1}$.

The experimental load increased considerably between 1969 and 1973, the rise being reflected by an average occupation of 76 channels at the end of the period compared with 45 in 1968.

The reactor has been used for development programmes concerning fast reactors, gas-cooled reactors, light water reactors and homogeneous reactors, as well as for applied research on fissile materials, structural materials and nuclear transmutations, for radioisotopes production, for pilot scale gamma irradiations and for basic research in nuclear and solid state physics.

1. INTRODUCTION.

The operation of the BR2 reactor and its associated installations is carried out under a contract concluded in 1960 between the Centre d'Etude de l'Energie Nucléaire (CEN/SCK) and the European Atomic Energy Community (EURATOM). Since 1968, the contribution of EURATOM has been limited to the detachment of personnel. Under the terms of this contract, the two partners use the installations jointly for their own purposes, but may also make them available to third parties.

In 1969 an agreement between CEN/SCK and the Gesellschaft für Kernforschung (G.f.K.) Karlsruhe came into force, under which the reactor occupation is shared equally by the German research centres (Karlsruhe, Jülich and others) on one hand, and by CEN/SCK and EURATOM on the other hand. This contract, signed for a first period of 5 years, is part of the collaboration programme on fast reactors between the Benelux and the Federal Republic of Germany. Special arrangements are made for experimentalists who are not covered by this agreement.

Responsibility for reactor operation and utilization is invested in the GEX (Groupe d'EXploitation du réacteur BR2 et de ses installations connexes). The group can provide a complete Irradiation Service, from design study to post-irradiation examination. The personnel totals 250, of whom 90 are for the preparation and operation of the irradiation experiments.

2. BRIEF DESCRIPTION AND SPECIAL FEATURES OF BR2

BR2 is a high-flux materials testing reactor, of the heterogeneous type, using 90 % enriched uranium as fuel, and beryllium and light water as moderator, the water also acting as coolant.

It has regularly operated with an experimental loading since 1963.

One of the interesting characteristics of BR2 is that the experimental channels are skew, the tube bundle presenting the form of a hyperboloïd of revolution (see fig. 1). This gives easy access at the top and bottom reactor covers, while maintaining a very high neutron flux at the core. One should stress the great flexibility of utilization, due to the fact that the fissile charge can be centred on different experimental channels. Although BR2 is a thermal reactor, it is possible to achieve neutron spectra very similar to those obtained in a fast reactor, either by the use of absorbing screen or by the use of fissile material within the experimental device.

3. REACTOR OPERATION

3.1. Core configuration

From 1969 to 1973, the reactor was operated with core configurations 6 and 7, centred on channel H 1 using one of the nine variants listed in table 1. The large number of variants utilized is due to continued attempts to use the flexibility of loading to the best advantage of the greatly varying experimental charge, while still maintaining the maximum length of operating cycle and keeping fuel consumption to the minimum.

It should be noted that when the sodium loop is loaded in channel H 1, the usual beryllium plug is replaced by an aluminium plug containing 6 special small diameter fuel elements, similar to the standard type but containing only the inner 3 fuel tubes; their main purpose is to compensate for the loss of reactivity caused by the loop.

Figures 2, 3 and 4 show the disposition of the fuel elements and control rods in the beryllium lattice for 3 typical core configurations.

3.2. Cycle

Reactor operation is carried out on the basis of operating cycles. In 1969 and 1970, the nominal cycle length was 3 weeks, and consisted of 6 days shut down for loading and preparation, followed by 15 days operation. This gave a theoretically attainable operating coefficient of 71.5 %.

From 1971 on, the cycle length was increased to 4 weeks, and consisted of 7 days shut down followed by 21 days operation, which gave a theoretical coefficient of 75 %.

On occasion, certain cycles had to be eliminated because of exeptional maintenance work, extensions to the routine shut downs and time lost through operating incidents.

3.3. Power

Nominal full power levels have been between 66.5 and 73.5 MW, depending on the core configuration used. During each cycle, the reactor power is modified, if necessary, to give the optimum irradiation conditions for the experimental load as a whole, while still respecting safety limits, and for this reason the actual full power levels have varied from 58.2 to 88.2 MW.

At each routine start up, it is customary to operate for short periods at 1 %, 10 % and 80 % of nominal full power, in order to carry out various checks, mainly on the irradiation conditions for the experiments, to determine the optimum power level and to find the best control rod alignment, having regard to flux perturbations and imbalance.

Several special campaigns have been carried out at powers between 1 MW and 52 MW, either for physics measurements or for the irradiation of fissile materials under tightly specified conditions.

3.4. Nuclear characteristics

Although the core configuration has been modified frequently, the essential characteristics of the irradiation conditions available for the experimental devices have not changed significantly. Table 2 gives, as an example, the nuclear characteristics for the channels of core configuration 7 B.

The maximum neutron fluxes attained are :

- thermal flux : $9.10^{14} n.cm^{-2}.s^{-1}$ (in channel H1 with Be plug)
- fast flux ($E > 0.1$ MeV) : $6.10^{14} n.cm^{-2}.s^{-1}$ (in a fuel element channel).

3.5. Operating Parameters

A typical set of operating parameters for the year 1972 is presented in table 3.

3.6. Calculation of the reactor loading

Due to the large number of highly absorbing rigs, to the loading of many series of rigs demanding identical irradiation conditions, to the generalised use of fuel elements containing burnable poisons, and to the presence of substantial amounts of ^3He in the beryllium moderator, the reactor loading has become so complicated that it is difficult to determine this by simple calculation. For this reason, a programme has been developped for the IBM 360-44 computer.

For each reactor cycle, each of the following steps is necessary :

- definition of the uranium content for each fuel channel ;
- determination of the thermal neutron flux, fast neutron flux and gamma heating for each fuel channel ;
- definition of the channel to be occupied by each rig to be loaded ;
- calculation of the heat flux at the hot spot and the corresponding maximum fuel plate temperature ;
- calculation of the excess reactivity, in order to determine the initial control rod heigth and the length of the cycle (in the BR2 reactor, all cycles normally continue until the control rods are fully withdran from the core) ;
- determination of the full power level ;
- calculation of the burn up of the cadmium absorbers in the control rods ;
- determination of the best alignment of the control rods ,

This method of calculation is considered satisfactory, bearing in mind that the majority of rigs have some sort of temperature regulation, and that a start-up with a new rig is carried out in a controlled manner. Nevertheless, in order to reconcile the often greatly divergent irradiation conditions demanded by the rigs, it has been necessary on occasion to vary the power during the cycle or to end a cycle with one or more rods partly inserted, or to take advantage of an enforced shut-down to modify the loading.

The effect of ^3He poisoning on the beryllium moderator has been measured several times, so that the evolution as a function of energy produced can be expressed as :

$$^{3}He \text{ reactivity } (\$ \text{ per shut-down day}) = -(7.3 \pm 0.6) \times 10^{-7} \times E$$

where E is the energy produced (MWd) and had a value of 1.18×10^{5} MWd at the end of cycle 1/73.

This moderator poisoning limits the length of any shut-down period, and will become more limitative as the fluence increases.

At the moment, with a routine shut-down of 7 days, the ^{3}He effect is easily compensated by the fuel charge; however, for a shut-down of several weeks, it would be necessary to unload certain of the heavily absorbing rigs before starting the reactor and burning up the poison. In this way, a shut-down of up to 4 months could be tolerated. Beyond this, in addition to unloading all experiments, special fuel elements without burnable poisons would have to be loaded and, if the shut-down continued too long, the beryllium would have to be replaced.

3.7. Operating incidents

The difference between the operating coefficients attained (55 to 65 %) and the maximum theoretically possible, (71.5 or 75 % depending on the cycles used) arises not only from two extensive shut-downs for exceptional maintenance work, but also from extension to routine shut-downs and operating incidents leading to unscheduled shut-down.

The two extensive shut-downs for exceptional maintenance work took place in 1970 and in 1971, the first being required for the repair of the concrete of the secondary circuit cooling tower basin, and the second for the replacement of the principal heat exchangers and the cooling tower packing.

The main causes of the extension to routine shut-downs were :

- physics measurement, such as finding the most suitable core configuration, determination of flux distribution and reactivity measurements including those necessary for the development of new fuel elements.
- preparations for special irradiation campaigns and exceptional work on rigs.
- special maintenance tasks, mainly concerning control rods , heat exchangers,primary pumps and leak tests of the containment building.
- a strike of CEN personnel.

The details of the various unscheduled shut-downs are given in Table 4.

Several of the operating incidents involved an inversion of the primary coolant flow within the core. This was because the primary circuit pressure had been reduced, to minimise leaking in the old heat exchangers, and the fall of temperature after some scrams was enough to reduce the primary pressure to the level at which the isolating valves closed. No damage was caused by these incidents, of which there were 11 in total, and which may be considered simply as unforeseen repetitions of the flow failure tests carried out in September 1963 and in June 1965.

As stated above, before the main heat exchangers were replaced, the primary circuit pressure was reduced progressively (as the hot spot allowed) so as to reduce the leaks from the primary circuit to the secondary circuit; this constituted a temporary modification of the general operating characteristics of the reactor. (known leaks from primary to secondary circuits, reduced primary pressure). The safety assessment of these modifications showed them to be acceptable both for the reactor and for its environment, on condition that certain specific operating instructions were applied.

In 1969, in 1970, and again in 1973, exceptionaly high levels of fission product activity were detected in the primary circuit. These high levels led to several premature shut-downs, either to localise the source, or whenever the ^{131}I level attained a value of 10^{-2} μCi.cm^{-3} (which corresponds to 5 % of the maximum credible accident figure and which was the maximum tolerated level).

An analysis of the observations carried out leads to the following conclusions :

- no rig having fissile material in single containment has ever leaked fission products.
- the source of the fission products was the fuel elements, both the alloy type after 2 to 4 years service, in demineralised water, and the new cermet type, of which one prototype and certain 1973 elements were faulty.

For the faulty prototype, an examination of the radiographs of the plates revealed, in one plate, a large UAl_3 grain whose dimensions should have led to the rejection of this plate before assembly. The grain size was such that the cladding thickness over the grain was virtually reduced to zero. The inclusion of this grain in the plate was due to a manufacturing fault during the preparation of the powder, and a modification to the procedure eliminated the possibility of this fault in the production plates.

On the other hand, the reasons for the 1973 failures are still being sought, although it would seem that some manufacturing or inspection failure could be the cause.

3.8. Operating evolution

Tables 5, 6 and 7 give a general view of the operating characteristics of the reactor.

3.9. Future plans

As far as the future is concerned the BR2 development programme aims at the adaptation of the irradiation capacity so as to accomodate the expected experimental loading. The aspects being specially considered at the moment are : :

- the irradiation conditions which would be obtained under a cadmium screen in the central channel H 1 or one of the peripheral 200 mm channels, for a loop surrounded by the small diameter S IIIs fuel elements or by a ring of elements of the ATR type.
- a new configuration to permit the simultaneous irradiation of two loops in the peripheral channels H3 and H4 under the required conditions, while still maintaining the high values in the central chanel H1.
- the maximum operating power.
  In the future, it will certainly become necessary to increase both the specific power and the total power; the specific power will need to rise from its current permitted level of 470 $W.cm^{-2}$ to 550 or even 600 $W.cm^{-2}$ at the hot spot and the total power will exceed 100 MW. These increases will be necessitated by the concurrent irradiation of several large loops and by the high fluxes they will require.
- an improved flow distribution of the primary coolant in the different reactor channels proportional to the power dissipated in the channel.

4. REACTOR EQUIPMENT

4.1. Fuel Elements

Cermet elements have been used regularly from cycle 1/71 onwards, although prototypes had been tested as early as 1969. Since January 1971, the majority of fresh elements loaded have been of this type. Those used until early 1972 had a fuel loading of 50 mg. ^{235}U per square centimetre, which gives, for a standard 6 tube element, a total

^{235}U weight of 330 g, compared with only 244 g for the alloy type. In addition, 2.8 g natural boron in the form of B_4C and 1.4 g natural samarium in the form Sm_2O_3 are incorporated into each element as burnable poisons. From cycle 4/72 onwards, a new type of cermet element has also been utilized, containing 60 mg ^{235}U per square centimetre, and giving a ^{235}U content for a 6 tube element of 400 g. The samarium content has been maintained at 1.4 g, but the boron content has been increased to 3.8 g

The generalised use of these new cermet elements has led to an increase in the reactor cycle length, a reduction in the variation in irradiation conditions during the cycle caused by the control rod movement (see figure 5), the possibility of overcoming the ever-increasing reactivity absorbtion effect of the rigs and a reduction in fuel costs by decreasing the number of fresh elements used each cycle and by increasing the mean burn-up.

To illustrate the increase in cycle length, two examples can be quoted :

- with the experimental loading of cycle 5/72, alloyed elements would have given a cycle length of 10 to 14 days, whereas the cermet elements gave 26.5 days.
- during cycle 8/73, the total energy dissipated was 2.505 MWd, compared with the maximum ever attained with alloy-elements of 1.044 MWd in cycle 7/69.

The alloy elements which contain no burnable poisons, are now used only for two specific applications : either to obtain a higher thermal neutron flux in some particular channel or to increase the reactivity available at start up.

The highest rated alloy element has been taken to an average burn-up of 54 %, which means 67 % at its hot-spot; corresponding figures for the cermet elements are 68 % mean burn-up and 79 % at the hot-spot.

4.2. Cooling system

In 1969 and 1970 there were frequent leaks in the main heat exchangers. Investigations revealed that these were partly due to vibration of the tubes against the baffle plates, but arose mainly from corrosion initiated by impurities in the secondary circuit water coming from the cement used to repair the cooling tower basin.

In 1971, the nine original heat exchangers, of the classical straight tube pattern, were replaced by three units, of a helical tube pattern (see figure 6). In January 1972, it was found that one tube of the new exchangers was perforated, this time by vibration of the unsupported straight lead-in section against a weld. After removal of the faulty tube, extensive and systematic vibration measurements were carried out, and it was found that, in two of the heat exchangers, several tubes experienced excessive vibration.

As an initial corrective measure, the manufacturer installed aluminium straps around the lead-in tube bundles, to reduce the vibration of the enter tubes. Subsequently, a woven stainless steel mat, 5 cm thick, was wrapped around the bundles to break up the impinging jet of primary water, which enters at right angles to the tube bundle. These modifications were carried out on all three exchangers.

No new leak has occurred since that time, and the noise due to vibration has been greatly reduced if not entirely eliminated, so that we consider the modifications to be satisfactory as a short term measure.

The original wooden cooling tower packing was replaced by plastic material. This modification of the towers and the replacement of the heat exchangers have led to a nominal cooling capacity of the system exceeding 100 MW.

4.3. Miscellaneous

Various works have been carried out either to facilitate the use of the reactor by the experimentalists or to improve the safety and efficiency of the system.

The following are the most important :

- the commissioning of the neutronography apparatursinstalled in the reactor pool. Extremely satisfying results have been obtained, especially on capsules containing fissile materials, some of which also incorporate a cadmium screen, on the control rods, to measure cadmium burn-up and on UO_2-PuO_2 fuel pins, to measure the microdensity. (some examples are shown in figure 7)
- the installation of a T 2000 data handler for the reactor and rigs.
- the modernisation of the reactor control instrumentation, and the design of new circuits, especially for power regulation.
- the completion of a flux measuring device for individual reactor channels, based on sampling of the outlet water and the determination of the ^{16}N and ^{19}O activities.
- the putting into service of various measuring equipment for primary circuit activity, including continuous monitoring of α activity, gamma spectrometry by Ge - Li detectors, and determination of activities of special interest such as ^{170}Tm.
- the installation of a rapid response measuring chain for fission product detection in the secondary circuit.
- the installation of a sampling point for the primary coolant leaving individual reactor channels in which singly-contained fissile material is being irradiated.

- the setting-up of a high resolution spectrometer for the checking of the burn-up level in fuel elements.

- the loading of new control rod absorbers, which contain cobalt grains at their lower end.

- the commissioning of an apparatus for the underwater measurement of the activity of capsules containing ^{60}Co.

5. REACTOR UTILIZATION

5.1. Facilities available

In BR2, the following facilities are available :

- in the pressure vessel
  - . core - within the fuel elements
    - in a special plug in the central channel
  - . reflector - in the beryllium (or aluminium) plugs
    - in the hydraulic rabbit
    - in the self-service thimble
- outside the pressure vessel
  - . in the beam tubes (radial or tangential)
  - . in the reactor pool

Table 8 shows the irradiation positions and the neutron fluxes available.

5.2. Irradiations carried out

Table 9 lists the irradiation experiments carried out in BR2 from 1969 to 1973.

The reactor has been used for experiments concerned not only with fast reactors, gas cooled reactors, light water reactors, and homogeneous reactors, but also with applied research on fissile materials and structural materials; in addition, irradiations have been carried out for nuclear transmutations especially for the production of radioelements of very high specific activity. Various gamma irradiations were also undertaken and the beam tubes were utilized for basic physics studies.

Between 52 and 76 reactor channels have been utilized by experimental equipment. The average distribution of the use of the channels is given in table 10.

Figure 8 shows, as an example, the relative distribution of the irradiations which took place in 1972 in the core and in the reflector, not counting the multi-purpose devices, measuring probes, and the nuclear models of FAFNIR rigs necessary to balance the reactor loading. The irradiations are divided into the following categories :

- study of the effects of radiation on
  - . fuel elements
  - . structural materials
- cobalt 60 production
- production of various radioisotopes
- transplutonium studies and direct conversion (actinium production).

Comparison is made on the basis of "Fast Irradiation Units" (U.I.R.). One UIR is defined as the product of a unit volume (1 cm^3) and a unit dose (10^{20} $n.cm^{-2}$) of neutrons of energy exceeding 100 KeV. The fast flux is used since BR2 irradiations are mainly concerned with the fast reactor programme; in 1972 irradiations for the sodium cooled fast reactor programme accounted for 56 % of the reactor utilization and the high temperature reactor programme accounted for a further 20 %.

5.3. Evolution of reactor utilization

Figure 9 shows graphically the values of the utilization coefficients for the core and for the reflector from 1969 to 1973.

Figure 10 shows the evolution in reactor occupation between 1963 and January 1973*. The following data has been incorporated in figure 10 :

- the total number of channels occupied at any one time in the core, in the reflector and in the pool.
- the hourly production in the core and in the reflector, not counting the hydraulic rabbit HR 1, the self-service thimble DG 36, the measuring probes and the FAFNIR models.
- the annual production in the core and in the reflector.

An analysis of the curves of figure 10 leads to the following conclusions :

- a growing interest has been shown in BR2, as indicated by the continuous increase in the number of reactor channels occupied (see also figure 11)
- the volume of services rendered increased regularly from 1963 to 1966, as shown by the curve of annual production. In 1967 there was a decrease in production due to the loss of running-time occasioned by the alpha-contamination incident, together with a decrease in the hourly production of ^{60}Co and other isotopes. In 1968 the increase recommenced and the production in 1972 was the highest ever attained.
- hourly production fluctuates strongly from cycle to cycle, due to the inherent nature of a test reactor and to variations in the demand for radioelements.
- hourly production in the core is strongly influenced by the presence of the sodium loop in the H 1 channel.

6. CONCLUSIONS

From 1969 to 1973 the BR2 reactor was utilized regularly, its operating characteristics being continuously adapted to the needs of

*provisionally, awaiting the final figures for the year 1973 the experimental load. For this reason, the full power level increased progressively from 67 MW to 85 MW, while the reactor occupation doubled, being 66 on average between 1969 and 1973 against 32 for the period 1963 to 1968.

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BR2 - Configurations 6 and 7. Variants utilized from 1969 to 1973.

Variants	Utilized in	Number of fuel elements	Control rods	Plug in H 1	Flux traps*	Reactor power (nominal) MW
6K - Be	1969, 1970	38	8 compensation 1 or 2 regulation	beryllium	0	66.5 to 68.5
6K	1969, 1970, 1971	38	" "	Al + 6 elements type SIIIs	0	70 and 71
6L	1969	35	" "	" "	3	67
6M	1969	36	" "	" "	2	69.5
6N	1969	37	" "	" "	2	70
60-Be	1969	37	" "	beryllium	1	67
7A	1971	38	6 compensation 1 or 2 régulation	Al + 6 elements type S IIIs	4	70
7B	1972, 1973, 1971	38			2	70 and 73,5
7B - Be	1972, 1973	38	" "	beryllium	2	70

* a reflector channel surrounded by fuel elements and thus possessing an elevated thermal neutron flux.

TABLE 1

TABLE 2

Nuclear characteristics of the BR2 configuration 7 B

Power = 73.5 MW

Conditions at start of cycle 10/72

Reactor channel	Fuel elements		Maximum flux (1) 10^{14} n.cm^{-2}.s^{-1}		φ fast / φ thermal	Maximum nuclear (3) heating (W.g^{-1} Al)
	Number of tubes	Average burn-up (%)	Thermal (2)	E>0.1 MeV		
A 30	6N G	10	2.47	4.64	1.88	13.9
A150	6I G	12	2.48	4.51	1.82	13.4
A210	6N G	10	2.43	4.57	1.88	13.4
A330	6N G	10	2.47	4.64	1.88	14.0
B 0	6N G	0	2.18	4.82	2.21	14.3
B 60	6N G	0	1.99	4.39	2.21	12.5
B120	6I G	0	2.07	4.93	2.38	13.6
B180	6N G	0	2.18	4.82	2.21	13.4
B240	6I G	0	2.07	4.93	2.38	13.4
B300	6N G	0	1.99	4.39	2.21	12.8
C 41	6I G	17	2.21	4.38	1.98	11.7
C 79	6N C	30	2.60	2.41	0.93	8.6
C101	5N C	31	3.20	3.99	1.25	10.2
C139	5N A	22	3.39	2.84	0.84	10.0
C221	5N A	23	3.39	2.74	0.81	9.8
C259	5N C	54	3.84	3.28	0.85	8.9
C281	6N G	0	1.76	3.75	2.14	10.1
C319	6I G	17	2.20	4.39	1.99	12.0
D 0	6N G	21	1.98	3.00	1.51	10.0
D120	6N G	21	2.31	4.58	1.98	11.1
D180	6N C	39	2.28	1.78	0.78	7.4
D240	6N G	20	2.29	4.59	2.00	10.9
E 30	6N G	18	1.70	3.51	2.06	9.7
E330	6N G	40	2.06	3.20	1.55	8.9
F 14	6N C	30	2.01	2.03	1.01	7.6
F 46	6N C	44	2.24	1.44	0.64	6.8
F106	6N C	44	2.26	1.44	0.64	6.4
F166	6N G	30	1.72	1.81	1.05	5.6
F194	6N G	31	1.73	1.79	1.03	5.6
F254	6N C	31	2.03	2.00	0.99	6.8
F314	6N G	0	1.34	2.96	2.21	8.7
F346	6N G	17	1.62	2.69	1.66	8.2
G120	6N G	17	1.19	1.86	1.56	5.9
G240	5N C	0	1.08	1.73	1.60	5.5
H 23	6N G	0	1.02	2.25	2.21	6.6
H 37	6N G	0	1.02	2.25	2.21	6.4
H323	6N G	0	1.02	2.25	2.21	6.3
H337	6N C	43	1.68	1.15	0.68	5.2
H1(4)	3S C	19	—	—	—	9.1

A Fuel element "alloy"
C Fuel element "Cermet" 330 g type (50 mg ^{235}U.cm^{-2})
G Fuel element "Cermet" 400 g type (60 mg ^{235}U.cm^{-2})

(1) Neutron flux at 150 mm below reactor mid plane
(2) Unperturbed neutron flux on the centre line of the fuel element $[n]_0^{0.5} V_0$ at 2200 m.s^{-1}
(3) Nuclear heating of an aluminium sample placed on fuel element axis at position of maximum flux
(4) At the centre of the aluminium plug, which incorporates a cadmium screen and six fuel elements with 3 concentric tubes (Type SIIIs)

TABLE 3

BR2 - Operating parameters 1972

Reactor operating characteristics	
Power rating, nominal	70 and 73.5 MW
Total operating time	230.4 days
Average operating time per cycle	20.9 days
Energy dissipated, total	17,568.2 MWd
Energy dissipated per cycle, average	1,597.1 MWd
Fissile charge at start of cycle (^{235}U)	10.5 to 12.4 kg

Circuits characteristics	Primary	Secondary	Pool
Flow ($m^3.h^{-1}$)	6,700 - 7,000	4,000 - 5,000	400
Inlet temperature (°C)	40 - 50	20 - 30	35 - 45
Inlet pressure ($kg.cm^{-2}$)	12.5	—	—
Δp across pressure vessel ($kg.cm^{-2}$)	2.8	—	—
pH	5.3 - 7.0	5.5 - 6.5	5.5 - 6.5
Resistivity ($\Omega.cm^{-1}$)	$2-6\times10^6$	10^5	10^6
Flow in purification line ($m^3.h^{-1}$)	20 - 25	100	20
Activity $\beta\gamma$ ($\mu Ci.cm^{-3}$)	$<10^{-1}$	$<10^{-6}$	$<10^{-4}$

Fuel elements	
Temperature, maximum nominal at hot spot (°C) for inlet temperature of 45°C	143 - 154
Heat flux maximum ($W.cm^{-2}$)	394 - 455
Water velocity between plates ($m.s^{-1}$)	9.4

Liquid effluent production		
Cold	$<10^{-5}$ $\mu Ci.cm^{-3}$	2,269 m^3
Warm	10^{-5} to 10^{-2} $\mu Ci.cm^{-3}$	10,302 m^3
Hot	$\geq10^{-2}$ $\mu Ci.cm^{-3}$	1,127 m^3

Gaseous effluents	
Ventilation of containment building	38,000 $m^3.h^{-1}$
Ventilation of machine hall	83,000 $m^3.h^{-1}$
Radioactivity at chimney	5×10^{-8} $\mu Ci.cm^{-3}$

Consumption	
Electricity	21×10^6 kWh
Demineralized water	436,132 m^3
Uranium-235	21.1 kg

TABLE 4

BR2 - Operating incidents (unscheduled shut-downs)

Type of incident	1969		1970		1971		1972		1973 ※	
	Number	Duration (h)	Number	Duration (h)	Number	Duration (h)	Number	Duration (h)	Number	Duration (h)
Control rod failures	4	46.6	3	46.3	2	91.5	1	50.4	1	0
Heat exchanger leaks	3	168.5	2	14.7	-	-	-	-	-	-
Fission products in the primary circuit	-	-	3	142.2	-	-	-	-	4	268.0
Other reactor equipment failure	6	69.7	6	159.6	7	43.7	5	127.5	3	162.3
Human errors	3	138.4	1	0.4	1	3.4	2	47.8	1	0.3
Experiments										
- Initial irradiation conditions not satisfactory	1	22.9	-	-	4	173.2	2	55.4	3	174.7
- Rig failure	2	126.5	1	14.4	1	55.4	4	227.1	3	63.9
- Parasitic signal	2	0.7	-	-	-	-	-	-	-	-
Total	21	573.3	16	377.6	15	367.2	14	508.2	15	669.2

※ preliminary results

TABLE 5

Evolution of the operating parameters of the BR2 reactor.

A. General data since date of commissioning.

Year	Configuration	Number of days operating	Nominal power (MW)	Energy produced (MWd)
1963	5	154.5	17 - 34	3,292
1964	5	226.5	34	7,595
1965	5 and 6	189.8	34 - 57	8,599
1966	6	219.8	57	12,576
1967	6	174.4	57 - 63	10,564
1968	6	213.9	58.5 - 67	13,168
1969	6	229.3	66.5 - 71	15,155
1970	6	218.0	67 - 70	15,057
1971	6 and 7	203.9	70	14,478
1972	7	230.4	70 - 73.5	17,568
1973*	7	248.0	70-73.5	20,521
Totals		2,308.5		138,573

* preliminary results

TABLE 6

Evolution of the operating parameters of the BR2 reactor

B. Operating analysis 1969-1973

	1969		1970		1971		1972		1973 ※	
	DAYS	%	DAYS	%	DAYS	%	DAYS	%	DAYS	%
REACTOR STATE										
Operational	229.3	61.8	218.0	59.5	203.9	55.2	230.4	64.7	248.0	65.7
Routine shut-downs	96.0	25.9	96.0	26.2	70.0	19.0	77.0	21.6	77.0	20.4
Extension of routine shut-downs	22.1	5.9	5.9	1.6	13.0	3.5	27.3	7.7	24.5	6.5
Unscheduled shut-downs	23.9	6.4	15.7	4.3	15.3	4.1	21.2	6.0	27.9	7.4
Large scale maintenance work	-	-	30.5 (1)	8.4	67.2 (2)	18.2	-	-	-	-
Totals	371.3	100	366.1	100	369.4	100	355.9	100	377.4	100
OPERATING COEFFICIENTS										
Theoretical		71.5		71.5		75		75		75
Actual		61.8		59.5		55.2		64.7		65.7

(1) Repairs to cooling tower basin

(2) Replacement of main heat exchangers and cooling tower packing

※ Preliminary results

Evolution of the operating parameters of the BR2 reactor

C. Reactor power and uranium consumption

	1969	1970	1971	1972	1973 ⁑
CORE CONFIGURATION					
Plug in channel H 1					
- beryllium	6K and 60	6K Be		7B Be	7B Be
- aluminium + 6 IIIs fuel elements	6K,6L,6M,6N	6K	6K 7A 7B	7B	7B
POWER (1)					
- nominal (MW)	66.5 to 71	67 and 70	70	70 and 73.5	70 and 73.5
- integrated (MWd)	15,155	15,057	14,478	17,568	20.521
URANIUM CONSUMPTION					
- Mass of ^{235}U (2) g	18,228	18,119	17,415	21,132	24,683
- Number of elements "alloy"	165 + 12IIIs	177 + 24IIIs	53 + 18IIIs	11	13
"cermet"	6	2	114 + 6 IIIs	142 + 18 IIIs	115 + 30IIIs
- Effective average burn-up (3) (%)	44.0	40.7	33.2	38.0	48.0
- Specific (3) (elements/10^3 MWd)	11.3	12.5	12.3	9.0	6.7

(1) Read from ^{16}N measuring chain
(2) Thermal balance
(3) Fresh element loaded in standard channels
⁑ preliminary results

TABLE 7

TABLE 8

Irradiation positions available in the BR2 reactor.

Configuration : 7B or 7B-Be   power : 88 MW

Irradiation position		useful dimensions mm	maximum nuclear characteristics		
			neutron flux		nuclear heating
type	number		thermal	E > 0.1 MeV	W/gr Al
1. in the pressure vessel					
- core					
. within the fuel elements	38	17.4 to 42.4	4.2×10^{14}	6×10^{14}	17
. in the central channel	up to 1	up to 200	- (3)	4×10^{14}	12
- reflector					
. 200 mm holes	up to 5	29.5 to 200.2	10^{15}	1.5×10^{14}	8
. 85 mm holes	19 to 34(2)	29.5 to 80.6	4.5×10^{14}	2.2×10^{14}	6
. 50 mm holes	10	29.5 to 46	1.3×10^{14}	5×10^{13}	2
2. outside the pressure vessel					
- beam-tubes	9	miscellaneous	10^7 to 5×10^{13}	-	-
- reactor pool					
. radioisotopes tubes	4	80	3×10^{13}	5×10^{12}	< 0.5
. pneumatic rabbit	1	21.8 x 8.7	10^{13}	-	-
. neutroradiography	1	80	3×10^7	-	-

remarks : (1) total number of irradiation positions : up to 97
(2) one 200 mm channel is equivalent to three 85 mm channels
(3) under cadmium screen

BR2 - PROGRAMME DES IRRADIATIONS -

PAYS	ORGANISMES	EXPERIENCES	CANAL TYPE	1969	1970	1971	1972	1973
ALLEMAGNE	BABCOCK	Boucle CO_2 graphite	D 180					
	G.f.K. KARLSRUHE	Mol 1 mat. de gainage	B120					
		Mol 2 "	CU C39					
		Mol 3 "	A - F					
		Mol 4 "	D180					
		Mol 5 "	C - D					
		Mol 6 Pu - Am - Np - Cm - H1	D120.H5.B0.H1 D60.D300.P					
		Mol 7 boucle Na en H1 -(OXYDES)	H 200mm					
		Mol 8 FAFNIR UO_2 - PuO_2	H à E					
		Mol 10 UO_2 - PuO_2 (Fluage radial)	K					
		Mol 11 UC - PuC	C et F					
		Mol 12 UO_2 - PuO_2 (Fluage axial)	L à K					
		Mol 13 UO_2 - PuO_2 (anneau)	K					
		Mol 14 PACT	Refl. Coeur					
		Mol 15 Carbure	K					
		Mol 16 Oxyde (C F C)	L - H					
	K.F.A.-JULICH	MOPS B Particules enrobées	C39 C255					
		MOPS. C Particules enrobées	G à A					
		MGR4 Graphite	B					
		Divers : Am. La. Ag	H2/210°-LO-C119					
		M.N.K. Barreaux U.C.N.	P à H					
		Email (Siemens)	C					
		UC_2	F					
	SIEMENS	Boucle CO_2 - élément comb. UO_2	L120 - H200mm					
	AEG	Barreau antimoine	D180					
	T.H.T.R. (K.F.A)	Particules enrobées MOPS. A	K à C					
BELGIQUE	C.E.N. BELGONUCLEAIRE	Boucle MFBS en H1	H1					
		Boucle MFBS en H3	H3					
		C.E.B.	P à G					
		P_2	F et N					
		FAFNIR UO_2 - PuO_2	C et F					
		Convoyeur hydraulique	K					
		Soufflet acier	F					
		Capsule VC particules enrobées	G à B					
		ACEC - MMN UO_2 - Gd_2O_3 (P6-10-11-13)	P à K					

TABLE 9

	C.E.N. métallurgie		
		Aiguille gaine acier ferritique	K - L
		Essais de compatibilité (CFC)	K-L
		Acier à chaud	A 309-D310
		Acier à froid	A 30
		S.E.N.A. UO_2 + Dy_2O_3	L à K
		Carbure POM et CIRCE	N à K
		F.F.C. UC-PuC	C
		Convoyeur hydraulique	K
		UO_2 Conductibilité thermique	Pou N
	C.E.N. phys. état solide	Diffraction neutronique	R1
	C.E.N. phys. neutron	Diffusion inélastique	R2
		Section efficace diffusion	R3
		Gamma de capture	R4 ou R5
		Mesures sect. efficaces U5 Pu9	T7
		Spectromètre à cristal γ	T2-T5
		Convoyeur pneumatique rapide	Piscine
		Spectromètre neut. à 3 axes	R4
	C.E.N. I.R.E.	Iridium	A,C,H,D 80
		Divers	A à C,H
		Convoyeur hydraulique	K
		D.G.	K ou D
		Tube P	Piscine
		Cobalt	N à M
	C.E.N. safeguards	Mesure de contrôle	T
	C.E.N. U.M.	Ra (Projet actinium)	E 30 ou D 300 et C
	UNIVERSITES	Divers	Divers
PAYS BAS	K.E.M.A.	UO_2-ThO_2 en suspension	L à K
	T.N.O.	Acier	C
EURATOM	ISPRA	Tm	M1
	KARLSRUHE	Pom-U-Pu (O_2-C-N)	L à K
		C.A.M.E.L. UN-PuN	F
	H.T.G.R.	Particules enrobées	O à B

avec BH Dragon et KfA

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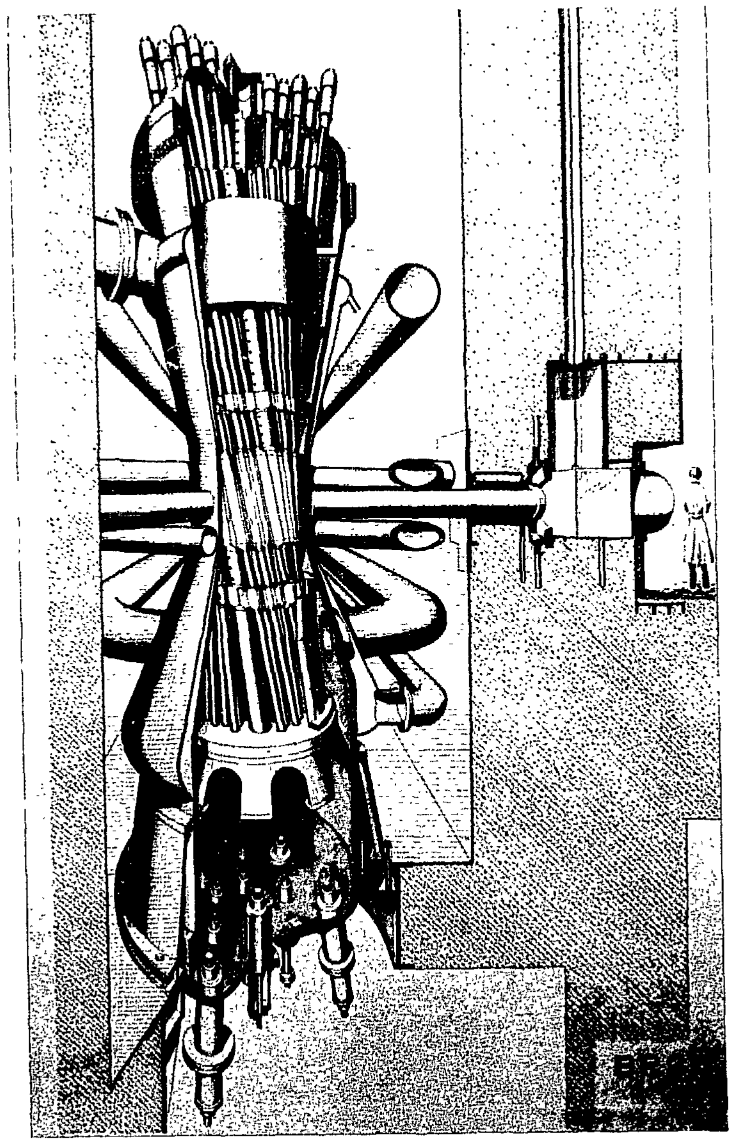
TABLE 10

BR2 - Average distribution of the uses of the channels.

Purpose	number of channels occupied				
	1969	1970	1971	1972	1973*
1. in the pressure vessel					
study of radiation effects					
. fuel and fuel pins	10	15	24	28	25
. structural material	7	9	6	6	7
transmutation programmes					
. production of actinium and transplutonium elements	1	2	4	3	3
. cobalt 60	18	17	16	13	12
. various radioisotopes	2	3	3	4	4
multi-purpose equipment (rabbit, self-service timble)	3	2	2	2	2
measuring probes and mock-ups	4	4	8	7	8
beryllium matrix samples	-	-	-	-	7
2. outside the pressure vessel					
fundamental physics research	6	6	7	7	6
multi-purpose equipment (pooltubes, neutro-radiography)	1	1	1	2	2
Total	52	59	71	72	76

* preliminary results

FIG.1



GENERAL VIEW OF THE BR2 REACTOR

FIG. 2

BR 2 CONFIGURATION 6K.Be

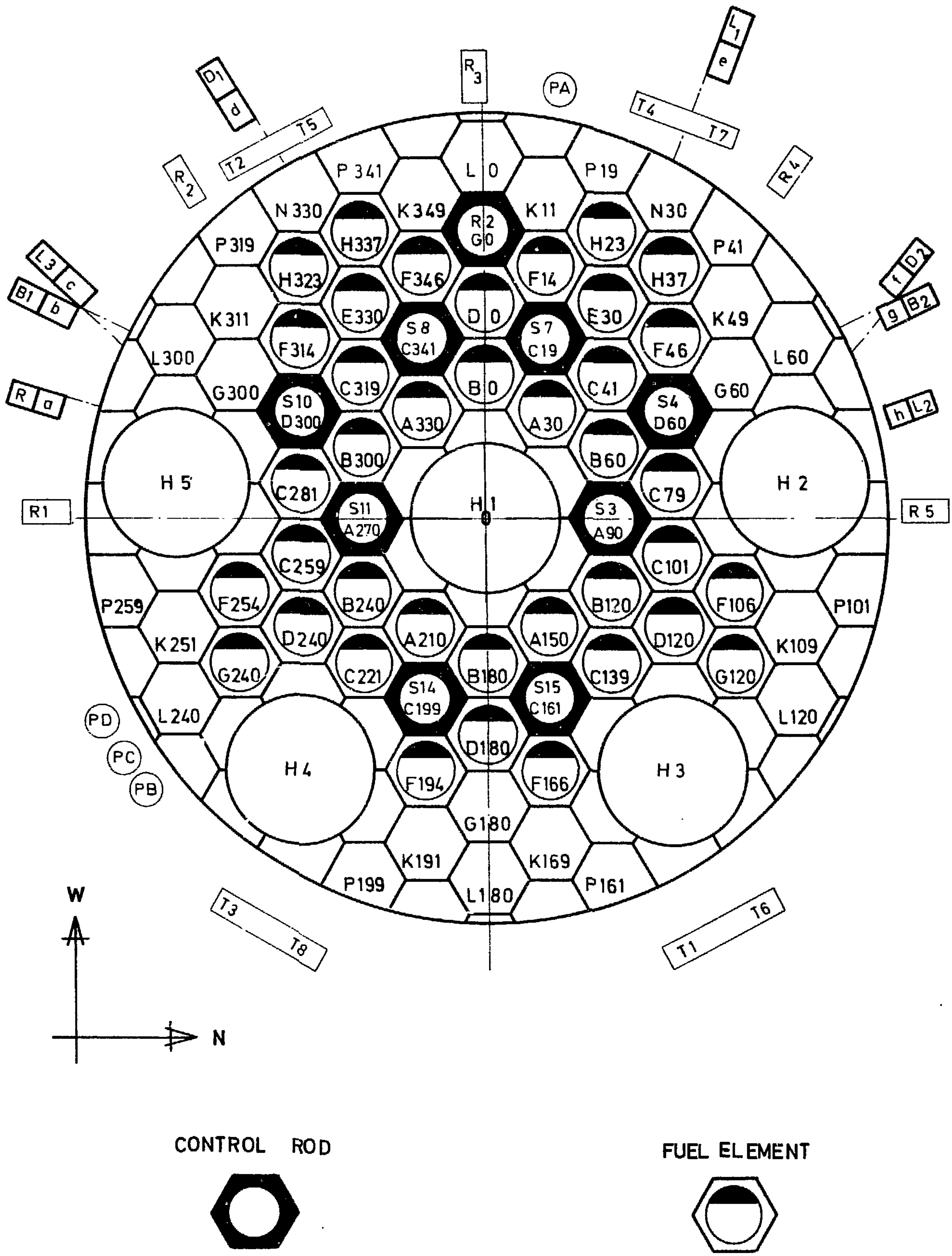


FIG.3

BR2 CONFIGURATION 7A

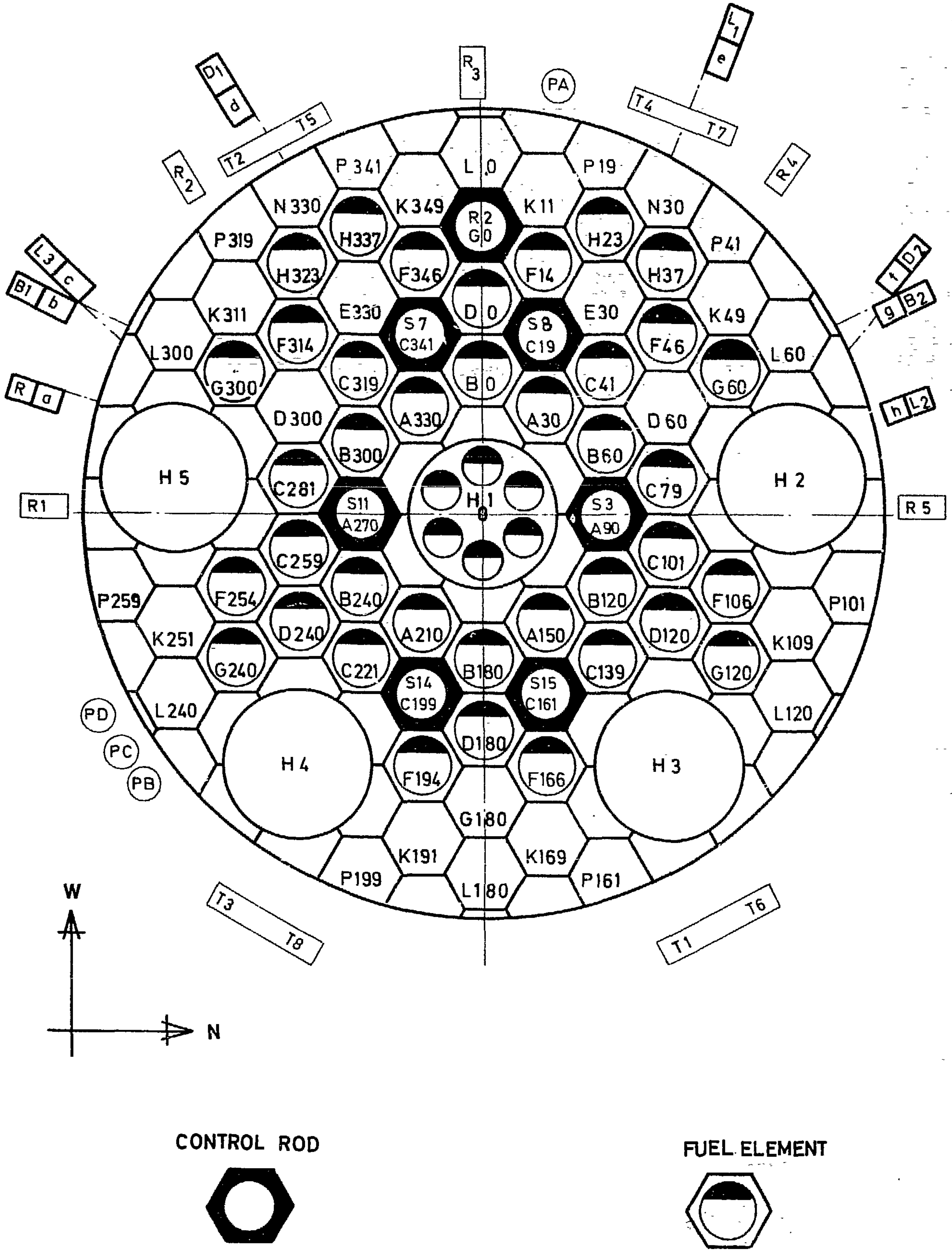
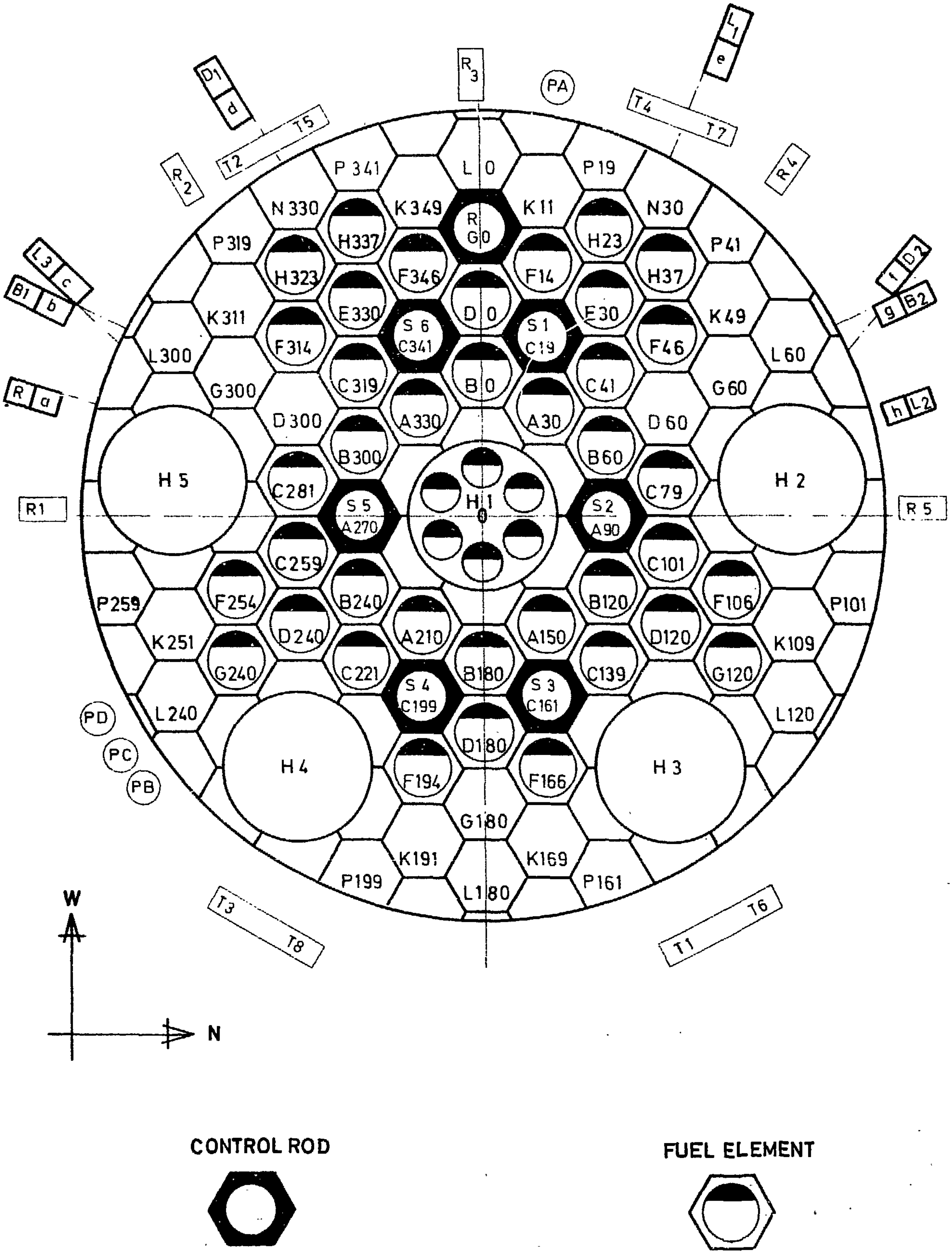
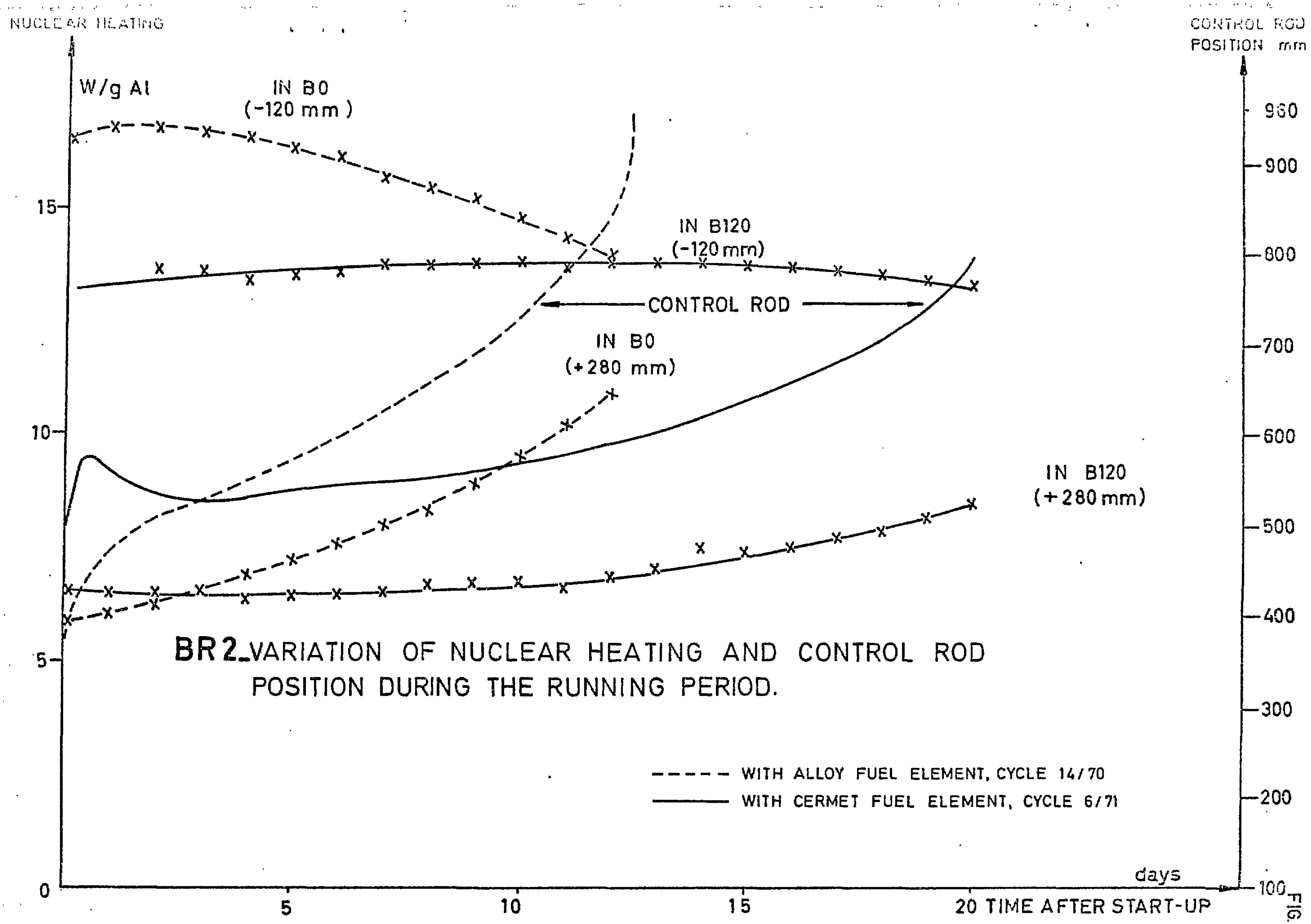


FIG.4

BR2 CONFIGURATION 7B



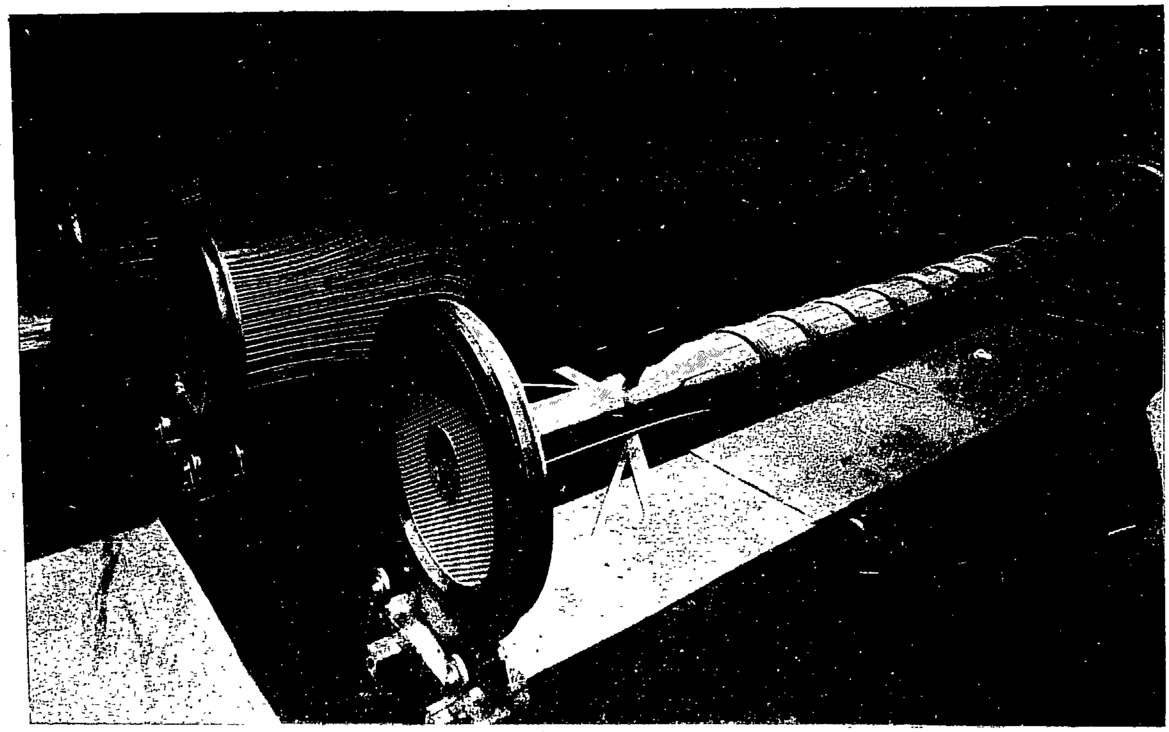


BR 2_VARIATION OF NUCLEAR HEATING AND CONTROL ROD POSITION DURING THE RUNNING PERIOD.

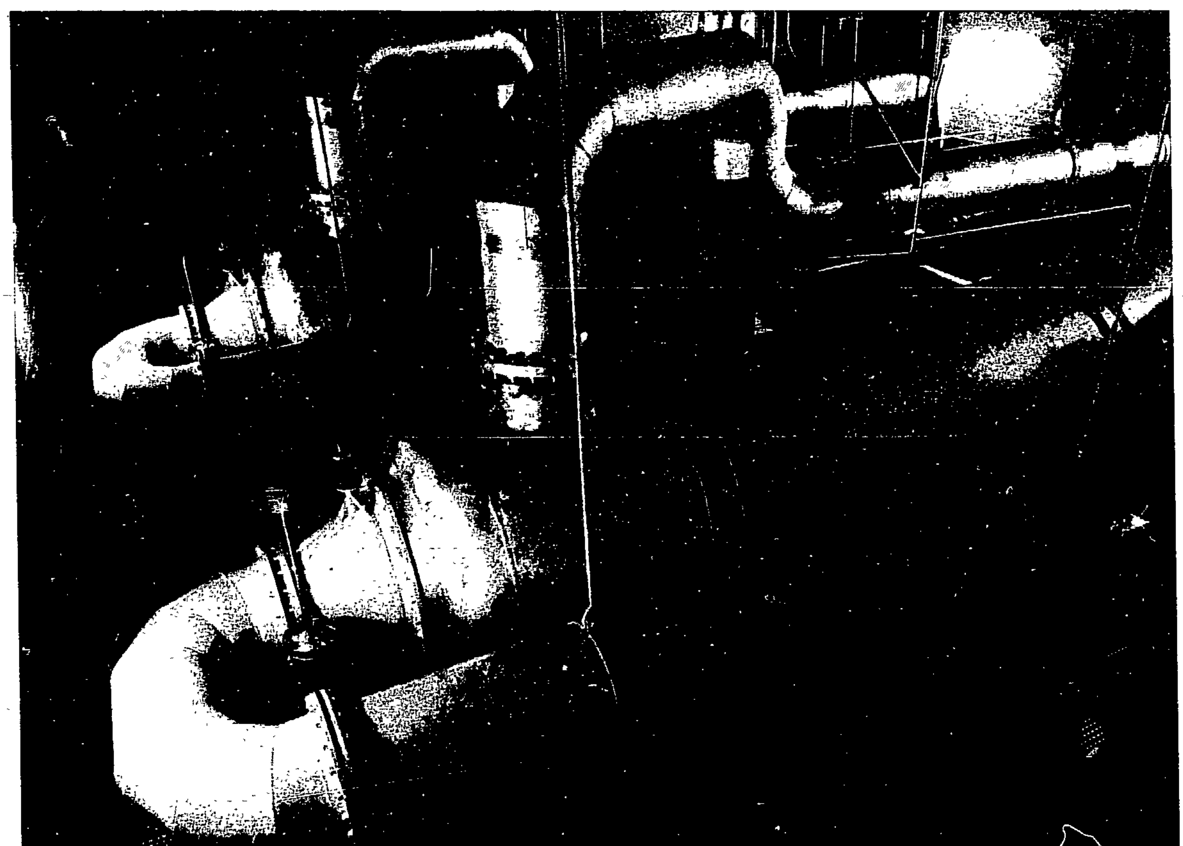
FIG.5

Fig. 6

GENERAL VIEW OF THE NEW EXCHANGERS OF THE BR2 REACTOR

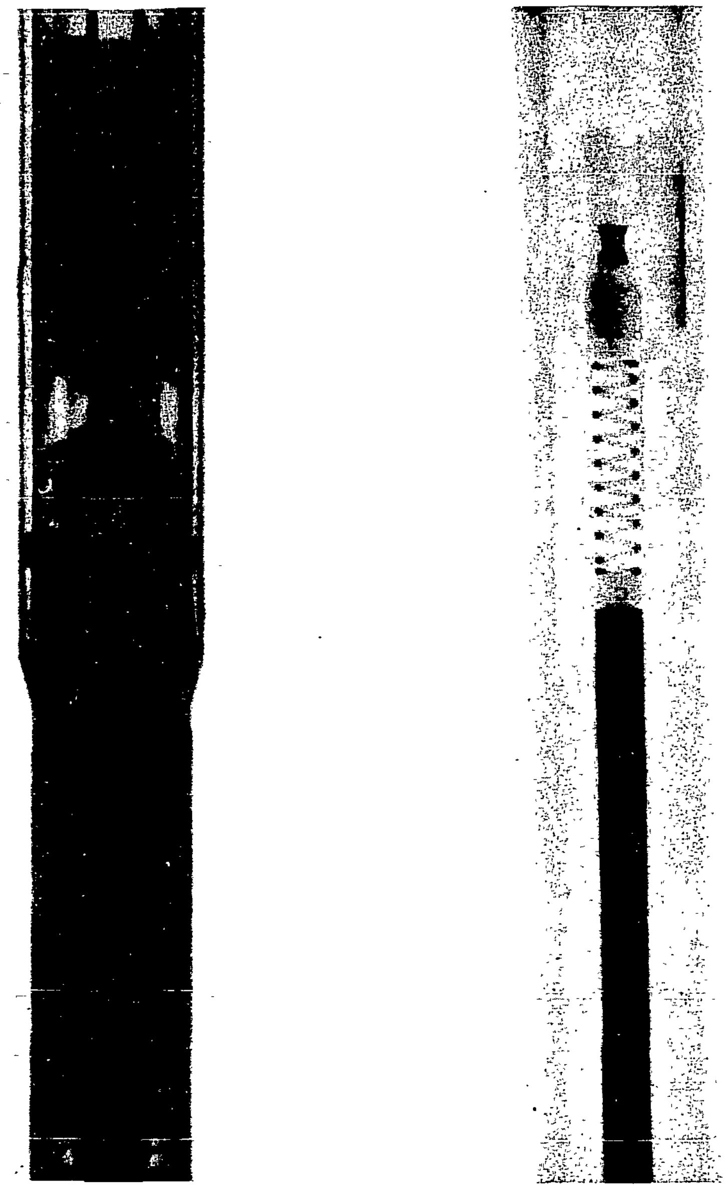


in construction



installed at BR2

FIG. 7



a b

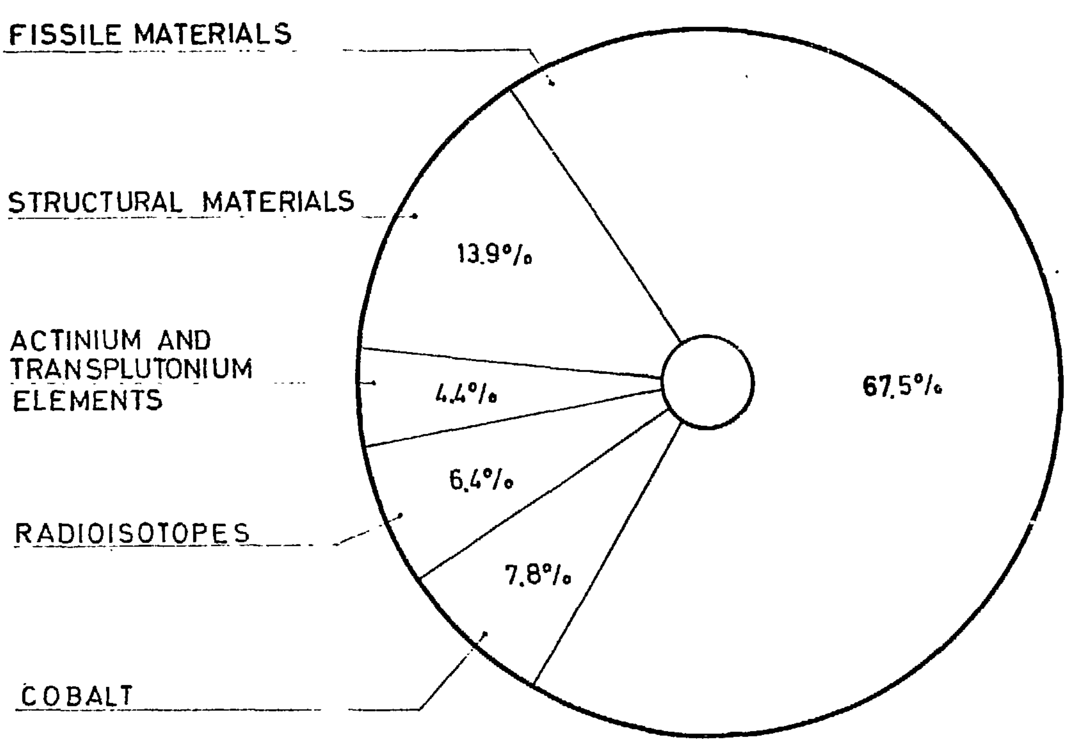
EXAMPLES OF NEUTRON RADIOGRAPHS TAKEN IN THE BR2 POOL INSTALLATION

a. Rig for the irradiation of fissile material with continuous creep measurement.

b. Irradiated UO_2-PuO_2 fuel pin.

FIG. 8

UTILIZATION OF THE REACTOR BR2 IN 1972



BASED ON "FAST IRRADIATION UNITS"
[1 $cm^3 \times 10^{20}$ n cm^{-2} (E>100 keV)]

FIG. 9

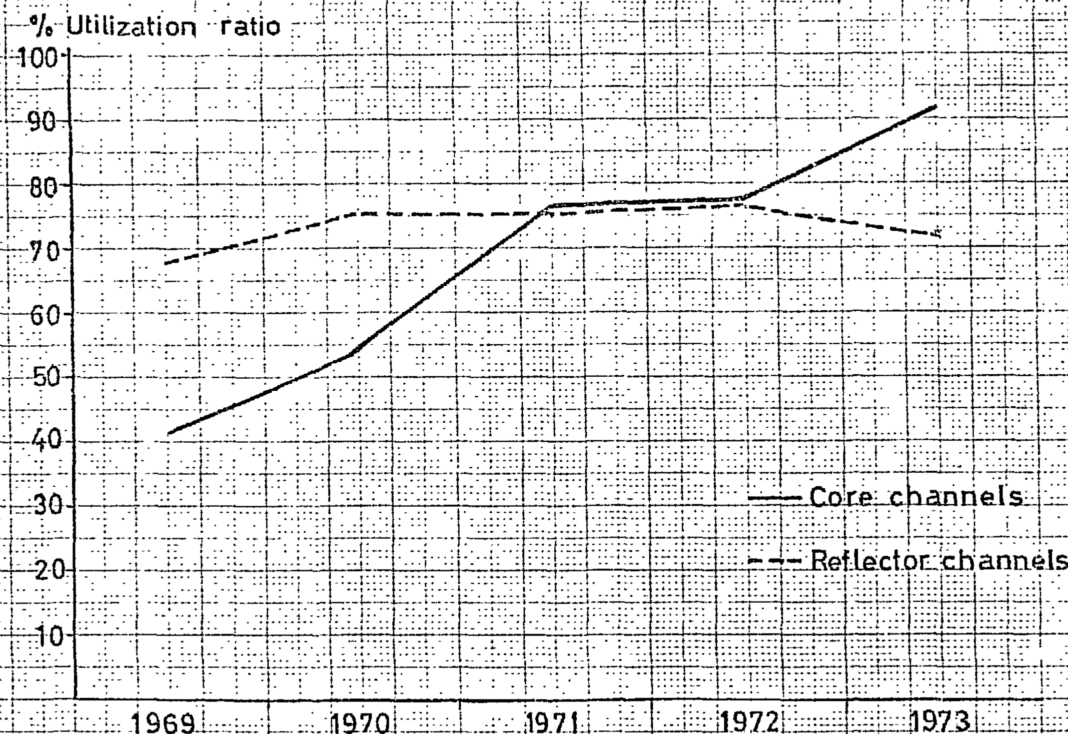
OCCUPATION OF THE REACTOR BR2 1969-1973

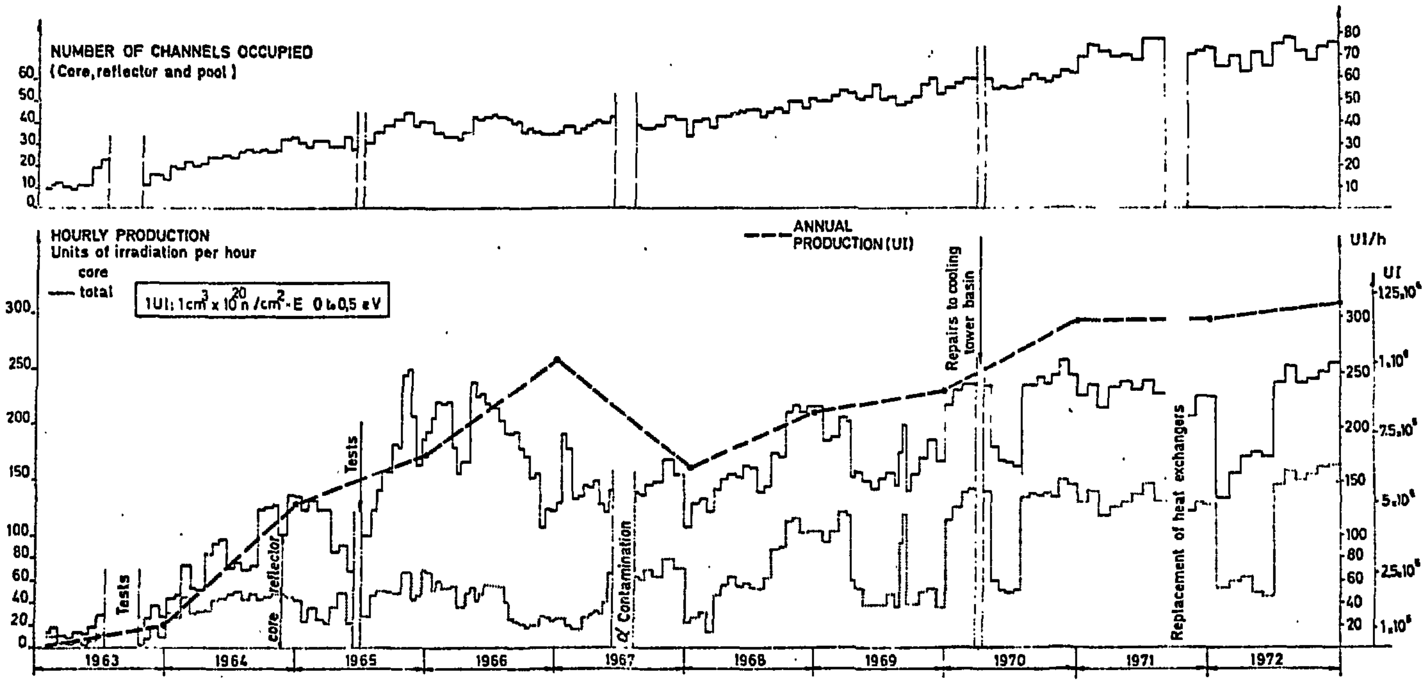
CORE AND REFLECTOR CHANNELS NOT INCLUDING SODIUM LOOPS

(1969-1973 averaged values)

38 Positions available in the core

44 to 38 Positions available in the reflector





EVOLUTION OF THE OCCUPATION OF THE REACTOR BR2 FROM JANUARY 1963 TO JANUARY 1973.

FIG.10

FIG.11

BR 2. AVERAGE NUMBER OF REACTOR CHANNELS OCCUPIED