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BR2 IRRADIATION DEVICES FOR HTGR FUEL

by

P. von der HARDT

1971

Report prepared at the CEN
Centre d'Etude de l'Energie Nucléaire, Mol - Belgium

Association No. 006-60-5 BRAB

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ABSTRACT

A general survey of BR2 fuel irradiations for high-temperature gas-cooled reactors is given.

Moreover, a universal capsule-type irradiation device (rig) is described in detail, together with its fission product sweep loop. Calculation methods, and results, for the thermal characteristics of the rig and for radioactive isotope transport are explained.

KEYWORDS

BR-2
HTGR TYPE REACTORS
COATED FUEL PARTICLES
IRRADIATION
CAPSULES
IN PILE LOOPS
FISSION PRODUCTS
GAS FLOW
HEAT TRANSFER
RADIOISOTOPES
TRANSPORT

TABLE OF CONTENTS. *)

*) Manuscript received on August 5, 1971

TABLES.

FIGURES.

ANNEX.

Summary.

The Belgian Materials Testing Reactor BR2 (MOL, Belgium) has been used as a test irradiation facility for HTGR* fuel and graphite specimens since 1963. (Ref. 17, 18, 19).

The present report deals with coated particles irradiation devices, particularly with the universal facility called Coated Particles Unit Rig (C.P.U.R.) which can be used for a large variety of parameters :

- loose coated particles, compacts or "coupons",
- fuel temperatures between 900°C and 1500°C,
- low enrichment or thorium-containing fuel,
- rig linear fission rating between 50 and 500 W/cm.

The first C.P.U.R. has been irradiated in BR2 from September 1970, to April, 1971, at 1350°C maximum fuel temperature.

The second C.P.U.R. has been under irradiation since the middle of February, 1971, with maximum fuel temperatures about 1200°C.

Rigs nr 03 and 04 are under fabrication, and the layout of two further rigs is being finalized.

Under operation, the main difficulties encountered with high-temperature fuel irradiations in BR2 were :

a. in-pile sections (rigs) :

- primary containment (capsule) leaks,
- thermocouple failures,

b. out-pile equipment (sweep loops) :

- unreliable liquid nitrogen supply for cold purification traps,
- inaccurate measurement of low level moisture and oxygen impurities in the sweep gas,
- problems related with high activity release to the reactor off-gas system when depressurizing at elevated fuel R/B rates,
- difficult control and measurement of very low gas flow rates.

---

* HTGR = High-Temperature Gas-cooled Reactor.
Within the scope of the present report, this term is used in its general meaning and not in correlation to a specific reactor development programme

1. General.

The first nuclear reactor ever built was a gas-cooled graphite-moderated assembly (CP-1, 1942). But, in spite of nearly 30 years of age, this type of reactor is still highly attractive *, and a large amount of R & D work is devoted to its advanced versions.

Irradiation testing programmes are concentrated on graphite (e.g., ref. 29) and coated particle fuels (ref. 2, 7, 13, 15, etc.),.the behaviour of which has been predicted by mathematical models (ref. 1 and 6).

BR2 offers a large variety of irradiation conditions for HTGR fuel tests (see table 1 and fig. 1 hereafter), together with its "traditional" advantages of high neutron flux levels and easy handling of the irradiation devices. The short-coming of BR2 is the limited space available inside its standard irradiation position (the standard rig has a maximum outside diameter of 34 mm), and the high cost for extra fuel consumption and special fuel element fabrication encountered with large diameter devices.

In connection with radiation damage to graphite, efforts have been made to give an accurate definition of fast neutron dose in MTR type reactors (ref. 27 and 26). This problem will not be enlarged upon in this report.

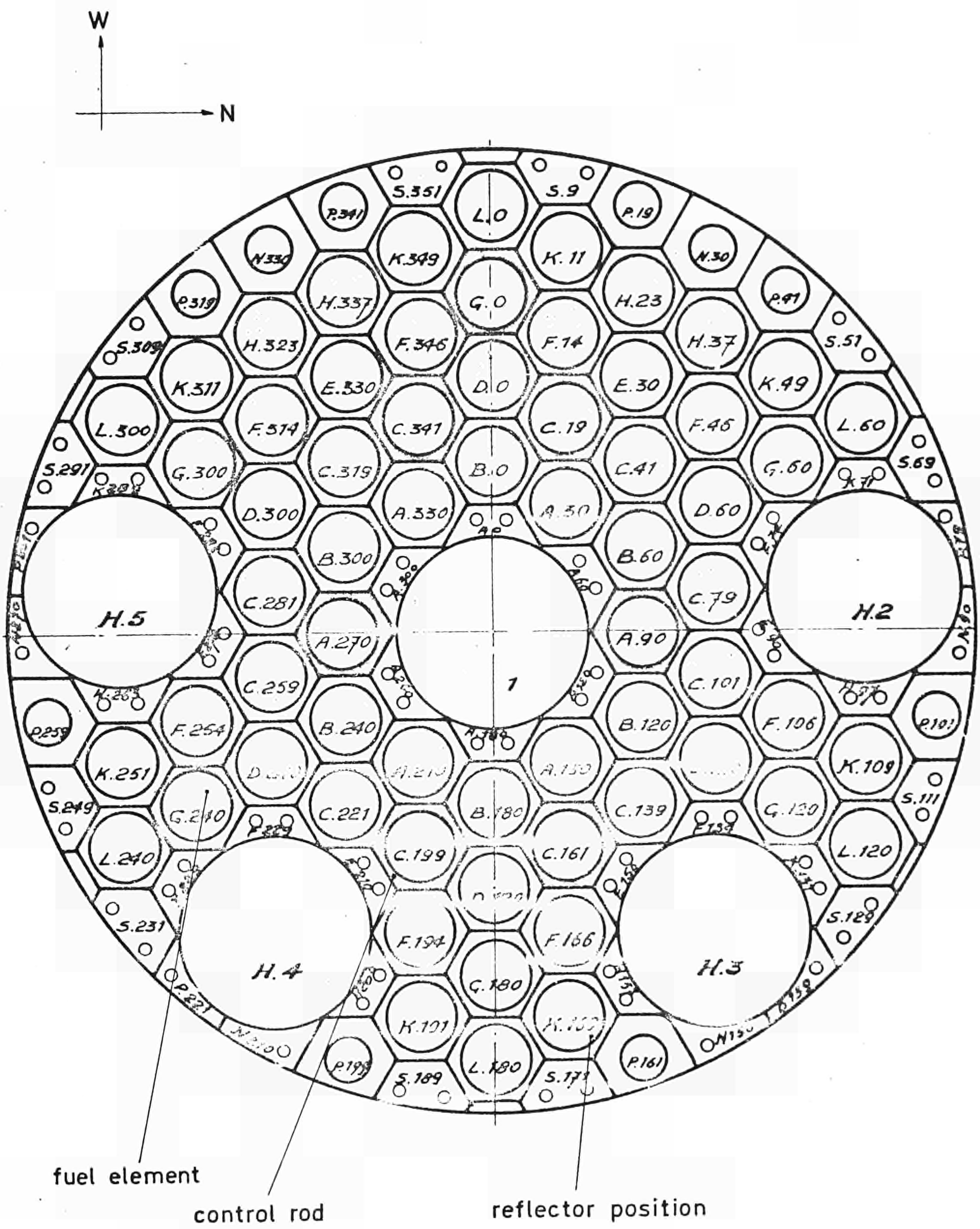
---

* Refer for example to :

- Nukleonik, 11 : N° 1, 44-53, 1968.
- JÜL - 519 - RG (1968).
- EUR 4328 e (1969/70).

FIG. 1

BR 2. REACTOR CORE LOADING DIAGRAM



T A B L E 1.

BR2. Nuclear Data of the Core for Reactor Cycle 05/71 (May/June, 1971). Maximum, Beginning-of-cycle Figures at 70 MW Total Reactor Power (Ref. 36).

Channel	Fuel element Type	Fuel element β %	Ø th.max $10^{14} n/cm^2s$	Ø> 0.1 MeV $10^{14} n/cm^2s$	H. max. W/gr. Al.
A 30 A330	VIn* VIi*	0 0	2.81	5.10	14.8 14.8
A150 A210	VIn* VIn*	0 0	2.87	5.21	15.4 15.4
B 0	VIi*	0	2.81	5.10	14.6
B 60 B300	VIn* VIn*	0 0	2.63	4.77	13.9 13.8
B120 B240	VIn* VIn*	0 0	2.73	5.34	15.4 15.5
B 180	VIi*	0	2.81	5.10	15.1
C 41 C319	VIi* VIi*	17 17	2.92	3.89	12.1 12.0
C 79 C281	VIn* VIn*	0 0	2.33	4.08	11.4 11.6
C101 C259	Vn VIi	21 20	4.01	4.26 4.59	11.7 12.5
C139 C221	Vn Vn	0 0	3.82	4.63	13.5 13.6
D 0	VIn	20	2.68	2.47	8.7
D120 D240	VIn* VIn*	17 18	2.89	4.62 4.56	12.9 12.8
D180	VIn	0	2.68	3.60	10.6
F 14 F346	VIn* VIn*	37 37	2.15	1.48	6.4. 6.4
F 46 F314	VIn* VIn*	40 41	2.55	1.88 1.78	7.6 7.4
F106 F254	VIn* VIn*	42 37	2.68	1.76 2.13	7.8 8.3
F166 F194	Vi VIn*	19 17	2.10 2.20	1.55 2.64	5.6 7.6
G 60 G300	VIi* VIi*	32 32	1.83	1.42	5.1 5.0
G 120 G240	VIi* VIn*	17 26	1.96	2.42 1.94	7.4 6.7
H 23 H337	VIn* VIn*	25 18	1.99 1.75	2.01 2.12	6.0 6.0
H 37 H323	VIn* VIn*	14 25	1.75 1.99	2.34 2.01	6.4 6.1
H 1/0 H 1/1 à 6	IIIs	21	3.50		8.8.

* Cermet elements.

Legend β = fuel element burn-up H = gamma heating

Vn, Vi, VIn, VIi, IIIs = indication of number of fuel plates per element, and type.

The 1971 operating rhythm of BR2 consists in 20 to 21 days of reactor on power and 6 to 8 days of shutdown for replacement of fuel elements and experimental devices, and maintenance.

Assuming 250 full power days of availability per annum and average irradiation conditions, the following performances can be achieved in HTGR fuel irradiation devices during 1 year gross irradiation time :

- Fast neutron dose (> 0,1 MeV)	6 to 8 x 10^{21} nvt.
- Burn-up	8 to 12% fima (fissions per initial metal atoms)

2. Review of HTGR Experiments in BR2.

2.1. Summary.

A summary of all experiments carried out in BR2 for gas-cooled graphite-moderated reactor programmes is given on table 2 hereafter.

Brief descriptions are contained in the following paragraphs.

2.2. Descriptions.

2.2.1. In-Pile Carbon Transfer Loop (IPCTL).

In the scope of the OECD Dragon Reactor Experiment, an in-pile helium loop has been operated in BR2 during the "early" years (Ref. 8, 11 and 13), for the measurement of graphite mass transfer under irradiation with various impurity levels in the gas. The first BR2 core to be operated over an extended period (configuration "5A") was arranged around the D O channel into which the loop in-pile sections were loaded in order to provide the required nuclear data. This core which was eccentric with regard to the H1 channel gave way to the concentric configuration now used (see fig.1) in 1965.

2.2.2. Fuel Ball Rig.

Two 60 mm diameter spherical AVR fuel elements have been irradiated (Ref. 16) in a rig similar to those used in Oak Ridge (Ref. 2) and Studsvik (Ref. 31). The experiment was loaded into a standard reflector channel of BR2 and conducted under essentially thermal neutron flux. Two similar tests had taken place beforehand in one of the pool side facilities. These tests were, in turn, the continuation of early in-pile experiments in the reactor BR1(Ref.5).

2.2.3. 1250°C Coated Particles Rig.

Loose coated particles have been irradiated in two unswept rigs. Each rig contained two sealed specimen carriers in which the particles were mounted between concentric graphite sleeves. (Ref. 16). The upper carrier was fitted with thermocouples. Temperature regulation depended on a helium-nitrogen gas gap between carriers and outer rig tube.

Table 2 - BR2 Irradiation Tests for Graphite and HTR Fuel.
(Reprinted from Ref. 32).

Experiment.	Target	Characteristics	Position.*	Number of Irradiations;	Period.
In-Pile Carbon Transfer Loop	Graphite Cylinders	Graphite corrosion 600-900°C, 10 kg/cm^2 3 g He/sec.	FE	9	63-66
Fuel Ball Rig	AVR Fuel Elements	1000°C surface temp. unswept	refl.	1	66
1250°C Coated Particles Rig	Loose Coated Fuel Particles	1250°C max.fuel temp unswept.	FE	2	66-67
Graphite Irradiation Rigs	Graphite	150 to 350°C, 500 to 700°C,750°C, 900°C, 1200°C.	FE	about 50**	63-69
"A" type Coated Particles Sweep Rigs	Loose Coated Fuel Particles	1250°C max.fuel temp. sweep gas loop.	FE	4	67-70
"B" type Coated Particles Sweep Rigs	Loose Coated Fuel Particles	900 to 1300°C fuel temp. 3 swept capsules per in-pile section.	FE	2 finished. 2 under irradiation.	69-71 71
Coated Particles Unit Rigs CPUR	Loose Particles Compacts, Coupons, etc.	900 to 1500°C fuel t. Various design possibilities.One sweep circuit per rig.	FE	1 finished. 2 under irradiation.	70-71 71
Boiling Water Capsules nr.8,11	Fuel compacts	900 to 1200°C max.fuel temp.Calorimeter device	refl.	1 finished	70-71
Hydraulic Rabbit Irradiations	Fuel compacts	Short-time irradiation	refl.	6	71
Sodium Loop Appendix (MFBS 6)	Loose Particles and Compacts.	Sealed specimen carrier	refl.	1	70-71

* (FE : fuel element position ; fast flux (> 0,1 MeV) up to 6 x 10^{14} n/cm^2 sec
(refl. : reflector position ; fast flux (> 0,1 MeV) up to 1 x 10^{13} n/cm^2 sec.

** Sponsors : CEA, KFA, U.K.A.E.A., etc....

2.2.4. Graphite Irradiation Rigs.

A large variety of irradiation experiments on graphite have been carried out, representing the major load in BR2 during the first five years of operation. The main experimenters were U.K.A.E.A., C.E.A., (Ref. 14) and the THTR programme. The rigs used featured gas mixture regulation, electrical heaters or combinations of both systems.

2.2.5. "A". Type Coated Particles Sweep Rigs.

As a further development of the 1250°C Coated Particles Rig mentioned in paragraph 2.2.3. above, sweep rigs were built in which the upper specimen carrier conducted a low speed flow of purified helium for fission product sweeping. A double sweep loop (able to serve two rigs simultaneously) was built in Mol, and operated by the experimenters up to March, 1971.

A flow diagram of the "A"-type sweep loop is given on fig. 2.

2.2.6. "B"-Type Coated Particles Sweep Rigs.

a. In-Pile Equipment (rig).

The design which is similar to a device used in GETR (Ref.3) was developed in collaboration with KFA Jülich. The outer rig tube (thimble) which is mounted into the lower reactor pressure vessel cover contains three specimen carriers (niobium). Each carrier is loaded with a stack of graphite capsules housing loose coated particles and surrounded by a graphite sleeve. Thermocouples are placed into the capsules and into the sleeve. Each carrier is connected to its own sweep loop for fission product release measurements, whereas the thimble conducts a binary gas mixture for overall temperature regulation. Moreover, the thimble can be moved axially in its irradiation position, by means of a remotely-controlled adjustment device, in order to position the specimen carriers symmetrically to the neutron flux distribution at any time of an irradiation cycle.

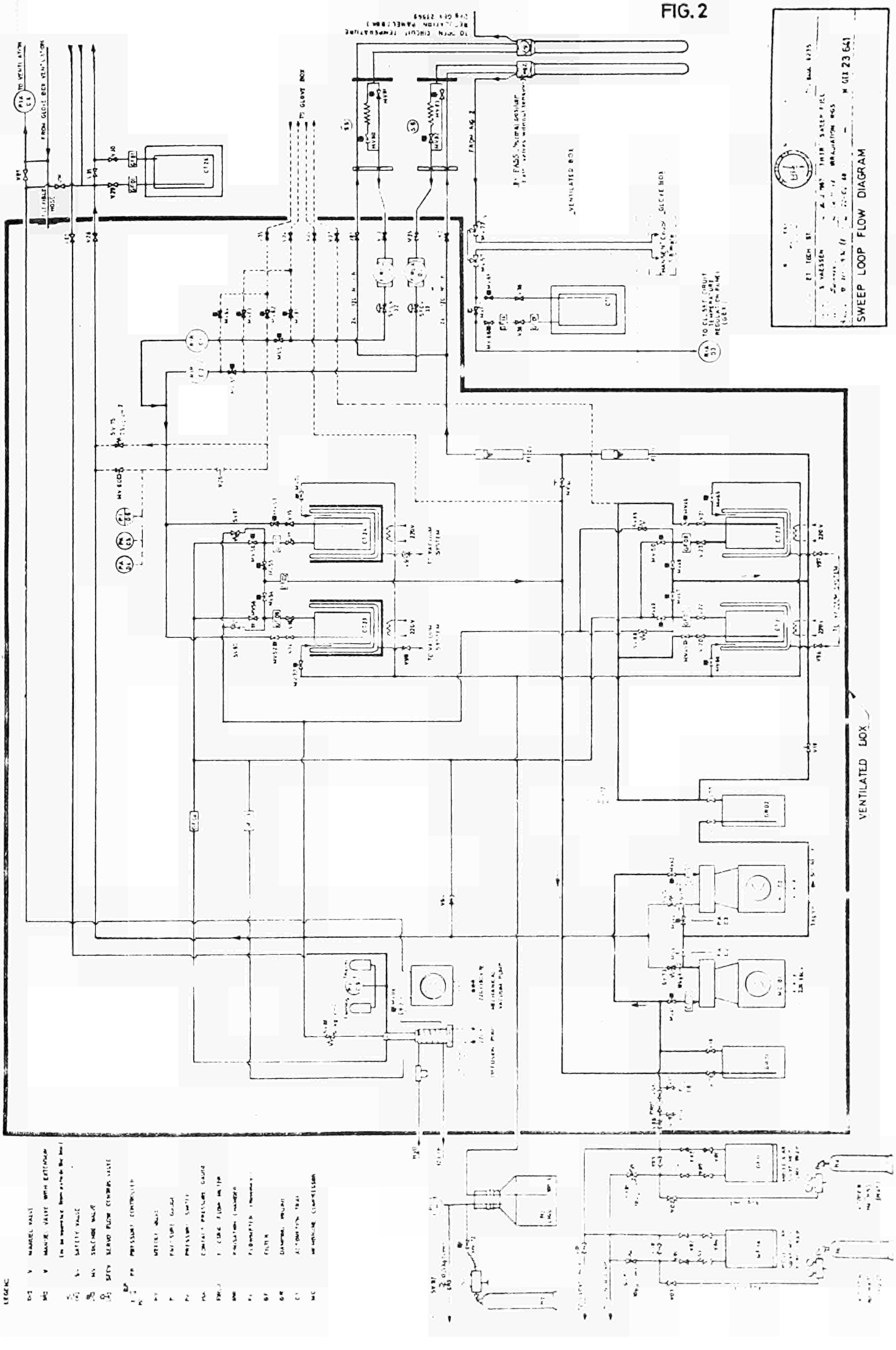
b. Out-Pile Equipment.

Two rigs can be irradiated simultaneously, requiring in total :

- two gas regulating loops,
- six sweep loops,

with their auxiliary equipment, like gas purification, gas sampling station, gamma spectrometry, gross activity scanning, etc...

FIG. 2



Similar installations are being used in the R 2 reactor, Studsvik, Sweden (Ref. 22).

The first rig, irradiated 1969/70, was intended to be operated as an unswept experiment, but leaks from the specimen carriers and elevated fission product release required a gradual transformation of the "inactive " temperature regulation system into an active sweep loop. Numerous safety analyses accompanied the transformation (Ref. 33).

The installation has been completed to the exact specifications, with two gas regulating loops, and six sweep loops, including a digital computer for process control and release rate calculations (Ref. 34).

2.2.7. Coated Particles Unit Rigs (CPUR).

The principles of developing irradiation devices from simple non-instrumented capsules to highly sophisticated assemblies are described in Ref. 21. The reverse direction was chosen for the design of the simple Coated Particles Unit Rig following the much more refined layout of the "B"-Type series described above (Ref. 32).

Design and analysis of the CPUR are described in detail in paragraph 3. of this report.

2.2.8. Boiling Water Capsules nr. 8, 11.

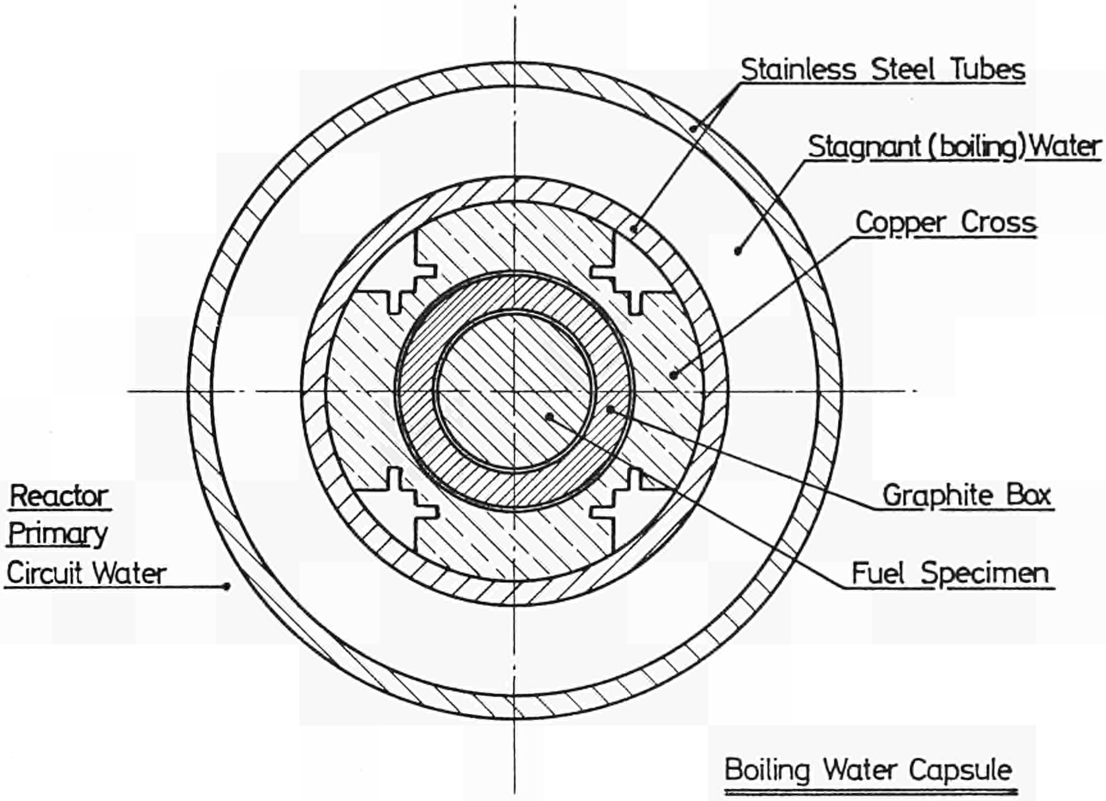
The Boiling Water Capsule (Ref. 11) design has been used in BR2 for the irradiation of fuel pins for the light water cooled and for the fast reactor programmes. A variant has now been employed for coated particle testing (Ref. 28), and a further rig with fuel compacts is under manufacture.

A simplified cross-section of the device is shown on fig. 3 hereafter.

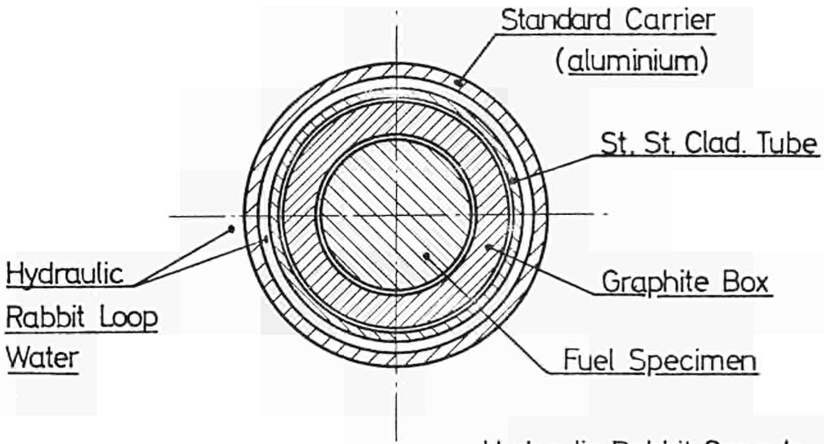
2.2.9. Hydraulic Rabbit Irradiations.

A number of short-time transient tests (Ref. 28) are being carried out in the BR2 hydraulic rabbit facility using standard carriers and a simple design for the capsule (see fig. 3). Irradiation durations vary between 1 and 30 hours.

FIG. 3



Boiling Water Capsule



Hydraulic Rabbit Capsule

"SPECIAL" IRRADIATION DEVICES

Scale 2/1

Schematic Cross Sections

2.2.10. Na Loop Appendix.

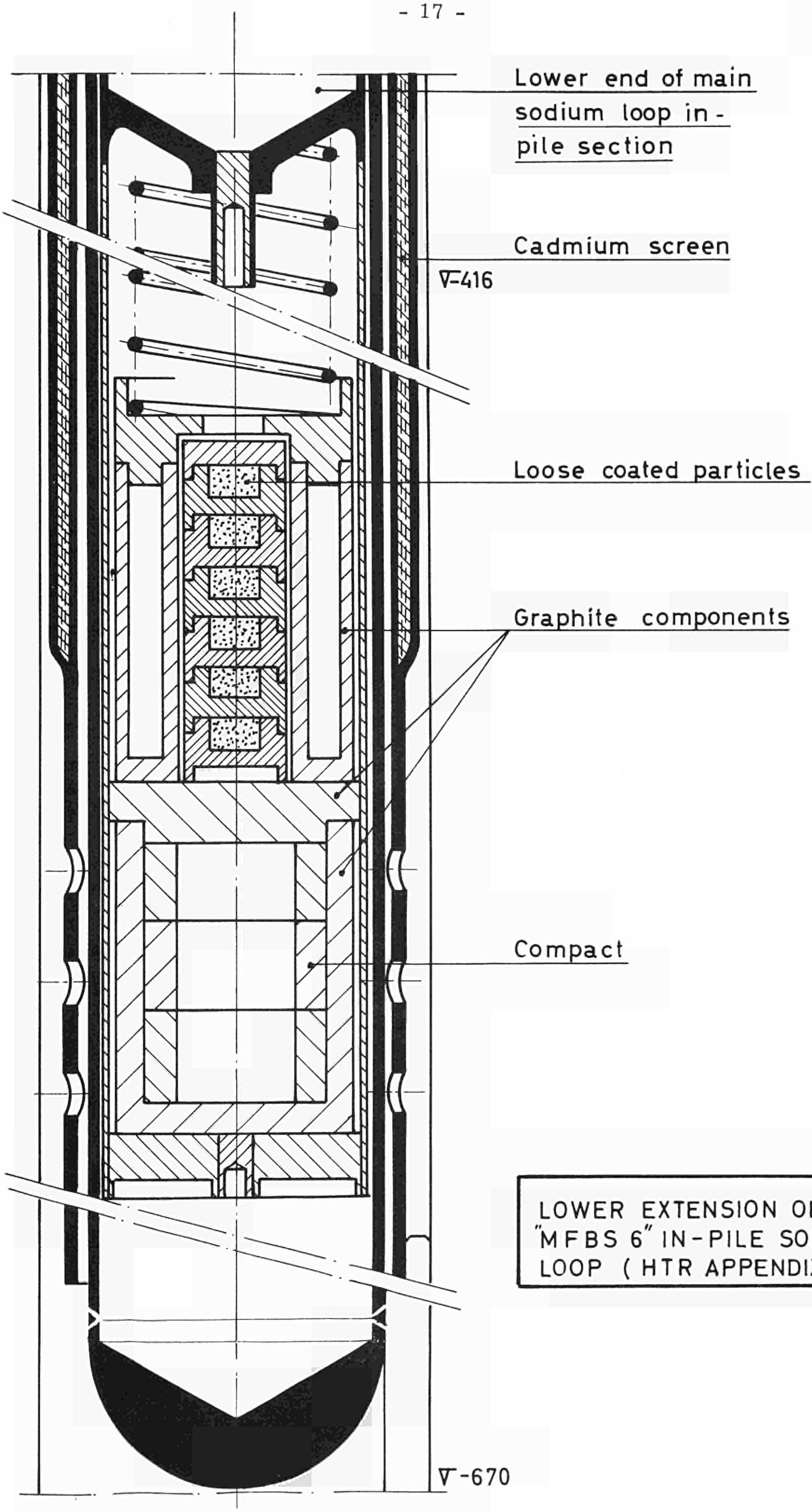
The sixth in-pile section of the BR2 200 kW in-pile sodium loop comprises a lower extension ("appendix") loaded with HTGR fuel (see fig. 4), sitting partly below the main loop Cd screen. The experiment has been under irradiation since January, 1970.

2.2.11. Other Gas Loops.

For completeness it should be remembered that two CO_2 loops have been operated in BR2 from 1966 to 1969 :

- a graphite corrosion loop and
- a fissile heat transfer loop (Ref. 10).

FIG. 4



LOWER EXTENSION OF THE "MFBS 6" IN-PILE SODIUM LOOP (HTR APPENDIX)

3. The Coated Particles Unit Rig (CPUR).

3.1. General.

Design work started early in 1969 for another double-loop sweep facility which was to comply with the following demands :

- simple layout of in-pile and out-pile material, resulting in both maximum reliability and minimum maintenance,
- utilisation of standard BR2 gas mixing panels,
- minimum rig assembling time,
- maximum design flexibility, enabling a large variation of experimental parameters such as target nature, target dimensions, heavy metal loading, fuel enrichment, irradiation temperature, fission rating, etc.

Moreover, the possibility of moving the in-pile section towards "hotter" reactor positions to compensate the target burn-up allows to obtain extended irradiation periods under constant temperature conditions, hence high burn-up and elevated fast neutron doses.

Figure 5 shows examples of possible specimen zone designs, whereas table 3 gives main data of the first six experiments in hand.

3.2. Description.

3.2.1. In-pile Section (rig).

The rig comprises either one extended swept and instrumented compartment, or an upper swept part and a lower sealed appendix. The targets, loose or compacted coated particles, are assembled in graphite sleeves (boxes) which, in turn are contained in a stainless steel carrier. The outer rig tube (thimble) contains the assembly against the primary circuit water, and extends upwards to the rig head in the reactor top cover. The "active zone" components are machined to particularly close tolerances in order to reduce the uncertainty margin of the thermal calculations.
A low flow of helium-neon mixture enters through the outer gas gap (between thimble and carrier tubes), penetrates into the lower end of the specimen carrier and is collected by the gas return tube in the upper end of the specimen carrier.

Hence the gas circulation assures both fission product sweeping and temperature regulation.

The gas leads are carried under secondary containment (flexible stainless steel hose).

Upper and lower specimen carriers are usually fitted with thermal and fast neutron flux activation monitors.

The choice of thermocouples for the upper carrier depends on space restrictions resulting from the specimen zone design, as well as on the experimenter's readiness to accept a central thermocouple well in the specimens.

Usually the outer graphite zone, running at temperatures between 500 and 1000°C, is fitted with 4 to 12 chromel-alumel couples, whereas noble metal couples (1 to 4) are placed in the centre. Current CPUR thermocouple specifications are given in table 4.

The problem of high-temperature thermocouples is briefly reviewed in paragraph 4.1. hereafter.

TABLE 3.

Coated Particles Unit Rig CPUR.
Data of the First Experiments.

Nr.	Type (see fig. 5)	"Hot Plane" Characteristics. Initial fissile material loading (mg U^5/cm)	Irradiation temperature (°C)	Burn-up (% fima*)	Fast neutron dose > 0,1 MeV (10^{21} nvt)	Period of irradiation	Remarks
01	Ia Ib	73	1350	5,4**	2,1**	September 1970 to April 1971	(finished)
02	Ia Ib	50	1300	10 to 12	5	February 1971 to end of 1971	
03	Ic	30	1150	8	6	November 1971 through August 1972	Adjustable version
04	Ib	18	1100	8 to 10	6		Adjustable version
05	Ia Ib	40	1300	12	6		
06	II	18	1100	4	4		

* fima = fissions per initial (heavy) metal atoms.

** calculated after the irradiation.

C.P.U.R.

FIG. 5

Examples of the specimen zone

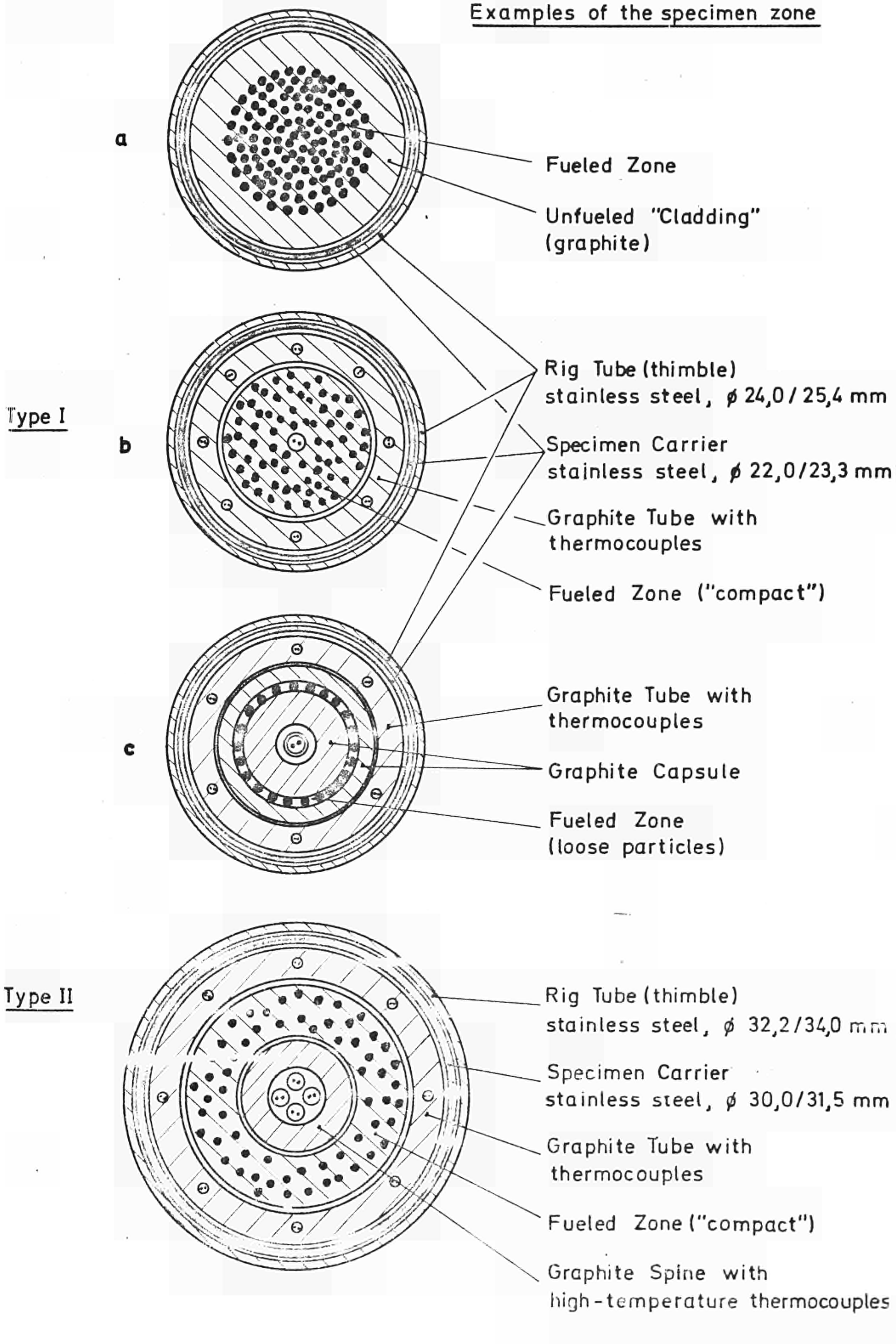


TABLE 4.

Thermocouple Specifications.

High-Temperature Thermocouples			
- Nature of the thermoelectric wires	-	W 3% Re/W 25% Re	
- Sheath material, hot section	-	Mo	
extension	-	AISI 316 stainless steel	
- Transition sleeve, diameter	mm	2,8	
length	mm	35	
- Sheath, O.D.	mm	2,0	throughout
length, hot section	mm	800	
extension	mm	5.700	
- Insulant, hot section	-	BeO	
extension	-	Al_2O_3	
Chromel-Alumel Thermocouples			
- Nature of the thermoelectric wires	-	chromel-alumel (HOSKINS)	
- Sheath, material	-	AISI 304 st. st.	
O.D.	mm	1,0	
length	mm	about 6.000	
- Insulant	-	Al_2O_3.	

Figure 6 is the X-ray photograph of an upper CPUR specimen carrier showing the arrangement of the different components.

3.2.2. Adjusting Device (see figure 7).

The aim of the axial displacement mechanism is to enable, at any time, the positioning of the upper specimen carrier symetrically to the instantaneous reactor neutron flux distribution. This will create nearly constant irradiation and temperature conditions and minimize temperature variations throughout each reactor cycle.

The electric motor, with fitted reduction gear box, is mounted on top of the support plate assembly inside a water-tight case.

It drives the central pinion through a sealed coupling. The rotation of the pinion is transmitted through two intermediate gears to the toothed nuts which create the simultaneous axial displacement of the hollow lead screws and the complete rig.

The adjusting device is operated by means of push-buttons placed into the control panel. The rig height can be controlled in any position on a digital counter indicating the number of rotations accomplished by the central pinion drive shaft.
The counting is based upon an inductive impulse device the polarity of which is inverted together with the motor direction of rotation.

The displacement speed is approximately 48 mm/min, and the total stroke about 200 mm.

A photograph of the adjusting device assembly (without motor casing and service lines) is shown on figure 8.

3.2.3. Out-pile Equipment.
Operation Principle.

The equipment outside the reactor primary vessel comprises the following components :
- reactor pool gas lines and thermocouple cables,
- the pool wall penetration port,
- two gas mixing panels,
- two purification units,
- a gas sampling station contained in a glove box,
- auxiliary systems (liquid nitrogen supply, off-gas lines, inert gas supply).

Figure 10 represents a block diagram of the C.P.U.R. gas loop components, as used for one experiment. Details of the main components are given on figures 11, 12 and 13, whereas the photograph figure 9 shows a view of the glove box distribution (solenoïd valve) panel.

The required helium-neon mixture is adjusted manually, and circulating by means of a membrane compressor, in the gas mixing panel. The purification unit which is built into an own secondary containment box comprises a molecular sieve gas drier and a low temperature active charcoal adsorber.

FIG. 6

C.P.U.R.

Radiograph of an Upper Specimen Carrier

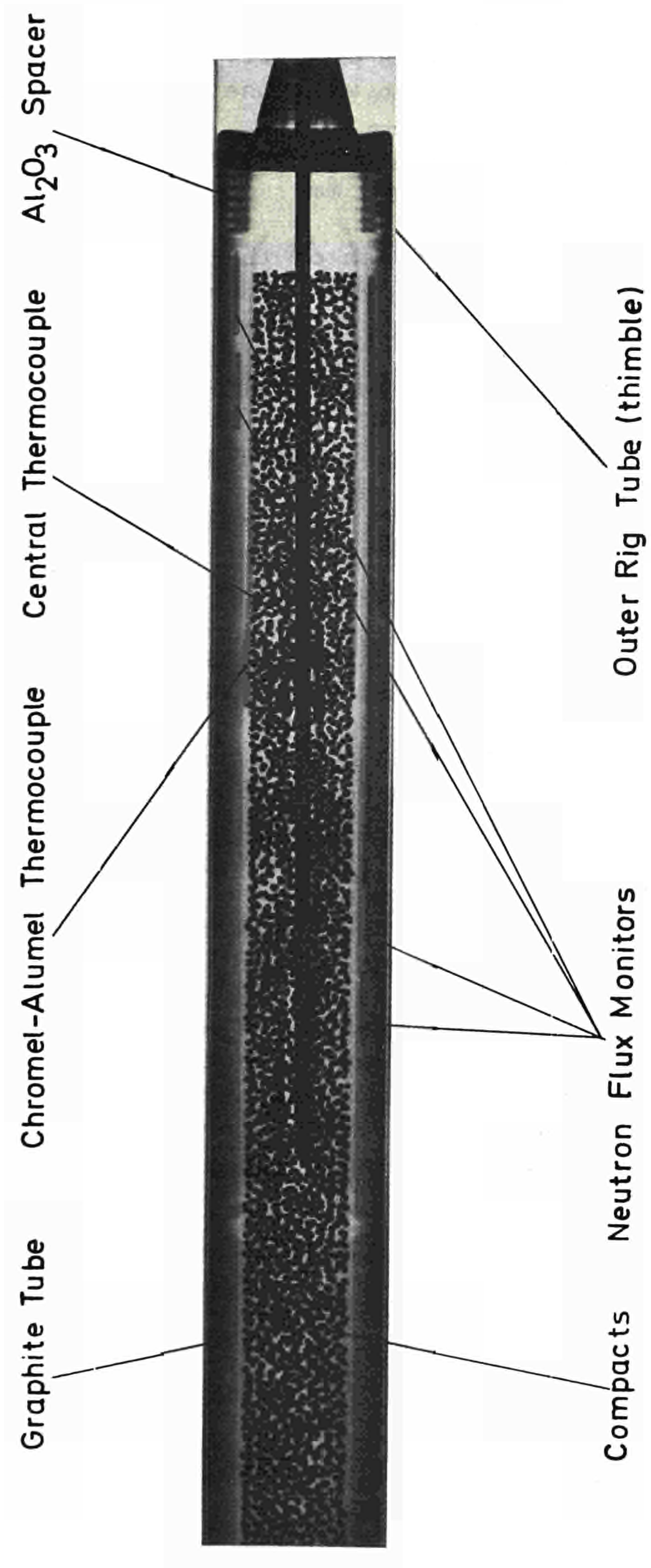
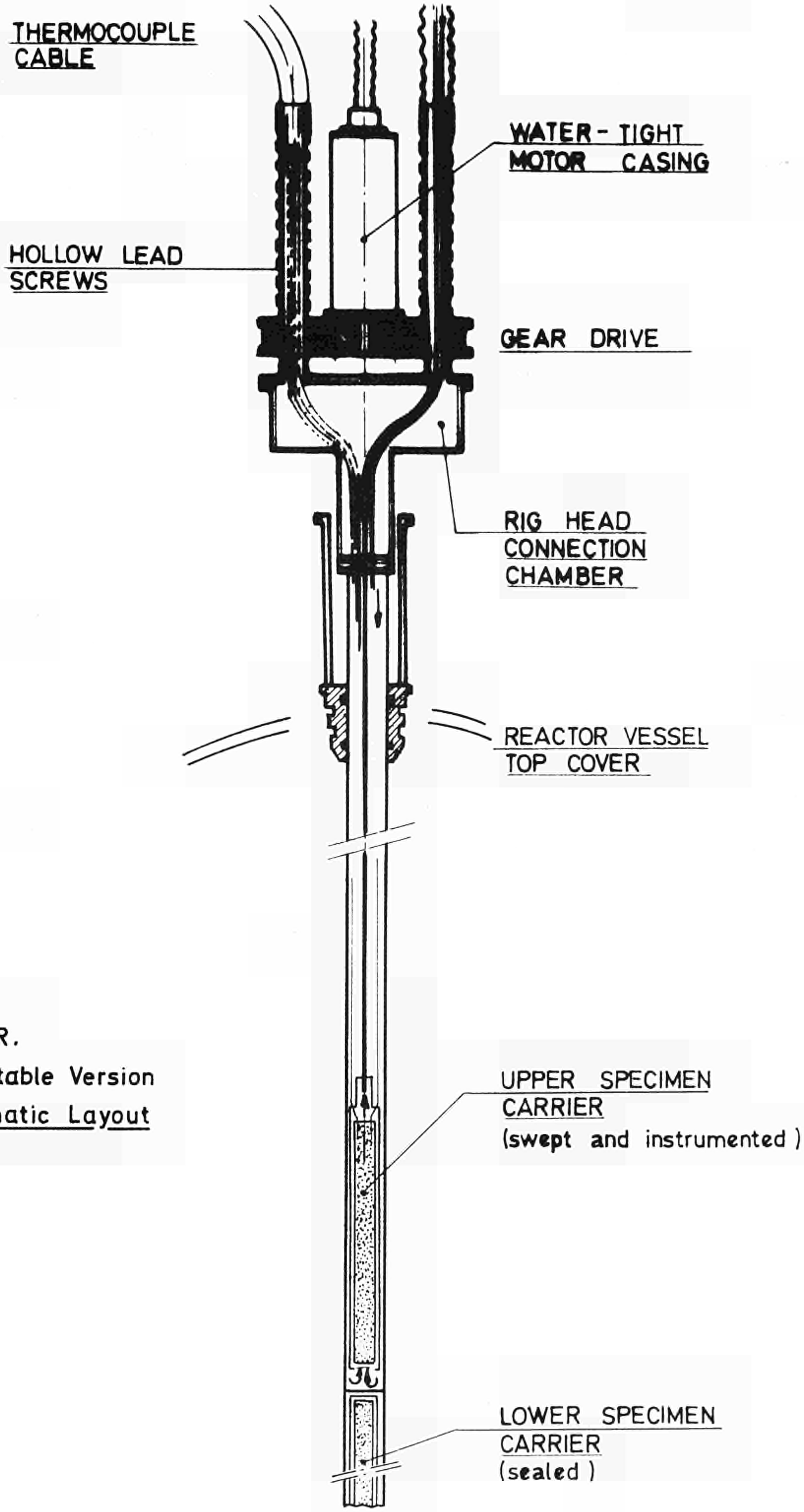


FIG. 7



C.P.U.R.
Adjustable Version
Schematic Layout

The "cold trap" is mounted inside a vacuum-isolated vessel with automatic liquid nitrogen supply and acts as a filter for both chemical impurities (O_2, N_2, H_2O, CO,...) and fission gases.

The gas sampling station inside the system glove box serves the two sweep loops. It contains the overall sweep gas gamma activity monitoring, and tap connections for loop gas sampling in calibrated volumes which are scanned off-line on a multi-channel analyzer.

Each component is fitted with its own vacuum group for cleaning and regeneration purposes.

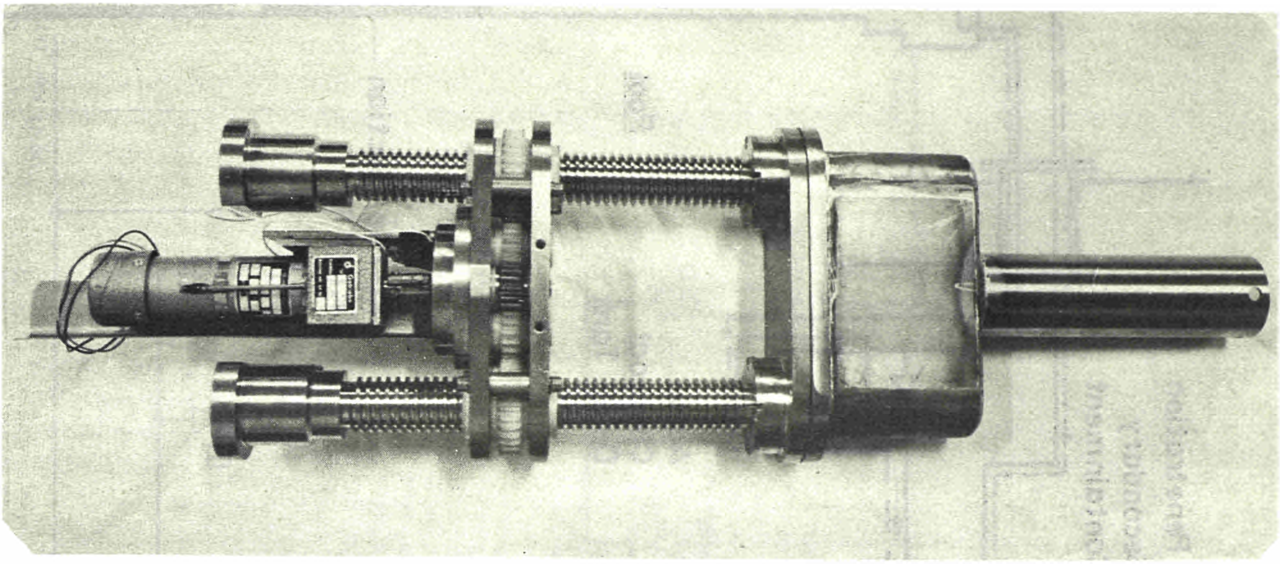
The loop instrumentation comprises the necessary pressure gauges, flow meters, etc... as well as moisture and oxygen meters, neon percentage indicators, and recorders for :
- rig thermocouples,
- sweep gas gamma activity,
- active charcoal absorber temperature.

Molecular sieve drier (T5) and low temperature active charcoal adsorber (T6) are fitted with incorporated controlled electrical heaters for regeneration during the reactor shut-down periods. Regeneration is achieved by first rinsing with nitrogen and then evacuating, both under 100 ... 200°C.

<u>Legend to figures 11, 12 and 13, operation details.</u>

The symbols used are explained in table 5 hereafter. The main purification loop consists of the compressor with its two damping vessels T1 and T2 (G.M.P.), the drier T5 and the low-temperature trap T6 (P.U.).
The gas flows through the following components : GF2, V8, V7, T1, V6, V11, NV42, V43, GF6, T5, GF7, V47, GF8, T6, GF9, V49, V12, NV29, FI3, V18, T2, V17, V9, GF1 and back into the compressor. The main loops contains about 28 litres of purified helium (or helium-neon) under 3 ata (28,4 p.s.i.g.) and circulating at about 1500 Ncm^3/min.
The pressure drop caused by NV42 is used to drive the rig loop via FI7, V40, rig, MV118, active charcoal trap, MV117, RIRA 01, MV112, V41 and back into the main loop.
The flow rate in this branch is 100 Ncm^3/min.
The glove box comprises the sampling station for two sweep loops, confined by solenoïd valves MV115, MV116, MV121 and MV122. It consists of a number of taps fitted with self-sealing couplings to which calibrated sampling volumes (VV) can be connected.

FIG. 8



C.P.U.R. Adjusting Device

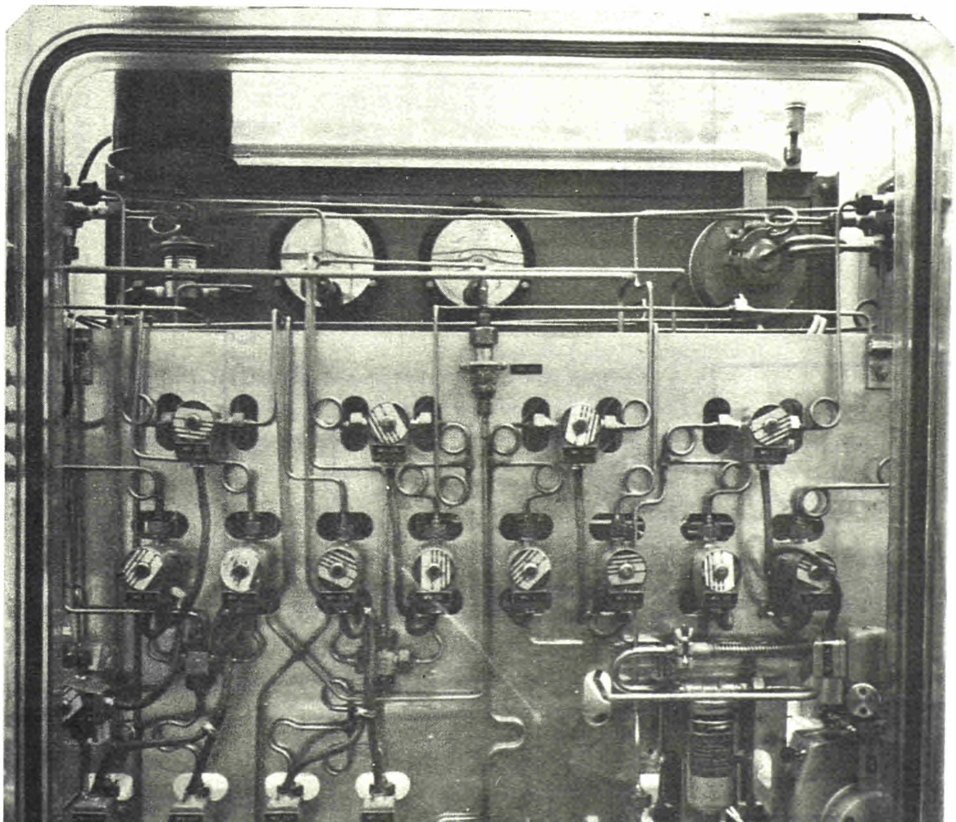
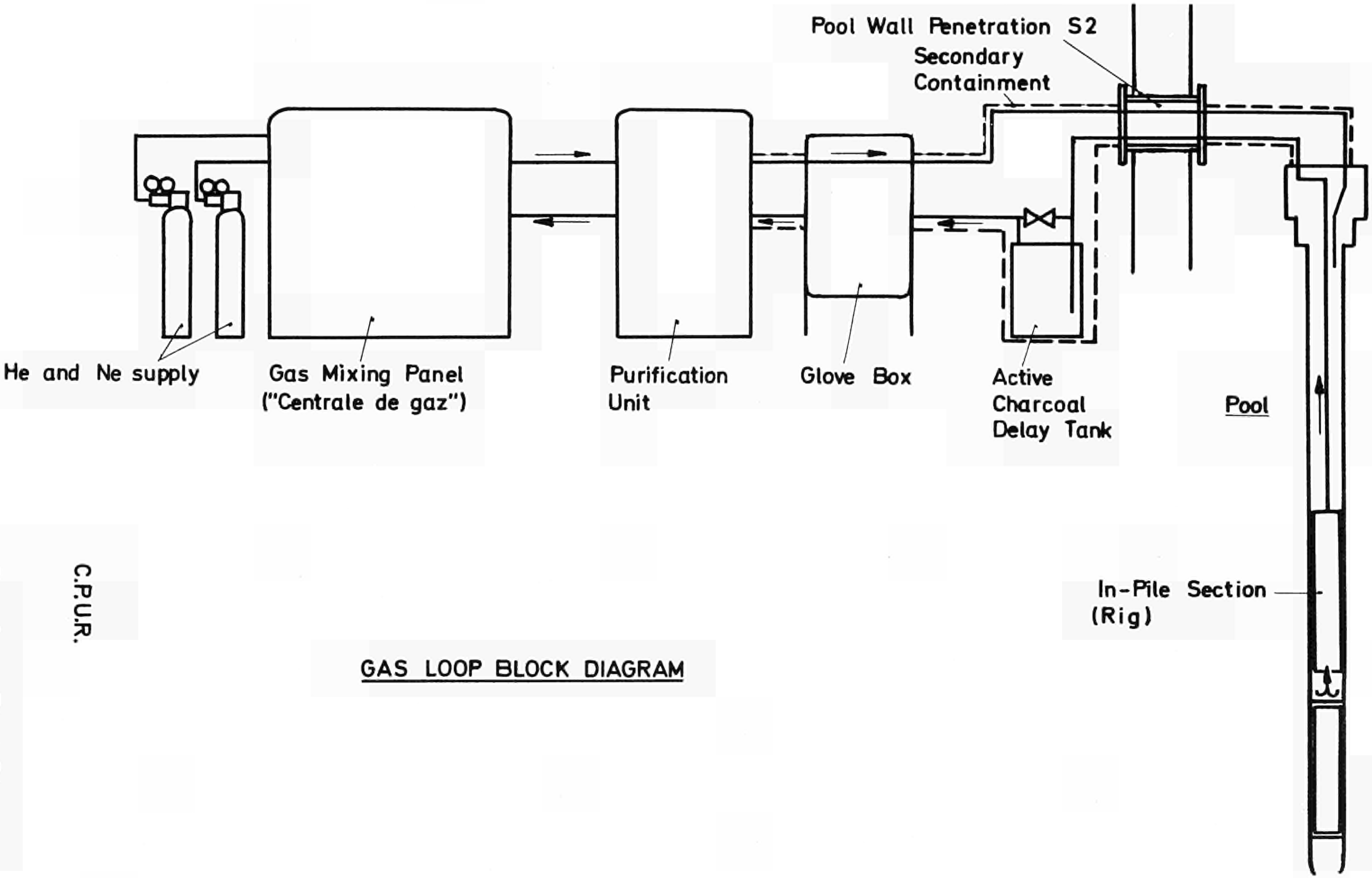


FIG. 9

C.P.U.R. Glove Box
Magnetic Valve Panel



GAS LOOP BLOCK DIAGRAM

FIG. 10

C.P.U.R.

Gas Loop Block Diagram

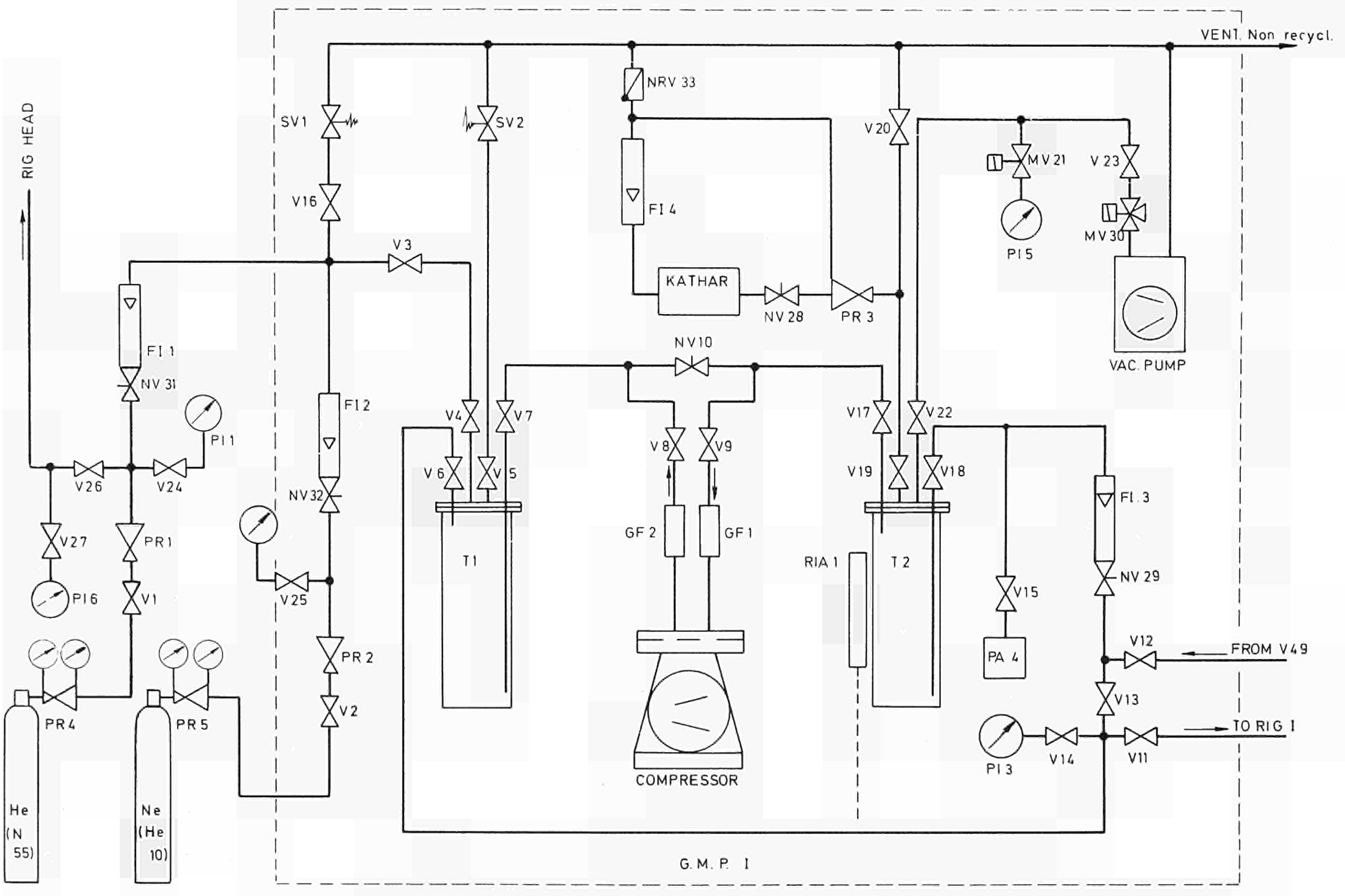


FIG. 11 GAS MIXING PANEL. FLOW DIAGRAM.

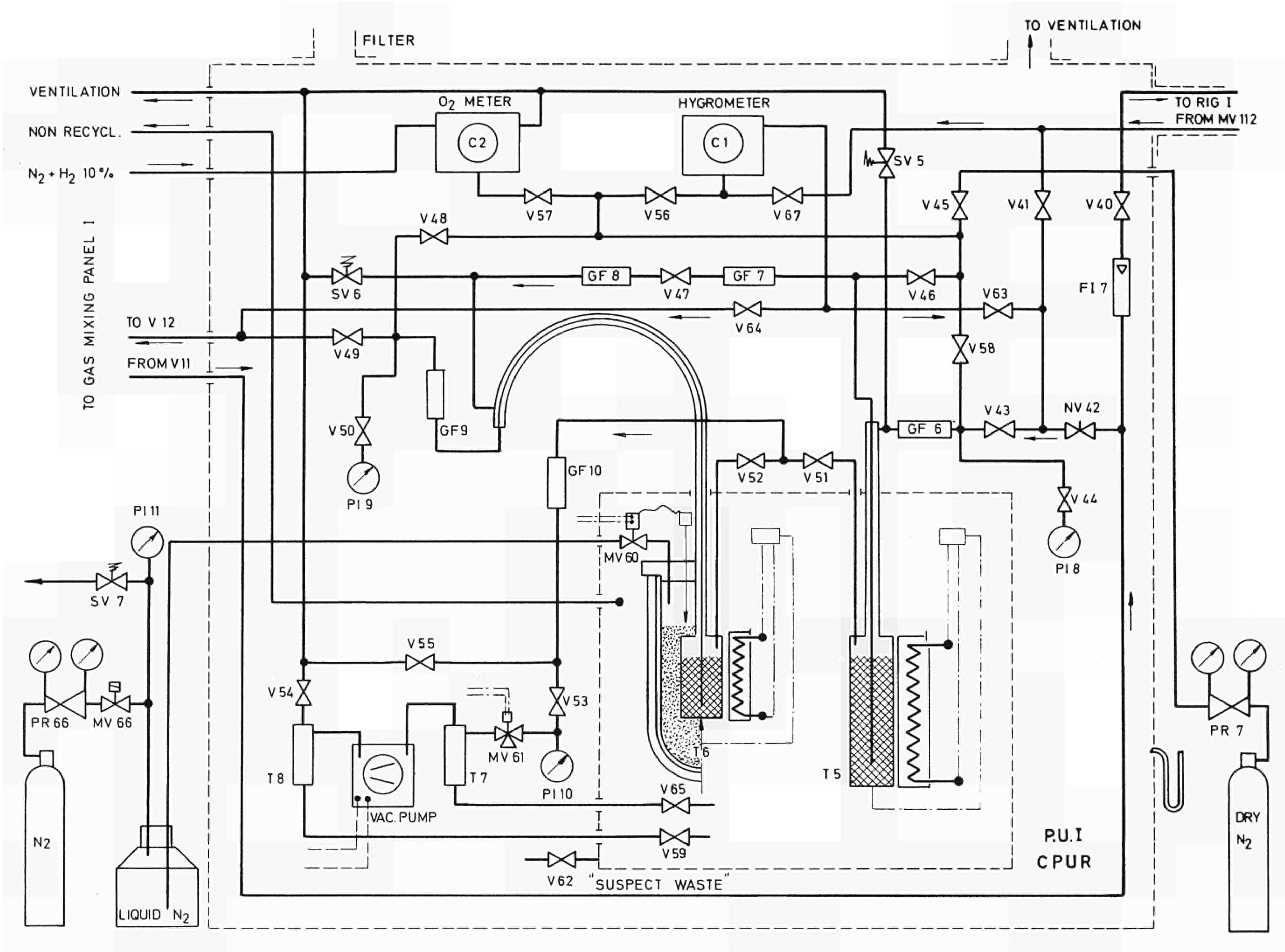


FIG. 12 PURIFICATION UNIT. FLOW DIAGRAM.

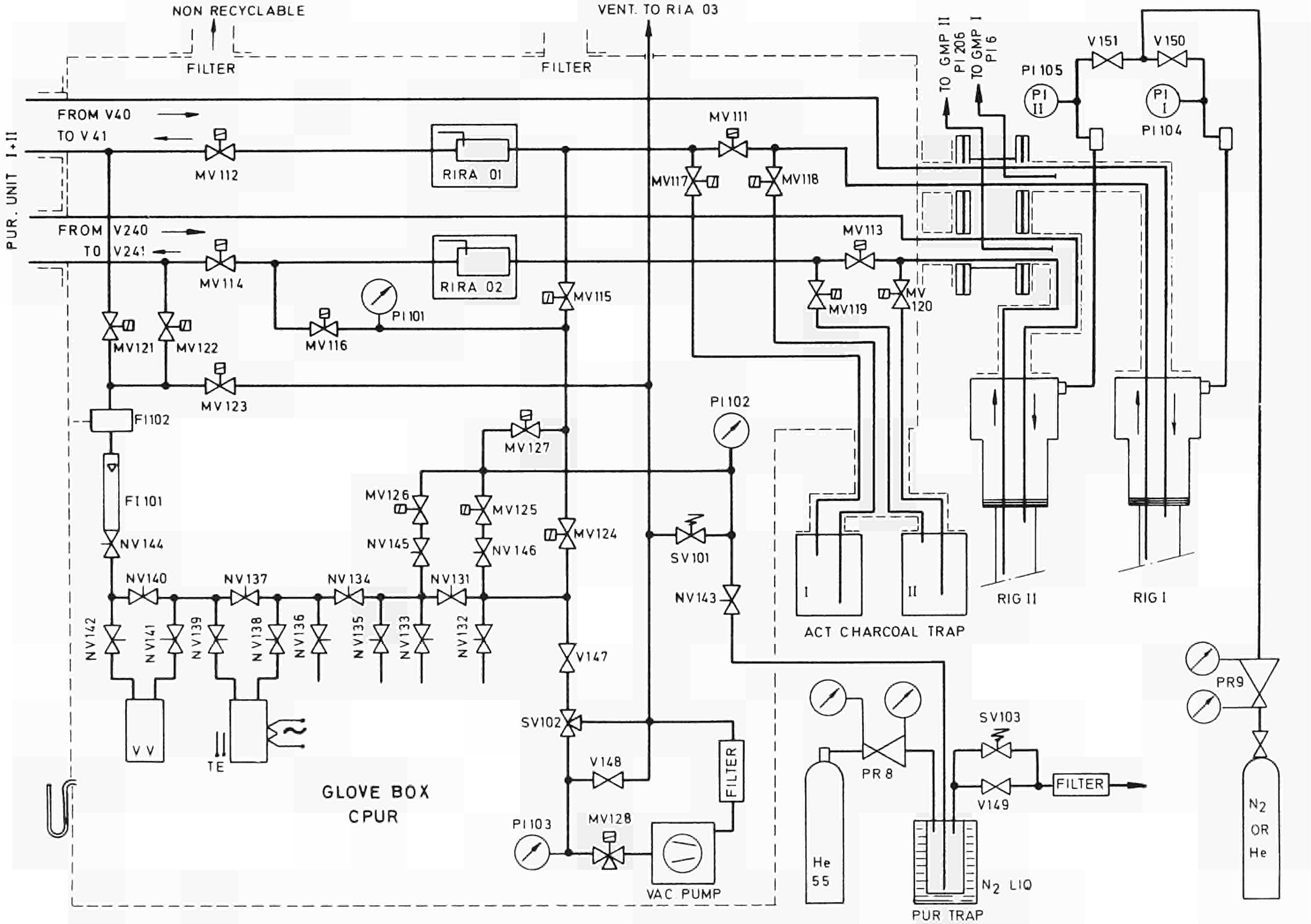


FIG. 13 GLOVE BOX. FLOW DIAGRAM.

Provisions are made for total adsorption using liquid nitrogen, and for controlled heater circuits for cleaning and regenerating purposes. The returning sweep gas can be directed through a 5 litre room temperature active charcoal trap, by means of a set of solenoïd valves (MV111, MV117, MV118), before reaching the remainder of the loop. A considerable desactivation of the gas is then achieved (see paragraph 3.4.2. hereafter) avoiding permanent circulation of activity in the loop. For low activity release from the rig and for gas sampling, the trap is by-passed, reducing the fission product transport time to about 7 minutes. An inlet of purified helium is provided into the glove box sampling station (NV143) for scavenging and transfer purposes.

There is a special electrical control panel for the glove box, carrying the controls for solenoïd valves, vacuum pump, electrical flow meter FI102 and heater circuits, as well as high voltage supplies, ratemeters and recorders for activity monitoring chains RIRA 01, RIRA 02 and RIA 03 (without recorder). An electrical interlock system of the solenoïd supply lines prevents faulty operation of the glove box, such as draining the main loop via MV 123 or pressurizing it via MV 127.

In the purification unit (fig. 12), both traps T5 and T6 are fitted with tube-in-tube heat exchangers which act as recuperators during the 150 to 250°C nitrogen regeneration. Moreover, the T6 exchanger serves as a recuperator during normal low temperature operation bringing about both limited ice formation on the tube and liquid nitrogen economy.

From a number of methods tested for the automatic liquid nitrogen supply to cold traps, the thermocouple control was found to be the most reliable : Several iron-constantan couples (st. st. sheathed) are placed onto the external surface of the trap which dips into the liquid nitrogen filled vacuum-isolated flask. They are connected to a galvanometer or to a potentiometric recorder, from which on-off contacts control the opening of supply valve (e.g. MV60 on fig. 12). The additional advantage of this method, as compared to the other principles, consists in the direct information on trap temperatures. In the case of the C.P.U.R. loops, recording of the trap temperatures is included for supervision of the liquid nitrogen level control and detection of any failure in the system.

The two impurity analyzers in the purification unit (C1, C2) are of the electrolytic type (hygrometer) and of the photometric type (O_2 meter).
The hygrometer operates continuously and in a closed circuit, whereas the O_2 meter is used occasionally and by "sacrificing" loop gas into the reactor off-gas system.
Operation of the O_2 meter requires the use of a reducing gas containing 10% H_2, as well as of a calibrated reference gas.

T A B L E 5.

Explanation of Symbols of Figures 11, 12 and 13.

V	Manual valve
NVR	Non-return valve
NV	Needle valve
MV	Solenoïd (magnetic) valve
SV	Relief (safety) valve
PR	Pressure reducer or pressure regulator
GF	Filter
T	Tank, vessel, trap
PI	Pressure (or vacuum) gauge
PA	Pressure switch
FI	Flow meter
KATHAR	Katharometer (Ne percentage indicator)
TE	Thermocouple
VV	Calibrated volume
RIA	Gamma dosimeter with alarm display
RIRA	Gamma cell with recorder and alarm display
G.M.P.	Gas Mixing Panel
P.U.	Purification Unit.

The main difficulties observed in the beginning of the sweep loop operation were (see paragraph 4.2.) :

- unreliable liquid nitrogen supply for the "cold traps"
- inaccurate measurements of low level moisture and oxygen impurities in the sweep gas,
- problems with high activity release to the reactor off-gas system when depressurizing at elevated fission gas release rates,
- oil vapour condensation from vacuum pumps in the off-gas lines.

3.3. Thermal Calculations.

3.3.1. Scope of the Problem.

The analysis is focussed on calculations of the central fuel zone and the graphite tube temperatures as functions of :

- the equivalent fissile material loading (EFM), variable in time and space,
- neutron flux and gamma heating, variable in time and space,
- the internal rig heat transfer conditions, variable with the helium-neon mixture composition and variable in time and space if dimensional changes of the graphite components by fast neutrons are to be taken into account (Ref. 29).

N.B.

a) the equivalent fissile material loading EFM is expressed in (mg U 235/cm). Fissile nuclei bred from U 238 and/or Th 232 are taken into account by definition

$$EFM = \frac{N^5 \cdot \sigma f^5 + N^3 \cdot \sigma f^3 + N^9 \cdot \sigma f^9}{\sigma f^5}$$

Where

N = loading density (mg/cm)

σf = microscopic fission cross section (cm^2)

Indices : 5 for U 235 3 for U 233

9 for Pu 239

b) "variable in time" means changing under irradiation, "variable in space" means changing according to the axial position in the rig.

The requirements from the experimenter's point of view are generally :
- maximum possible initial fissile material loading. This requirement originates from the statistical character of coated particle irradiation testing, it implies that at least 10.000 particles should be irradiated under identical conditions,
- constant irradiation temperature in time and space,
- optimum choice of the irradiation sequence, i.e. : select throughout the irradiation the "hottest" channels possible in order to achieve specified burn-up and fast neutron dose as soon as possible, but without ever exceeding the maximum admissible temperature.

3.3.2. Principles of Solution.

Three degrees of accuracy can be defined for calculations predicting the thermal behaviour of a C.P.U.R. :

1. The constant loading, constant flux approach.

The rig is fitted with a constant initial fissile material loading density. Calculations are limited to the "hot plane" (plane of maximum neutron flux and gamma heating) where constant flux conditions are assumed throughout each cycle.

2. The variable loading, constant flux approach.

A constant axial flux distribution is assumed for each cycle, and the initial fissile material loading density is selected to give the best possible approximation to constant axial fuel temperature distribution.

3. The variable loading, variable flux solution.

The axial distribution of neutron fluxes and gamma heating is considered variable for each cycle, and the initial fissile material loading density is selected to give the best possible approximation to a constant fuel temperature profile.

3.3.3. Calculation Examples.

3.3.3.1. Constant Loading and Flux.

A computer programme has been established (Ref. 35) calculating rig component temperatures as functions of :

- geometry and materials involved,
- gas atmosphere inside the rig,
- fission rating from the fueled zone,
- gamma heating,

taking into account :

- variation of gas gap thermal conductivity with temperature,
- thermal expansion of the components,
- radiative heat transfer.

Results of such calculations for a Ic-type C.P.U.R. are plotted on diagram figure 14. Dotted and full lines indicate temperatures for minimum and maximum assembly gas gaps, respectively, as resulting from component machining tolerances. The gamma heating figure of 6 W/g corresponds to a low flux position (see table 1 of this report) as a starting point of the irradiation campaign. From the diagram, the required linear fission rating is selected to give the specified irradiation temperatures. On the other hand, the unperturbed thermal neutron flux for the starting position is known (1,7 to 2,2 x 10^{14} n/cm^2 s, see table 1), and the corresponding perturbed (= effective) thermal neutron flux in the fuel zone can be calculated by means of the rig "self-shielding coefficient".
This coefficient, which has been calculated (Ref. 37) using an IBM 360 version of the GMS 1 code, is defined as :

$$f = \frac{\phi \text{ averaged over fuel zone cross section}}{\phi \text{ unperturbed, on centre-line of the idealized configuration}}$$

where

$$\phi = \left[\int_0^{0,5 \text{ eV}} n(E)\,dE \right] \cdot V_{2200 \text{ m/sec}}$$

As a good approxiate average, f is taken = 0,7.

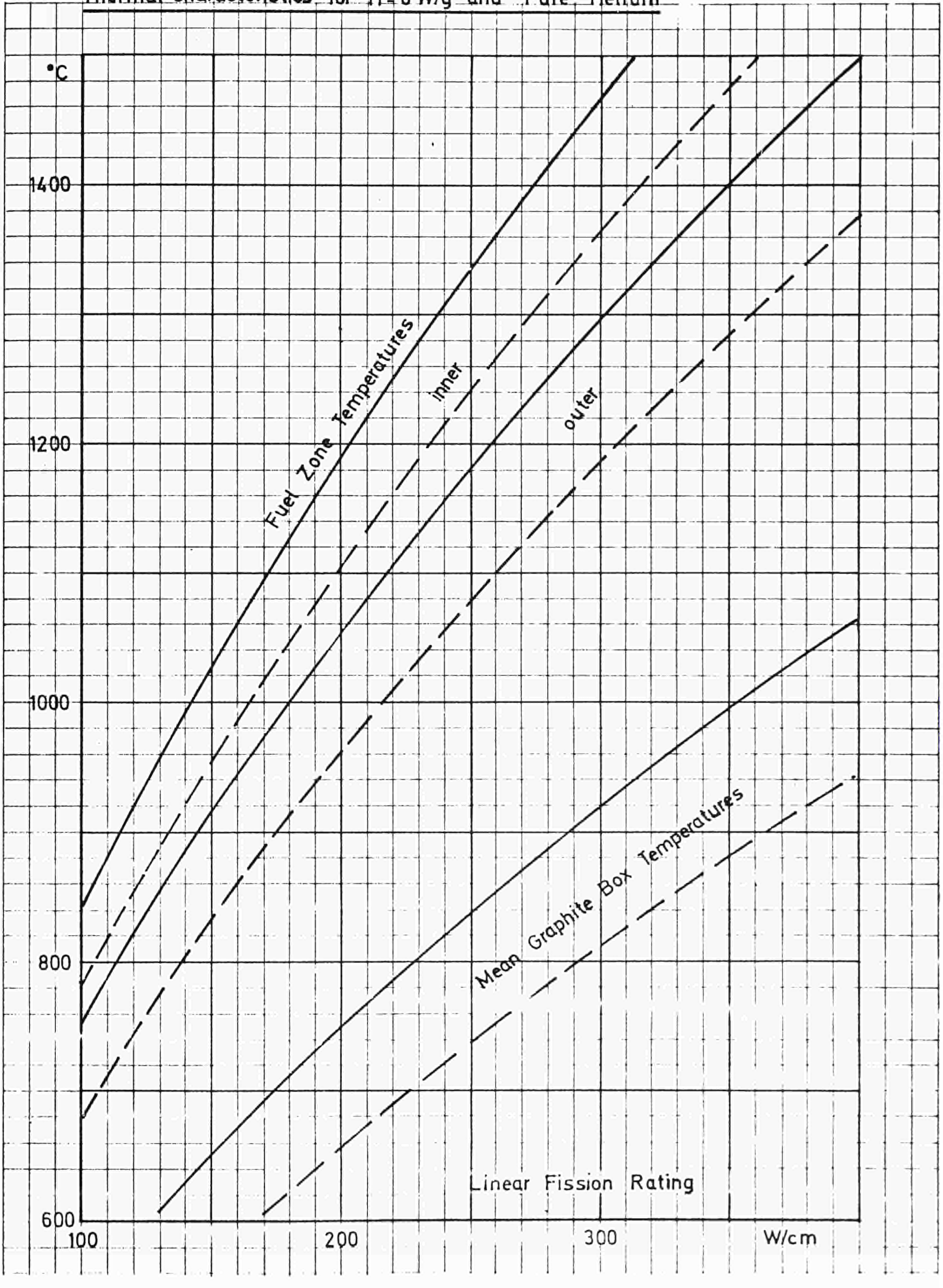
From effective thermal neutron flux and required linear fission rating, the necessary initial fissile material loading is then determined.

C.P.U.R.

FIG. 14

Typical "Hot Plane" Diagram

Thermal Characteristics for H = 6 W/g and Pure Helium



In an analoguous manner, one calculates the "end-of-irradiation" condition, using :

- the final EFM, as figured out from burn-up and breeding,
- an irradiation position with about 14 W/g gamma heating and 3,5 to 4 x 10^{14} cm^2 s unperturbed thermal neutron flux,
- helium-neon gas filling with up to 90% Ne proportion.

An example of beginning and end-of-cycle temperatures is given on figure 15.

Figure 16 represents the relative EFM for different initial quantities of Th232, as calculated (Ref. 4) for an assumed constant effective thermal neutron flux of 2,4 x 10^{14} n/cm^2 s. The considerable influence from bred nuclei for long-term irradiations is clearly to be seen.

3.3.3.2. Variable Loading, Constant Flux.

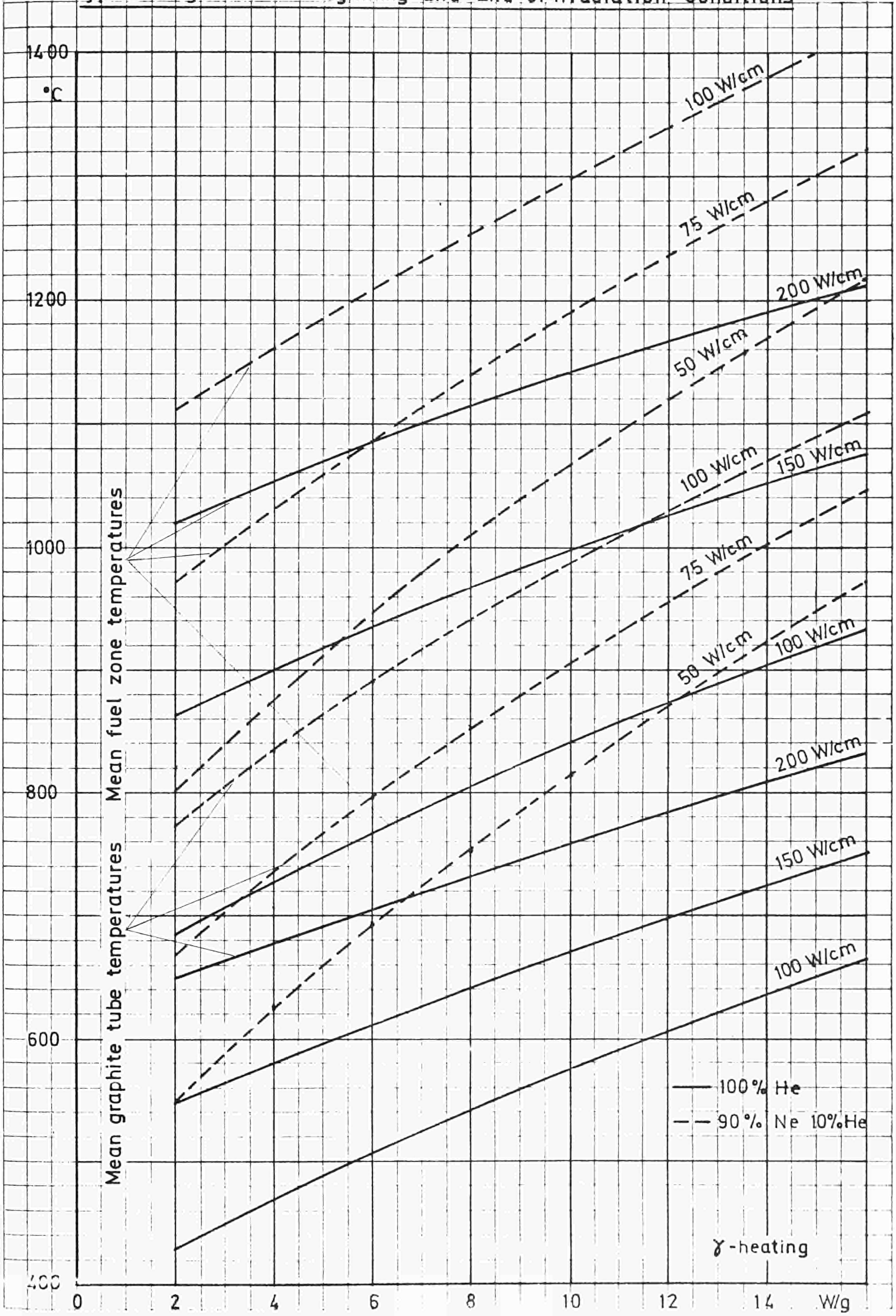
The neutron flux and gamma heating profile as represented on figure 17 is assumed to be constant in time and in its axial position with regard to the rig specimen carriers. This latter condition can be fulfilled by means of the adjusting device (see paragraph 3.2.2. of this report). The initial fissile material loading is then determined for each level of the specimen carrier(s), starting with the "hot plane" so that at any point the maximum admissible fuel temperature is reached at least once during the irradiation, but never exceeded. An example of axial fuel distribution for this approach is represented on figure 18.

A comprehensive computer programme has been elaborated for this work, a schematic flow sheet of which is given in the annex to this report.

In an analoguous manner, one can analyze a "constant loading-variable gas gap" concept for temperature evolution in time.

C.P.U.R. FIG. 15

Typical Diagrams for Beginning and End-of-Irradiation Conditions



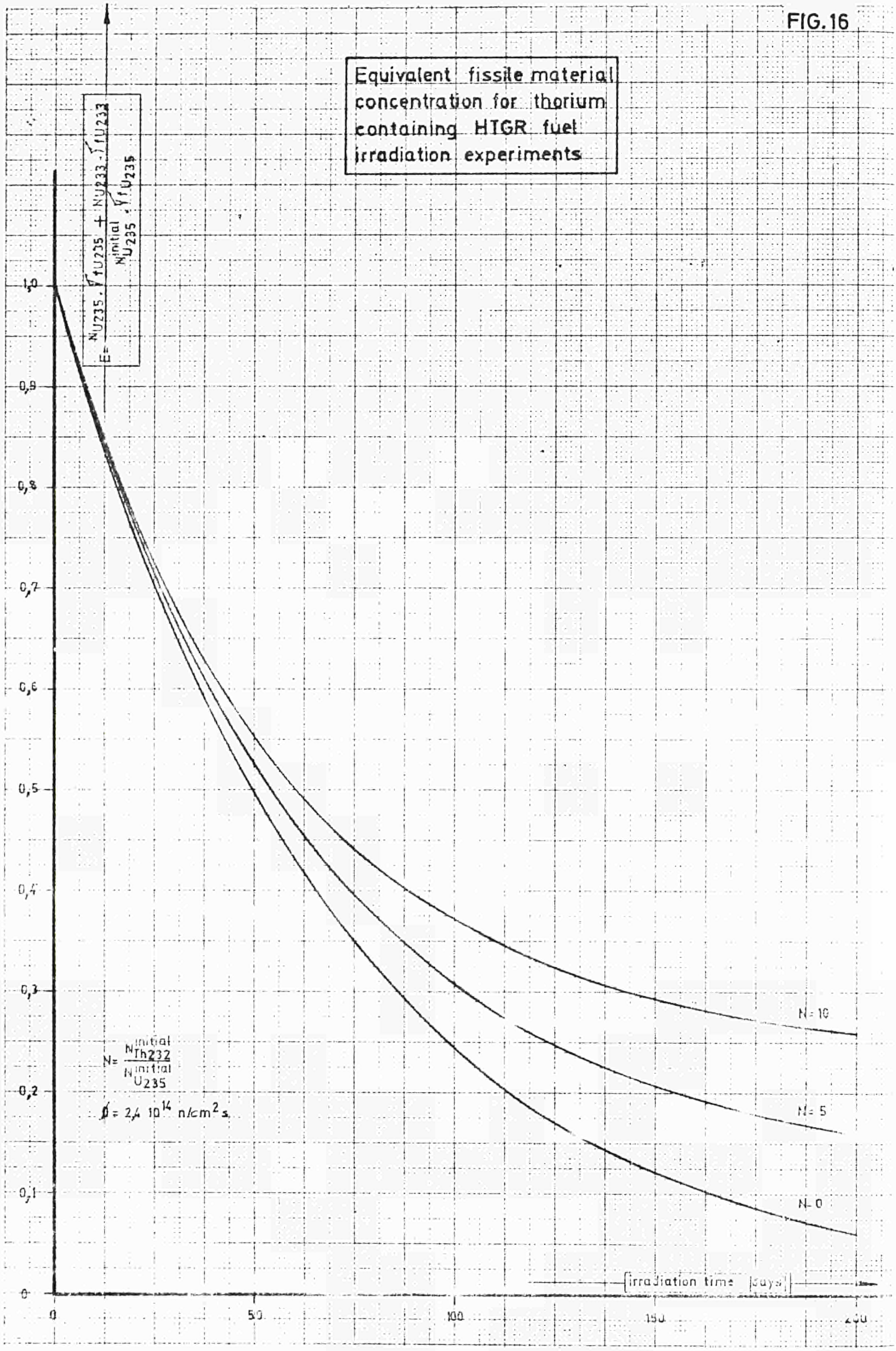
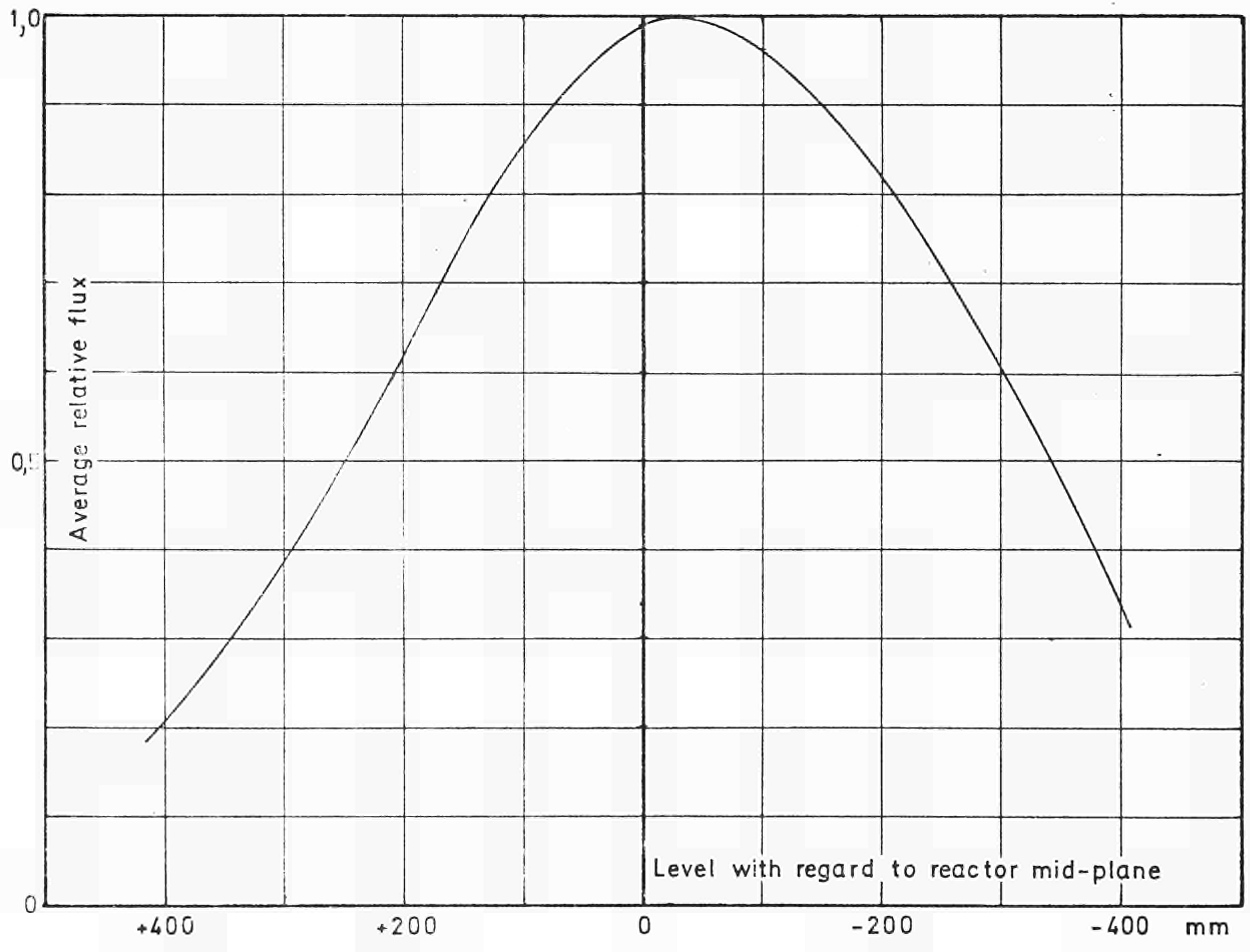


FIG. 16
Equivalent fissile material concentration for thorium containing HTGR fuel irradiation experiments
$E = \frac{N_{U235} \cdot \sigma_{fU235} + N_{U233} \cdot \sigma_{fU233}}{N^{initial}_{U235} \cdot \sigma_{fU235}}$
$N = \frac{N^{initial}_{Th232}}{N^{initial}_{U235}}$
$\phi = 2{,}4\ 10^{14}\ n/cm^2 s$
N=10
N=5
N=0
irradiation time [days]
1,0
0,9
0,8
0,7
0,6
0,5
0,4
0,3
0,2
0,1
0
0
50
100
150
200

FIG.17

Assumed Axial Profile of Average Neutron Flux and Gamma Heating



3.3.3.3. Variable Loading, Variable Flux.

The transition to the most general case is easily achieved by the above-mentioned programme by introducing time-dependent local flux and gamma heating coefficients. Normally, calculations for the beginning, the middle and the end of an irradiation cycle are sufficient.

Moreover, the shut-down case has been incorporated into the programme, allowing a more accurate prediction of the Np239 - Pu239 and Pa233 -U233 transitions between two irradiation cycles.

In its most general form, the code is used not only for the design of new rigs but also for the current evaluation of experiments under irradiation.

3.4. Sweep Gas Activity and Safety Analysis.

3.4.1. General.

Information on the concentration of radio-active isotopes in the loop is required for :

a. evaluation of the hazards resulting from gamma radiation emitted by certain loop components, and from contamination of the reactor containment building brought about by leaks in the circuit.

b. calculation of R/B* ratio for some of the fission gases.

The calculation methods used are based upon the following simplifying assumptions :

1. Isotopes with half-lives below 1 minute and those with fission yields below 0,5% are disregarded, as well as those without gamma ray emission.

2. All fission products considered escape from the fuel under irradiation with the same R/B* ratio. Short-lived precursors (like Sn133, Sb133, Te133) are not taken into account.

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*$R/B = \frac{\text{release rate}}{\text{birth rate}}$

Typical Initial Fissile Material Loading Profile

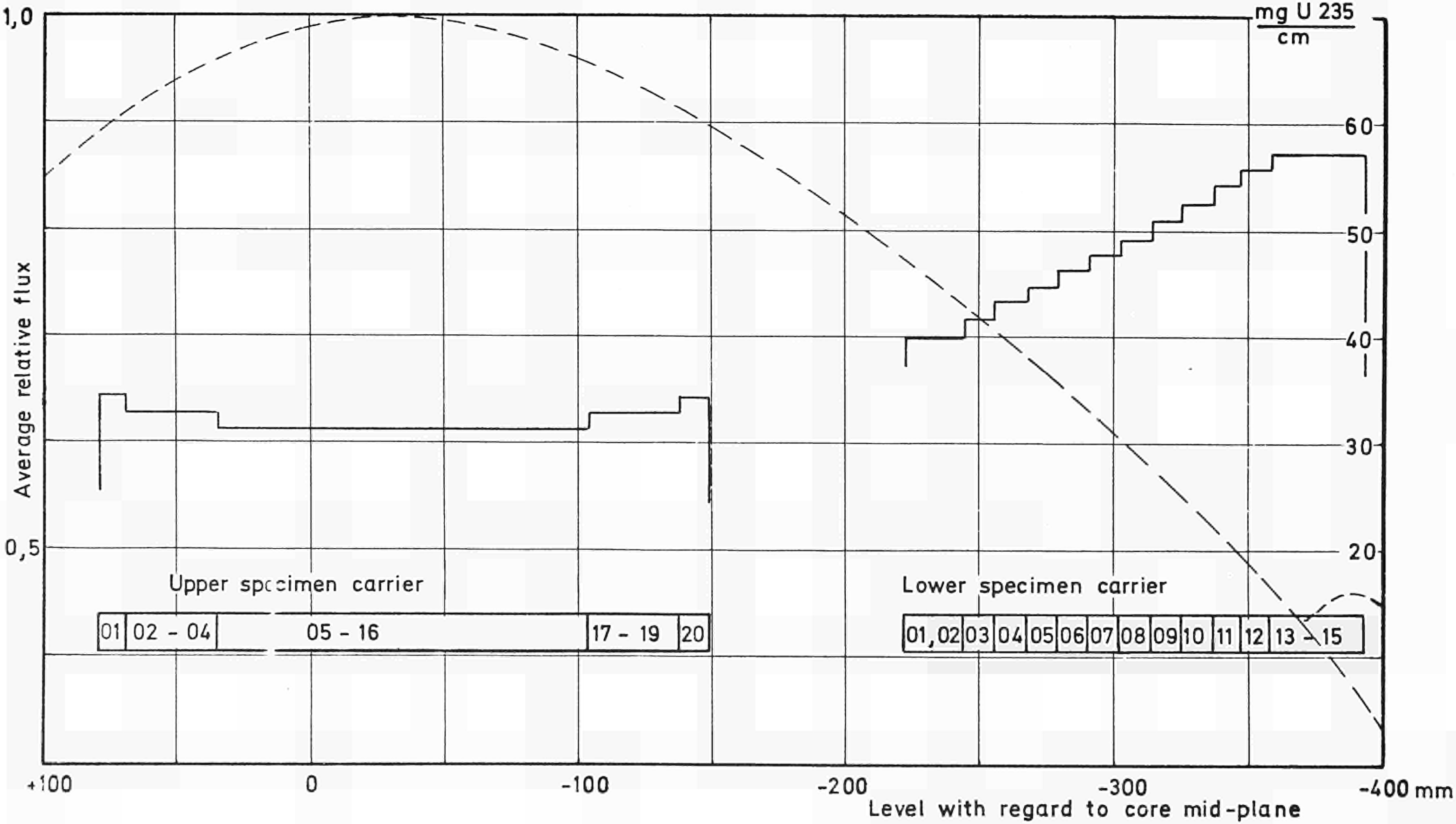

FIG. 18

3. The irradiation time preceding the moment of analysis is assumed to be sufficiently long so that all isotopes considered, except Kr85, are in equilibrium concentration.

4. The fission product transport velocity is identical to the gas displacement speed inside the tubes.

3.4.2. Specific and Accumulated Activities.

The calculations have been carried out using currently available data on fission products* and on xenon and krypton adsorption on active charcoal**. Main data are (see figure 19) :

- total fission power of the swept fuel volume	kW	5
- volatile fission product release fraction R/B	-	$10^{-5}...10^{-2}$
- volumetric sweep gas flow rate	cm^3/sec	0,56
- transport time between fuel and inlet to 5 litres active charcoal trap	sec	405
- delay time in the 5 litres active charcoal trap for Xe isotopes	hrs	750
for Kr isotopes	hrs	38
for halogens	(infinite).	

The main results of the analysis are given in table 6 below.

Kr85 concentration is calculated for 200 days of irradiation. A correction has been applied to the in-pile concentration (hence, release) of Xe135, with regard to its exceptionally high thermal neutron adsorption cross section, assuming an average perturbed flux of 2×10^{14} n/cm^2 s. I 129 is not taken into account, due to its extremely low specific activity.

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* ORNL 2127 Nucleonics (Nov. 1960).

** DP Reports Nr. 46 (1961), 244 (1963), 258 (1964). CEA-BIB-184 (1970).

TABLE 6

Sweep Gas Specific and Accumulated Activities.

Isotope	Fission Yield (%)	Half-life τ	Decay Constant k (s^{-1})	Gamma Emission (MeV)				Daughter Isotope	Specific before ACT	Activity* after ACT	Accumulated Activity** in ACT
Br 83	0,51	2,3 h	$0,838.10^{-4}$		0,051(20)			Kr 83 m	0,03	-	201
Br 84	0,92	32 min	$3,61.\ 10^{-4}$	0,27...	3,9	0,88	1,9	Kr 84	0,26	-	388
Kr 83m	0,51	1,86 h	$1,04.\ 10^{-4}$	0,009(100)		0,032(100)		Kr 83	0,04	-	199
Kr 85m	1,3	4,4 h	$0,45.\ 10^{-4}$	0,158(78)		0,305(22)		Kr 85 Rb 85	0,05	$12,36.10^{-5}$	629
Kr 85	0,293	10,3 y	$2,17.\ 10^{-9}$		0,52(1)			Rb 85	$0,05.10^{-5}$	$0,05.10^{-5}$	0,1
Kr 87	2,49	78 min	$1,48.\ 10^{-4}$	0,40	0,85	2,57		Rb 87	0,30	-	1103
Kr 88	3,57	2,8 h	$0,688.10^{-4}$	0,19	2,4...			Rb 88	0,19	$1,56.10^{-5}$	1541
Kr 89	4,8	3,2 min	$3,61.\ 10^{-3}$	0,21	0,45	0,60		Rb 89	3,05	-	464
I 131	3,1	8,05 d	$0,997.10^{-6}$	0,284(6)	0,64(9)	0,364(82)		Xe 131	0,002	-	1256
I 132	4,38	2,3 h	$0,838.10^{-4}$	0,53(30) 0,78(80)	0,67(100) 0,96(20)			Xe 132	0,29	-	1916
I 133	6,9	20,8 h	$0,917.10^{-5}$	0,53(94)	0,85(5)	1,4(1)		Xe 133	0,04	-	2807
I 134	7,8	53 min	$2,18.\ 10^{-4}$	0,86	1,1			Xe 134	1,20	-	3002
I 135	6,1	6,7 h	$0,288.10^{-4}$	1,14(37)	1,28(34)	1,8(11)		Xe135m Xe135	0,13	-	2542
Xe 133	6,9	5,27 d	$1,525.10^{-6}$	0,081(100)	0,16(0,1)			Cs 133	0,008	$12,66.10^{-5}$	2811
Xe135m	1,8	15,6 min	$7,42.\ 10^{-4}$		0,53			Xe 135	0,79	-	580
Xe 135	6,3	9,2 h	$0,209.10^{-4}$	0,25(97)	0,61(3)			Cs 135	0,003	-	99
Xe 137	6,15	3,9 min	$2,96.\ 10^{-3}$	0,27	0,45			Cs 137	4,16	-	769
Xe 138	5,74	17 min	$6,79.\ 10^{-4}$	0,42	1,78...			Cs 138	2,24	-	1808

* Units are : $\mu Ci/cm^3$ for $R/B = 10^{-5}$ and mCi/cm^3 for $R/B = 10^{-2}$
ACT = 5 litres active charcoal trap.

** Expressed in μCi for $R/B = 10^{-5}$ and mCi for $R/B = 10^{-2}$.

3.4.3. Safety Considerations.

3.4.3.1. Radiation Fields.

a. Pipework.

The calculations used for table 6 do not apply to the maximum possible radiation fields around the loop pipework near pool wall penetration S2, because of shorter possible transport times. Early estimations taking into account short-lived isotopes down to a few seconds of half-life resulted in fairly elevated gamma doses and lead to the use of additional shielding. However, experience showed that even under unfavourable conditions, i.e. :

- several kW of fission power,
- R/B for the "guide" isotopes (Xe 133, Kr 85m, Kr 88) 10^{-3} to 10^{-2},
- high flow rates (short transport times),

the gamma radiation doses measured at contact of tubes in the reactor building did not exceed about 25 mrem/h. The considerable difference between calculated dose (several R/h) and practice can be explained by the conservative assumption that short-lived isotopes are released with the same R/B as the above-mentioned "guide" fission products. In reality, short-lived fission products escape at a much lower rate. For this reason, no more shielding on pipework is provided for the recent installations.

b. 5-Litre Active Charcoal Trap.

This trap has a double shielding with a total of 15 cm Pb. At continuous operation with 5 kW fission power and $R/B = 10^{-2}$, the dose rate at the outer shielding surface is 25 to 50 mrem/h.

c. 0,5-Litre Low Temperature Bed T6.

T6 will finally settle all remaining fission products from the carrier gas stream. The accumulated total activity is 45 mCi maximum (i.e. for $R/B = 10^{-2}$), with a dose rate of about 20 mrem/h at contact of the (unshielded)purification unit. Roughly the same dosis can be expected for $R/B = 10^{-5}$, but by-passing T5.

3.4.3.2. Leaks.

Contamination from escaping sweep loop activity has been investigated for two cases (see figure 19) :

A. Failure in the gas return line at the inlet to the glove box system (Leak 1). The escaping active gas dilutes into the secondary containment atmosphere which is then sucked into the reactor building off-gas system ("ventilation non-recyclable").

B. Failure between gas mixing panel and purification unit (inlet line) where no secondary containment is provided (Leak 2). The loop depressurizes into the reactor building, with potential contamination of the working areas by escaping activity.

Results :

The calculations have been supported by a number of laboratory tests which were carried out during the "cold" run of the loop, using realistic lengths and diameters of tubing and measuring mainly leak flow rates and depressurization times for different leak sizes.

All calculations are based on 5 kW fission power in the swept fuel zone and R/B = 10^{-2}.

Case A.

Leak 1 brings about the following phenomena :

- time to drain the entire loop overpressure	sec	200
- total possible activity escaping into the glove box	Ci	23
- resulting specific activity in the glove box atmosphere (assuming homogeneous dilution)	mCi/cm^3	0,045
- flow rate from glove box to reactor stack	cm^3/sec	1100
- time for one complete glove box air exchange	min	8
- resulting total specific activity in reactor stack	$\mu Ci/cm^3$	$1,5 \times 10^{-3}$
- resulting weighed specific activity in reactor stack	$\mu Ci/cm^3$	5×10^{-5}.

For the assessment of the "weighed" activity, short-lived isotopes are disregarded, and only the following are taken into account : Kr87, Kr88 + Rb88, I131, I132, I133, I134, I135. The high level alarm of the gaseous gamma monitor in the stack is set to 5×10^{-6} µCi/cm^3 and would thus be largely exceeded. However, iodine is unlikely to appear out-of-pile at the same R/B rate as noble gases. Measurements taken suggest an apparent escape rate for iodine between 0,1 and 1% of the R/B for noble gases. Under such circumstances (i.e. R/B = 10^{-2} for noble gases and 10^{-4} for halogens) the weighed specific activity in the reactor stack would be about :

$$1,1 \times 10^{-5}\ \mu Ci/cm^3$$

(mainly Kr87, Kr88-Rb88), i.e. still bring about a transient "high gamma" alarm at the stack. Due to the limited duration, this is considered to be acceptable.

Case B (leak 2).

Here, the potential hazard is characterized by flow inversion and direct escape of active gas from the in-pile section via the inlet line (FI7, V40). Immediate flow inversion has been confirmed by the out-of-pile tests (see fig. 19), but calculations show that the loop would probably be completely depressurized through the leak before any active gas reaches the failure point. This is due to the large volume of inactive gas occupying the upper part of the rig tube (thimble), as well as to volume ratios and flow sections in the out-pile portion. However, for safety reasons, complete mixing of active gas into the upper part of the thimble in the moment of flow inversion has been assumed, and the resulting escape rates were calculated. The results are resumed as follows :

- time to drain the entire loop overpressure,			
assuming a leak equivalent to a Ø 1 mm hole		sec	190
assuming a leak equivalent to a Ø 3 mm hole		sec	23
- the total possible activity escaping into the reactor building			
assuming complete dilution with the non-active gas in the upper part of the rig,	1 mm hole	Ci	3,8
	3 mm hole	Ci	14
- contamination of the upper part of the reactor building (5th and 6th floor)	1 mm hole	µCi/cm^3	$2,5 \cdot 10^{-4}$
	3 mm hole	µCi/cm^3	$9,3 \cdot 10^{-4}$

Again the "weighed" contamination would be much smaller, i.e. in the 10^{-5} $\mu Ci/cm^3$ range, but still largely in excess of the admissible short-time exposure concentrations of

$1,2.10^{-7}$ $\mu Ci/cm^3$ (I.C.P.R.).

Practically, this event would call for a reactor shutdown and evacuation of the reactor building upon alarms from local gas and dust activity monitors. However, considering the remote probability of the accident conditions (simultaneously : full power, full R/B, complete mixing, important leak) this risk has been accepted.

Note that the I131 concentrations remain below MPC under all assumptions.

3.4.4. Gas Sampling and R/B Calculations.*

Following up the total sweep gas activity and measuring R/B throughout the irradiation give the best image of fuel particles behaviour during testing. The release rate of an isotope can be correlated to its specific activity in the sweep gas by :

$$A_S = 5,8.10^6 \cdot P \cdot \frac{Y \cdot r}{D \cdot \tau} \cdot e^{-\frac{0,693\ V}{D \cdot \tau}}$$

where

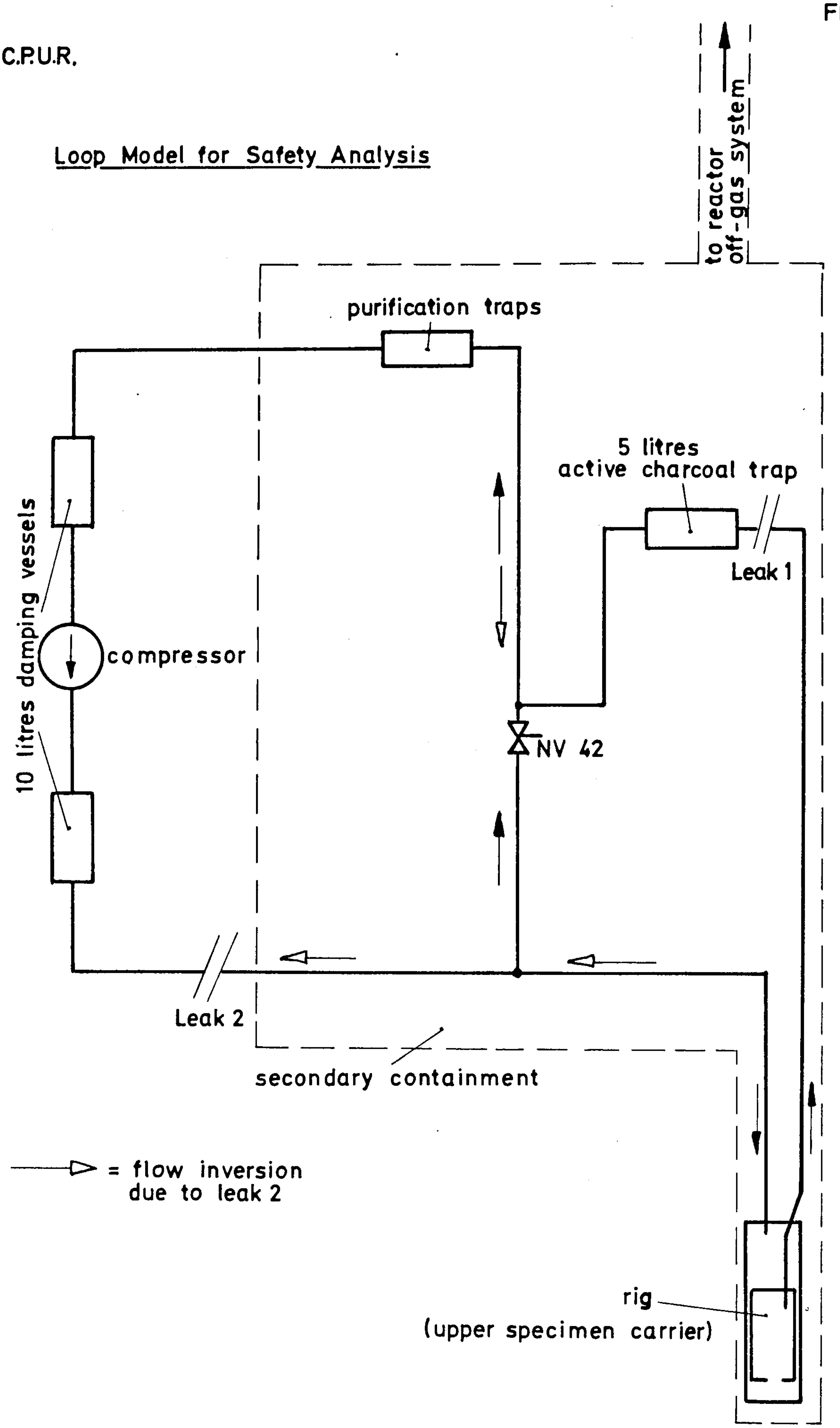
A_S	= specific activity of the isotope considered in a gas sample taken	($\mu Ci/cm^3$)
P	= total fission power of the swept fuel zone at time of gas sampling	(kW)
Y	= isotope fission yield	(%)
r	= R/B	-
D	= sweep gas flow rate	(cm^3/s)
τ	= isotope half-life	(s)
V	= total pipe volume between fuel zone under irradiation and sampling point	(cm^3)

* See also D.P. Report 715 (1970).

FIG. 19

C.P.U.R.

Loop Model for Safety Analysis



Some of the long-lived noble gas isotopes can be measured directly : The sweep gas is deviated into the sampling station (glove box) where a calibrated vessel is filled. According to the overall activity in the sweep gas, 5 cm^3 to 100 cm^3 vessels of standardized geometry are chosen. They can be placed directly onto the Ge-Li detector of the BR2 4000-channel-analyzer which is installed in an own laboratory outside the reactor containment building (low gamma back-ground). Isotopes currently measured are the "guide" fission products Kr85m, (Kr87), Kr88, Xe133, (Xe135).

The results are directly available in ($\mu Ci/cm^3$) for R/B calculation.

Short-lived isotopes can be determined indirectly through gamma scanning of their daughter products.
One of the techniques used is described as follows :

- Connect an active-charcoal-filled sampling trap to the glove box system.
- Provide liquid nitrogen cooling of the trap for "absolute" adsorption.
- Divert the return gas flow through the trap, measuring flow rate and adsorption time.
- Allow for decay about one hour after switching back to normal flow path.
- Heat and evacuate the trap in order to purge the remaining long-lived noble gases.
- Count the trap on the gamma spectrometer.

For practical reasons the accuracy of the method described is not very high : The half-lives of the daughter products concerned are either short (Kr89 $\longrightarrow$ 15 m Rb89, Xe 138 $\longrightarrow$ 32,3 m Cs138) or long (Xe 137 $\longrightarrow$ 30a Cs137) resulting in timing errors or weak gamma signals. Moreover, the transport mechanism of these isotopes is not well known. Consequently they could be swept into the sampling trap from decays between rig and glove box or partly purged out with the long-lived noble gases. More "sophisticated" methods, like a charged wire trap (Ref. 2) have not yet been used in BR2.

3.5. Present State of the C.P.U.R. Series (see table 3 of this report).

The prototype CPUR was irradiated in BR2 from September 1970 to April 1971, reaching :

- 137 equivalent full power days of irradiation,
- 1350°C constant fuel centre temperature,
- 2.1×10^{21} n/cm^2 fast neutron dose ($>$ 0.1 MeV),
- a burn-up of 5.4% fima or 60% fifa.

Post-irradiation examinations are being carried out by KFA Jülich.

The second CPUR has been under irradiation since February 1971, running at about 1200°C maximum fuel temperature. Both rigs contain fuel compacts in the upper and lower compartments.

Further rigs and another double-loop sweep facility are under construction. The scheduled loading for the end of 1971 is four experiments in parallel.

4. General Remarks on Operating Experience.

4.1. In-Pile Sections (rigs).

An analysis of the 12 HTGR fuel rigs which have been irradiated in BR2 from 1966 to 1971 shows two main types of operating difficulties.

a. Internal leaks.

Leaks from the specimen carrier towards the outer rig gas atmosphere caused fission gas contamination of the regulation gas circuit in some of the early rigs and required special safety measures on the out-pile equipment (activity monitoring, active charcoal traps, shielding). In most of the cases this phenomenon is attributed to the use of niobium as a carrier material which exhibits severe embrittlement due to oxygen take-up at elevated temperatures.

b. Thermocouples.

Failures of different types of high-temperature thermocouples occurred frequently in the early rigs (Ref. 16 and 24).

At present, the choice of possible noble-metal thermocouples has been deliberately limited to tungsten-rhenium couples (W 3% Re/W 25% Re or W 5% Re/W 26% Re) with BeO insulation which are reported to be reliable up to 2200°C (Ref. 23). As far as sheath material is concerned W 25% Re (or W 26% Re) and Mo have been retained, whereas Ta had to be discarded because of the formation of highly contaminating Ta 182.

4.2. Out-Pile Equipment.

The operation of sweep gas loops from 1967 to 1971 has caused the following main difficulties :

a. Liquid nitrogen supply.

The choice of the most adequate automatic supply system is explained in paragraph 3.2.3. of this report.

Insufficient thermal insulation of ducts and vacuum flasks caused untolerable liquid nitrogen consumption and extended ice-formation. The practical rules deduced from the observations are:
- liquid-nitrogen conducting tubes have to be isolated with at least 5 cm of foam insulation,
- welded stainless steel vacuum flasks do not supply the required degree of insulation over extended periods, and have to be replaced by glass containers.

b. Impurity Measurement.

The measurement of impurities on low level (i.e. below 10 vpm) by means of commercial instruments suffers from the following shortcomings :

- extended stabilization times (up to 1 hour) due to desorption phenomena,
- different readings on two identical instruments connected to the same gas stream (suggesting uncertainties of up to 100% in the 1 vpm region),
- limited life of the measuring cells,
- necessity for the utilization of calibration and/or regeneration gas bottles for certain instruments.

c. Activity Release.

Gas from the sweep loops has to be released into the reactor building off-gas system

- for pressure corrections under irradiation,
- for loop depressurization, and regeneration of the traps during shut-down periods.

No difficulties were encountered during off-power gas release since the overall loop activity decreases rapidly after shut-down, down to negligible levels. However, on-power release comprising an important fraction of short-lived isotopes can cause transient activity alarms in the reactor building and in the stack (see paragraph 3.4.3.2. of this report). Solutions to this were proposed by the loop operating instructions which prescribe release from low-activity points and with limited flow rates.

d. Low Flow Rate Measurements.

Accurate readings from low flow rate gas streams can be obtained by means of laboratory glass tube and float flow-meters. Difficulties occured due to

- fragility of the meter tubes,
- pressure dependence of the readings,
- inadequate leak tightness of several models used,
- extreme sensitivity to loop impurities (active charcoal dust, vacuum pump oil vapour) settling in the meter tube.

e. Oil Vapour Condensation from Vacuum Pumps.

Simple vapour traps turned out to be inadequate for the complete condensation of vacuum pump oil in the off-gas lines, and oil was found in extended sections of the loop tubing. From the experience gathered the use of large, water-cooled traps is considered as imperative.

5. Concluding Remark, Acknowledgments.

Capsule-type irradiation devices have been and are being irradiated in BR2 for the different high-temperature gas-cooled reactor concepts.

The existing devices are being extrapolated for various experimental purposes, including increased fission rating and temperatures, as well as layouts for gas-cooled fast breeder reactor studies.

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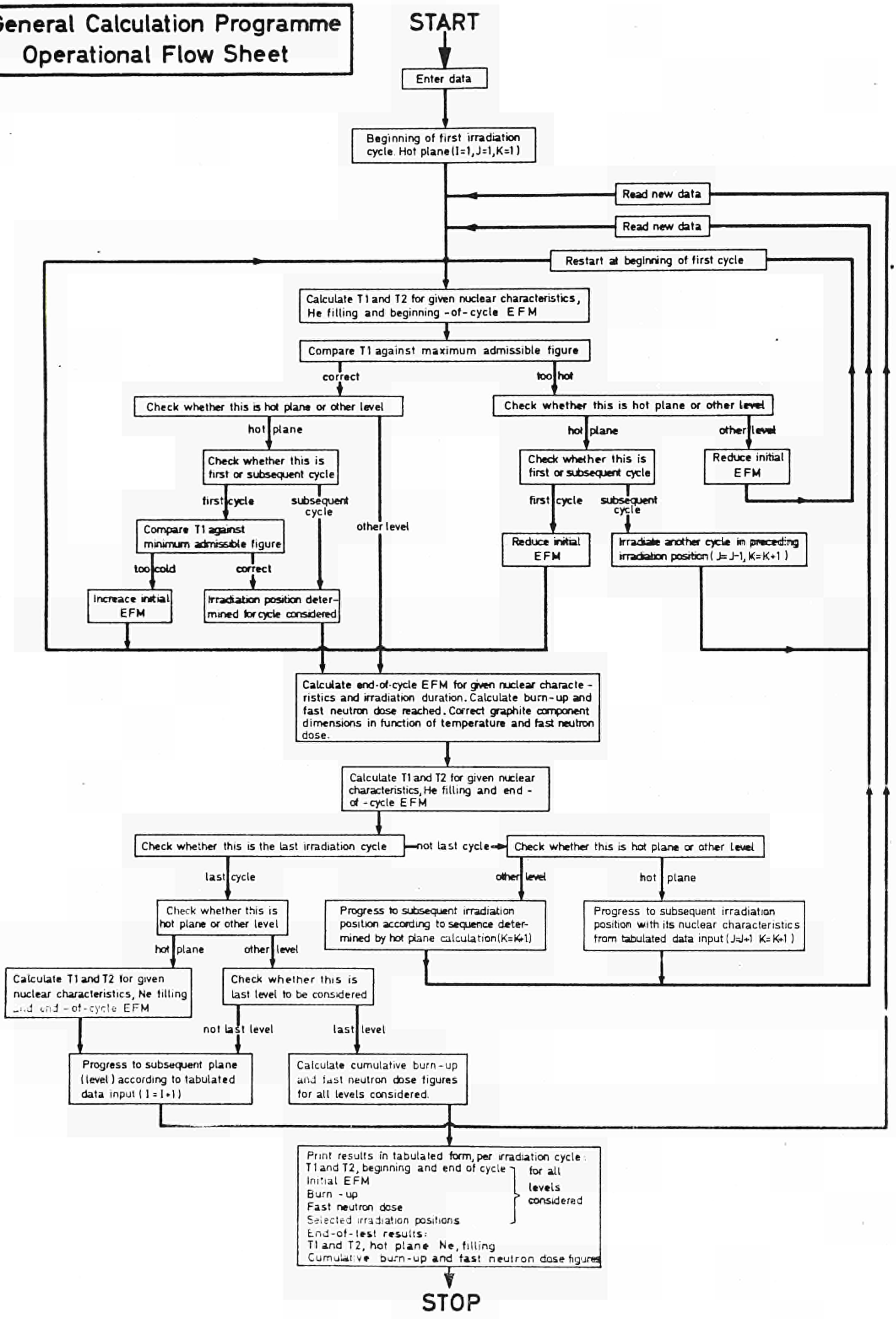
General Calculation Programme Operational Flow Sheet



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