PROGRAMME PROGRESS REPORT

January - June 1980

COMMISSION OF THE EUROPEAN COMMUNITIES
JOINT RESEARCH CENTRE
Petten Establishment
The Netherlands

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Operation of the High Flux Reactor

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JOINT RESEARCH CENTRE (JRC)
PETTEN ESTABLISHMENT

HFR DIVISION

Abstract

Within JRC's 1980/83 programme, the materials testing reactor HFR continued its scheduled operation.
During the first half year 1980, it reached 82.3°/o availability at 45 MW.
The occupation by irradiation experiments and radioisotope production devices varied between 59 and 76°/o.

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PROGRAMME PROGRESS REPORT, JANUARY - JUNE 1980

1. INTRODUCTION TO THE HFR PROGRAMME

The Council Decision of March 13, 1980, concerning the 1980/83 JRC research plan, defines for HFR Petten:

F. OPERATION OF LARGE-SCALE INSTALLATIONS
Supplementary programme

F.1. Operation of the HFR reactor (nuclear activity)

It also allocates a four years' budget of 52.22 MUCE and a staff of 88, of which 41 are research staff.
The proposed Technical Annex to the Council Decision reads as follows:

F.1. OPERATION OF THE HFR PETTEN
The operation of the reactor will continue to the benefit of the research programmes of the participating Member States and for the JRC's own requirements. Outside clients will also be able to use the irradiation facilities on payment.
During the next programme the teams will continue the maintenance and upgrading of the reactor and the development and improvement of the irradiation equipment and apparatus to enable this installation to hold its place among the most important means of irradiation of the Community.

1. DESCRIPTION

Within the new programme the reactor operation will be pursued for the benefit of research projects of participating member states as well as for own JRC projects. Available irradiation space will then be used, against payment, by other clients and for radioisotope production. The 1980-1983 irradiations will be carried out in support of:
- light water reactor development, in particular transient fuel irradiations and experiments in the scope of reactor safety,
- fuel and structural material testing for high temperature reactor development,
- fast reactor fuel development; advanced fuel tests under stationary and transient conditions as well as structural material irradiations,
- fusion reactor materials development.

The horizontal beam tubes, on the other hand will be used for nuclear physics and solid state physics research.
Improvements of the pool side irradiation facilities are planned within the new programme. They will be introduced together with the replacement of the reactor vessel. Moreover, the construction of a remote encapsulation facility for pre-irradiated fuel pins is foreseen.
For the development of reduced enrichment fuel for research and test reactors, the Petten Establishment will continue to act as a meeting point for experts in the field. It will also carry out irradiations and post-irradiation examens on test elements with enrichments below 20°/o ^{235}U, as a contribution to the international effort to reduce the quantities of highly enriched uranium used worldwide.

2. EXTERNAL COLLABORATION

With most Nuclear Research Centres within the Community.

3. PLANNING

Operation and utilization of HFR Petten are continuous tasks, following detailed time schedules which are updated periodically.

Unlike most of the other JRC research programmes, "HFR Operation" is not formally subdivided into individual projects. For the sake of the Programme Progress Reports, however, three projects are defined:
- 1. HFR Operation and Maintenance
- 2. Reactor Utilization
- 3. General Activities

(see Table of Contents).
On a lower level of subdivision, the term "project" appears again when referring to irradiation projects and individual supporting activities. These are also addressed by the "HFR Project Cards" which are used as a medium term management tool for JRC staff and reactor users.
The following "poster" pages inform about the main aspects of operation, utilization, and management of HFR Petten.

MAIN CHARACTERISTICS

ORR TYPE

45 MW (50)

H_2O COOLED/MODERATED Be MODERATED/REFLECTED

MTR (PLATE) TYPE FUEL ELEMENTS,

93 % ENRICHED, 1 g ^{10}B PER ELEMENT

29 IRRADIATION POSITIONS,	11 HORIZONTAL BEAM TUBES
COOLANT PRESSURE 0,24 MPa,	COOLANT TEMPERATURE 313 K
ANNUAL AVAILABILITY 75 %	AVERAGE UTILIZATION 70 %

MAIN UTILIZATION

- FAST BREEDER REACTOR STRUCTURAL MATERIALS IRRADIATIONS
- FAST BREEDER REACTOR FUEL PIN TESTING UNDER ABNORMAL CONDITIONS AND UNDER OPERATIONAL TRANSIENTS
- LIGHT WATER REACTOR FUEL PIN POWER RAMPING
- HIGH TEMPERATURE GAS COOLED REACTOR GRAPHITE AND FUEL ELEMENT IRRADIATIONS
- NUCLEAR STRUCTURE AND SOLID STATE PHYSICS EXPERIMENTS
- RADIOISOTOPE PRODUCTION, ACTIVATION ANALYSIS
- NEUTRON RADIOGRAPHY, NEUTRON DOSIMETRY DEVELOPMENT

HFR Petten

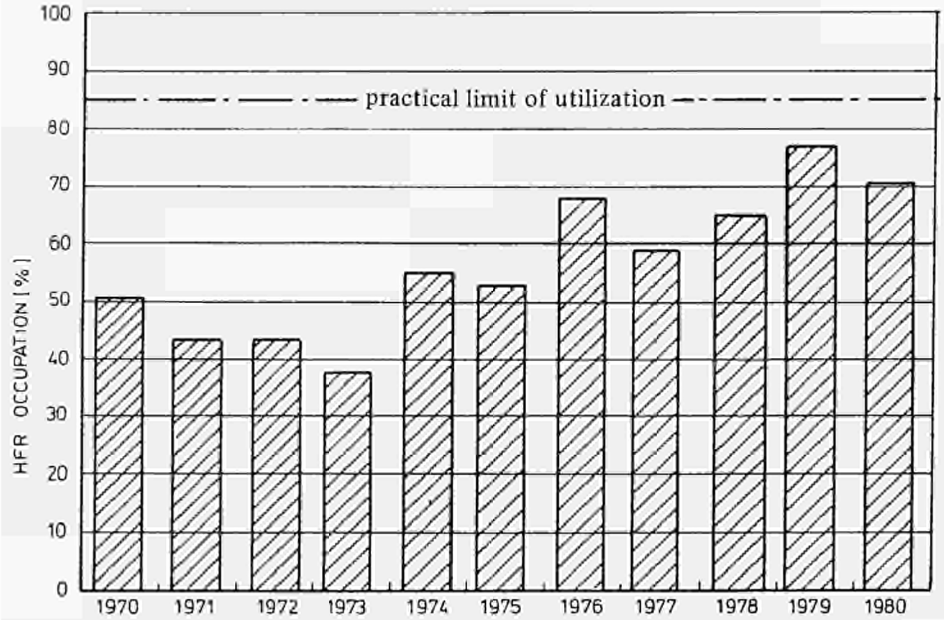
UTILIZATION

WHY ARE THESE EXPERIMENTS CARRIED OUT IN THE HFR?

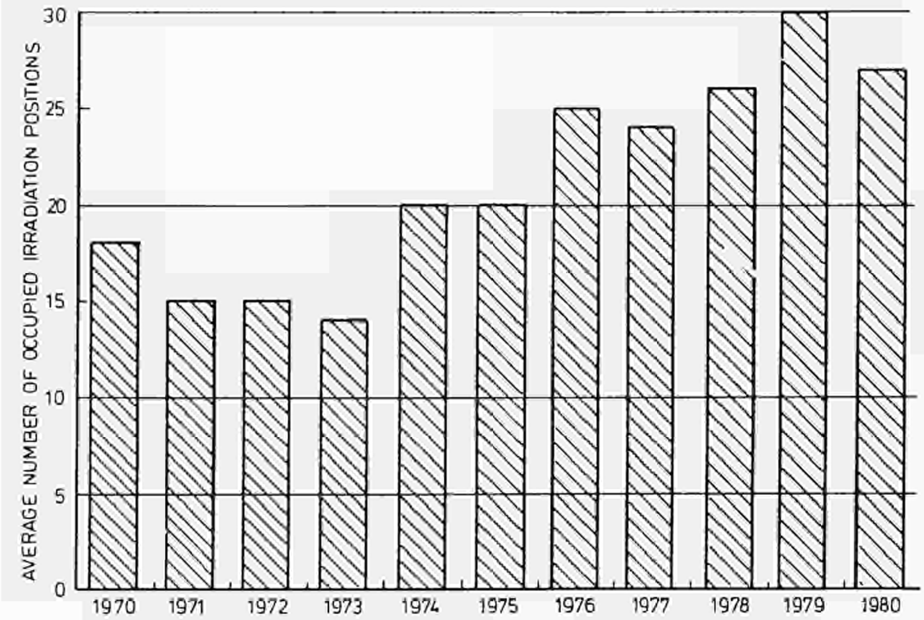
- FAST BREEDER REACTOR STRUCTURAL MATERIALS IRRADIATIONS: SEVERAL HIGHLY RELIABLE IRRADIATION CAPSULES AND SPECIAL HOT CELL EQUIPMENT AVAILABLE
- FAST BREEDER REACTOR FUEL PIN TESTING UNDER ABNORMAL CONDITIONS AND UNDER OPERATIONAL TRANSIENTS: LONG-STANDING EXPERTISE, AVAILABILITY OF THE PSF, AND OF SPECIAL IN-PILE INSTRUMENTATION, ADVANCED CONTROL EQUIPMENT
- LIGHT WATER REACTOR FUEL PIN POWER RAMPING: AVAILABILITY OF THE PSF AND OF LARGE CONTROL EQUIPMENT
- HIGH TEMPERATURE GAS COOLED REACTOR GRAPHITE AND FUEL ELEMENT IRRADIATIONS: WELL-KNOWN FLUX SPECTRA, SEVERAL PROVEN CAPSULE TYPES, HOT CELL RE-ENCAPSULATION FACILITY FOR ACTIVE SAMPLES
- NUCLEAR STRUCTURE AND SOLID STATE PHYSICS EXPERIMENTS: NUMEROUS EXPERIMENTAL INSTALLATIONS, LONG-STANDING EXPERTISE
- RADIOISOTOPE PRODUCTION, ACTIVATION ANALYSIS: REGULAR REACTOR OPERATIONS, MANY SPECIAL FACILITIES, HIGH NEUTRON FLUXES
- NEUTRON RADIOGRAPHY, NEUTRON DOSIMETRY DEVELOPMENT: MODERN, PURPOSEFUL EQUIPMENT, WELL-KNOWN FLUX SPECTRA

HFR Petten

REACTOR OCCUPATION BASED UPON IRRADIATION UNITS



REACTOR OCCUPATION BASED UPON THE NUMBER OF IRRADIATION POSITIONS



UTILIZATION CATEGORIES	(First half of 1980)
1. Materials testing	
Graphite and HTR fuel	10%
Structural materials	20%
LWR fuel	10%
Fast breeder reactor fuel	11%
2. Solid state and nuclear physics	21%
3. Radioisotope production facilities	17%
4. Miscellaneous	11%

HFR Petten

STANDARD IRRADIATION FACILITIES

1. NON-FISSILE MATERIALS TESTING

Graphite, stressed or unstressed, metal or graphite / He environment ⌀ 6.....20 mm; 300........1200 °C
Specimen recycling.

Steel, unstressed, Na environment, tensile specimens, ⌀ 6.....12 mm, 550......650 °C
also: non-cylindrical (Charpy, CT, etc.)

Low temperature Al specimen capsules (various sizes)

Graphite and steel creep facilities with on-line measurement

In-core instrumentation test rigs

2. FISSILE MATERIALS TESTING, IN-TANK FACILITIES

Single or triple, double-walled Na (NaK)- filled, 500.......1000 °C clad temperature, 400.......1200 Wcm^{-1}
Optional: Neutron screen, central thermocouple, fission gas pressure transducer, ...

Single or double-walled HTGR fuel rigs, graphite / He environment, 100......1000 Wcm^{-1},
800.....1500 °C fuel temperature. Continuous fission gas sweeping and analysis.

Under development: BF_3 operated power transient facility.

3. FISSILE MATERIALS TESTING, POOL SIDE FACILITY (PSF)

Single-walled Boiling Water Fuel Capsule (BWFC), variable power, 70...150 bar water pressure,
200....800 Wcm^{-1}, 250.....350 °C clad temperature. Continuous fission product monitoring
Pre-irradiated fuel pins
Optional: Different types of fuel pin instrumentation

Double-walled, Na (NaK)-filled, single or double carrier capsule. Variable power, 500....1200 Wcm^{-1},
400.....800 °C clad temperature. Fuel pin length up to 500 or 1600 mm.
Optional: Different types of fuel pin instrumentation.

Under development: Encapsulation facility for pre-irradiated fuel pins

4. MISCELLANEOUS

Different radioisotope production rigs, mostly reloadable during reactor operation.

Gamma irradiation facility.

Borosilicate glass pellet capsule

Two neutron radiography installations

Beam tube nuclear and solid state physics equipment: Several diffractometers and spectrometers, mirror and filter systems, with ancillary cryogenic equipment and process computers

5. STANDARD OUT-OF-PILE CONTROL FACILITIES

Gas mixing and control panels
Cooling water control circuits
Micro-processor based data loggers

HFR Petten

UPGRADING AND DEVELOPMENT

1966 - 1979

- POWER INCREASES 20 TO 30, 30 TO 45 MW
- INTRODUCTION OF BURNABLE POISON FUEL
- SEVERAL CORE CONFIGURATION CHANGES
- COMPLETE REPLACEMENT OF REACTOR AND GENERAL PURPOSE EXPERIMENTAL INSTRUMENTATION
- NEW IN-TANK EXPERIMENT PENETRATIONS
- SEVERAL IMPROVEMENTS ON MAJOR PLANT SYSTEMS
- IN-HOUSE COMPUTER CODE DEVELOPMENTS

FUTURE DEVELOPMENTS

- REPLACEMENT OF THE REACTOR VESSEL (1982/83)
- DEVELOPMENT OF REDUCED ENRICHMENT FUEL
- STUDIES FOR A POWER INCREASE TO 60 MW
- NEW REACTOR AND EXPERIMENT DATA LOGGERS (1980/81)
- MEDIUM ACTIVITY LABORATORY (1984/85)
- NEW DISMANTLING CELL TRANSFER SYSTEM (1980/81)
- ENLARGED COMPUTING FACILITIES
- NEUTRON BEAM QUALITY IMPROVEMENTS.

HFR Petten

PROGRAMME MANAGEMENT

- OPERATION AND UTILIZATION WITHIN JRC PLURIANNUAL RESEARCH PROGRAMMES
- OPERATION BY EXTERNAL ORGANIZATION UNDER CONTRACT
- PREDETERMINED FOUR YEARS' FUNDING, DETAILED ANNUAL BUDGETS
- LONG-TERM SUPPLY AND SERVICE CONTRACTS
- CENTRAL DECISION MAKING, VARIOUS ADVISORY COMMITTEES
- DETAILED PERMANENT PROJECT AND RESOURCE PLANNING
- INTEGRATED PROJECT WORKING GROUPS
- SEMI-ANNUAL PROGRAMME PROGRESS REPORTS AND FINANCIAL SURVEYS.

- TYPICAL HFR BUDGET:

REACTOR OPERATION AND MAINTENANCE STAFF (70)	20%
REACTOR MODIFICATIONS AND DEVELOPMENT	8
GENERAL SITE SERVICES	17
JRC TECHNICAL AND ADMINISTRATIVE STAFF (88)	29
TOTAL FUEL CYCLE (IN EQUILIBRIUM)	15
MAJOR INVESTMENTS	2
ELECTRICITY, WATER, INSURANCES	4
EXPERIMENTAL EQUIPMENT	5
	100%

Projects

2. PROJECTS

2.1 HFR Operation and Maintenance

2.1.1 Objectives

The operation, maintenance, and supporting activities of HFR Petten during the first half year 1980 aimed at
- inspection, maintenance and overhauling of all systems during the annual service period in January
- regular operation, with a high availability, for the remainder of the reporting period.

2.1.2 Methods

A detailed working schedule for the maintenance outage and a precise operating calendar had been established by the end of 1979. They called for six reactor cycles between the middle of January and the end of June, 1980.
More details on HFR operation practices have been published earlier [1,2,3,4].

2.1.3 Results

2.1.3.1 HFR Operation

Operation Summary

In the first half year of 1980, HFR was operated on full power of 45 MW during six complete reactor cycles. The yearly maintenance and shut-down period started on December 24, 1979 and lasted until January 14, 1980.
Table 1 gives a survey of the reactor operation schedule. The relation between shut-down and operation periods is given in Fig. 1, while Fig. 2 presents the relative occupation of the different HFR irradiation positions during this half year.

A detailed specification of the experimental utilization of the HFR during cycles 80.01 up to 80.06 is given in Figures 3 to 8.
Control rod position (bankwise) and reactor power during these reactor cycles are given in Figures 9 to 14.

Table 1 Survey of reactor operation

Dates (1980)	Programme	Accumulated reactor power
24.12.79-14.01.80	Maintenance period	
15.01 - 11.02	Cycle 80.01	1052 MWd
12.02 - 10.03	Cycle 80.02	1150 MWd
11.03 - 07.04	Cycle 80.03	1157 MWd
08.04 - 05.05	Cycle 80.04	1051 MWd
06.05 - 02.06	Cycle 80.05	1166 MWd
03.06 - 30.06	Cycle 80.06	1162 MWd
	Total	6738 MWd

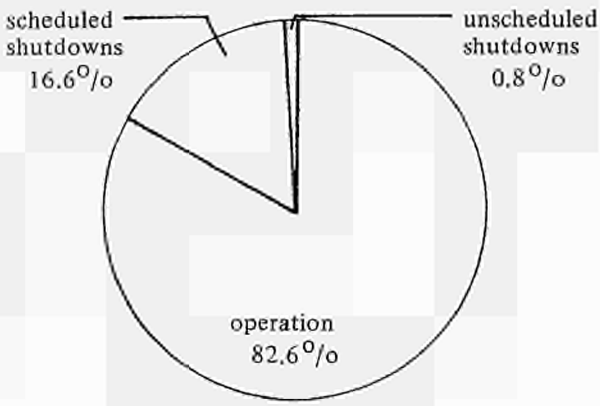


Fig. 1 HFR Operation.

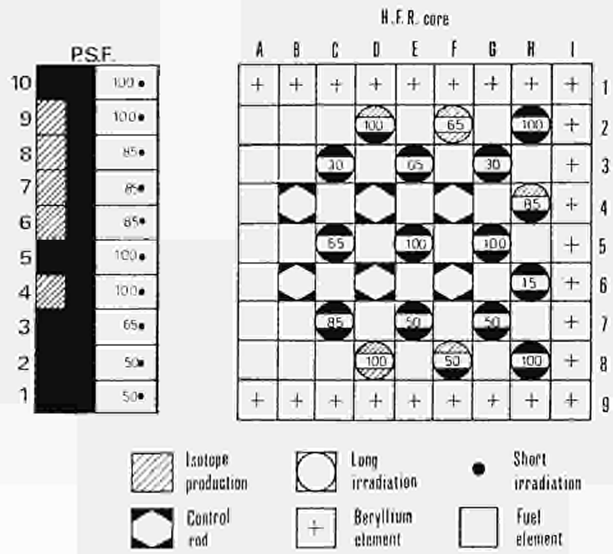
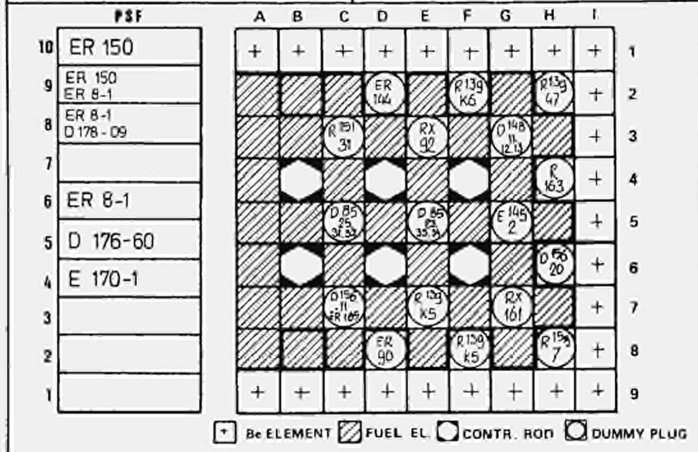


Fig. 2 Occupation of HFR irradiation positions in the first half year of 1980.

H.F.R. REACTOR LOADING FOR CYCLE : 80-01

START UP DATE : 16-1-1980
END OF CYCLE : 11-2-1980

REACTOR POWER : 45 MW



EXP. No.	DESCRIPTION	POSITION	*
R151-31	Niobium and Vanadium irr.TRIO 129-20,3 legs (NIRVANA)	C3	
D85-25	Low temp.Graphite TRIO 129-18, leg I	C5	
D85-31,32	Intermed.Temp.Graphite TRIO 129-18,leg II,III	C5	
D156-11	Graphite Creep Rig TRIO 129-12,leg I(DISCREET)	C7	
ER185-01	Automatic Gas Station TRIO 129-12, leg II	C7	
ER144	High Flux Facility for Isotopes (HIFI)	D2	
ER90	Reloadable Isotope Facility (RIF)	D8	
RX92	HFR Al. Irradiation	E3	
D85-34	Intermed.Temp.Graphite TRIO 129-17, leg III	E5	
D85-23,33	Low Temp.Graphite TRIO 129-17, leg I, II	E5	
R139-K6	Steel Irr. in Sodium TRIO 129-22, 3 legs(SINAS)	F2	
R139-K5	Steel Irr. in Sodium TRIO 129-03, 3 legs(SINAS)	F8, E7	*
D148-11,12 13	Mixed Carbide Irr.in special TRIO, 3 legs(CATRI)	G3	
E145-02	Austenitic Steel Irr.in Sodium TRIO 129-16(AUSTIN)	G5	
RX161	Gamma Heating Measurements	G7	
R139-47	Steel Irr. in Sodium REFA 170-07	H2	
R163	Instrumentation Test Capsule (INTEC)	H4	
D156-20	Graphite Creep TRIO 129-23,3 legs (DISCREET)	H6	
R158-7	Zircaloy Creep Irradiation (HOBBIE)	H8	
E170-01	Power Cycling Exp. POCY spec. supp.	psf 4	
D176-60	Power Ramp Exp. BWFC supp. 19	psf 5	
ER8-1	High Flux PIF (HF-PIF)	psf 8,6,9	
D178-09	Power Ramp Exp. BWFC supp. 9	psf 8	
ER150	Neutrography Camera	psf 9,10	
	* 23/1 Exp. from F8 to E7		

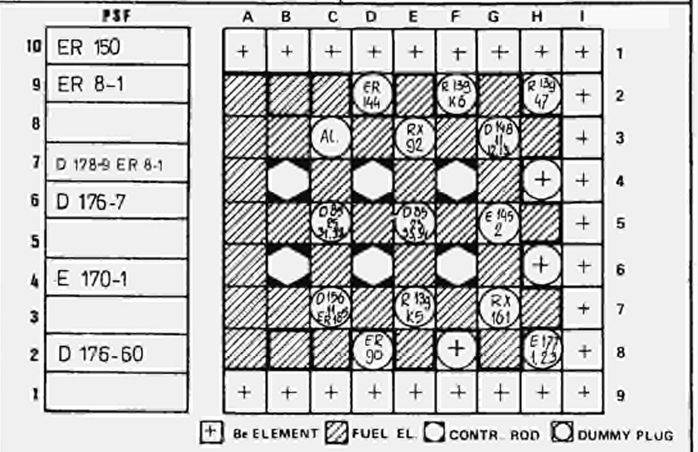
*REMARKS F: first loading P: plug type exp. no vessel entry S: short irradiation

Fig. 3 HFR reactor loading for cycle 80.01

H.F.R. REACTOR LOADING FOR CYCLE : 80-02

START UP DATE : 13-2-1980
END OF CYCLE : 10-3-1980

REACTOR POWER : 45 MW



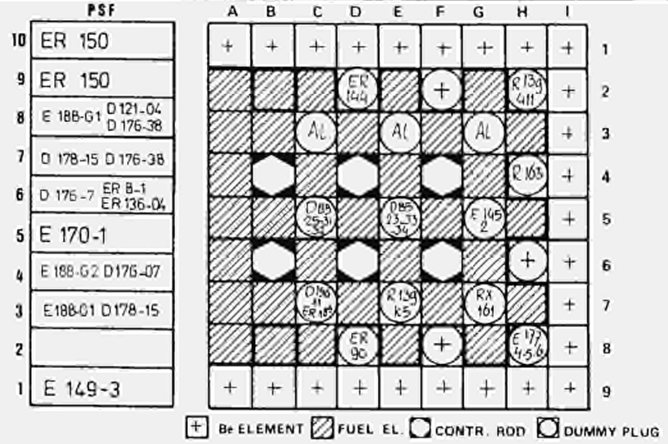
EXP. No.	DESCRIPTION	POSITION	*
D85-25	Low Temp. Graphite TRIO 129-18, leg I	C5	
D85-31,32	Intermed.temp.graphite TRIO 129-18,leg II, III	C5	
D156-11	Graphite Creep Rig TRIO 129-12 leg I(DISCREET)	C7	
ER185-1	Automatic gas station TRIO 129-12 leg II(AGATHE)	C7	
ER144	High Flux Facility for Isotopes (HIFI)	D2	
ER90	Reloadable Isotope Facility (RIF)	D8	
RX92	HFR Al. irradiation	E3	
D85-34	Intermediate temp.graphite TRIO 129-17,leg III	E5	
D85-23,33	Low temp. graphite TRIO 129-17, leg I, II	E5	
R139-K5	Steel irr. in sodium TRIO 129-9,3 legs (SINAS)	E7	
R139-K6	Steel irr. in sodium TRIO 129-22, 3 legs (SINAS)	F2	
D148,11 12, 13	Mixed carbide irr.in special TRIO 3 legs (CATRI)	G3	
E145-2	Austenitic steel irr. in Sodium TRIO 129-16 (AUSTIN)	G5	
RX161	Gamma Heating Measurements (TRAMP)	G7	
R139-47	Steel irr. in Sodium REFA 170-7	H2	
E177-1,2,3	Austenitic steel and tensile samples irr. in Sodium TRIO 129-24 (FANTASIA)	H8	
E170-1	Power cycling exp. (POCY) spec. supp.	psf 4	
D176-60	Power ramp exp. BWFC supp. 19	psf 5,2	
D173	FBR fuel pin irr. (DUELL) psf supp. 30	psf 5	
ER136	Fissile Isotope Target (FIT)	psf 6,5	
D176-7	Power ramp exp. BWFC supp. 14	psf 6	
D178-9	Power ramp exp. BWFC supp. 15	psf 7	
ER8-1	High Flux Poolside Isotope Facility (HFPIF)	psf 9, 7	

*REMARKS F: first loading P: plug type exp. no vessel entry S: short irradiation

Fig. 4 HFR reactor loading for cycle 80.02

H.F.R. REACTOR LOADING FOR CYCLE : 80-03

START UP DATE : 12-3-1980
END OF CYCLE : 7-4-1980
REACTOR POWER : 45 MW



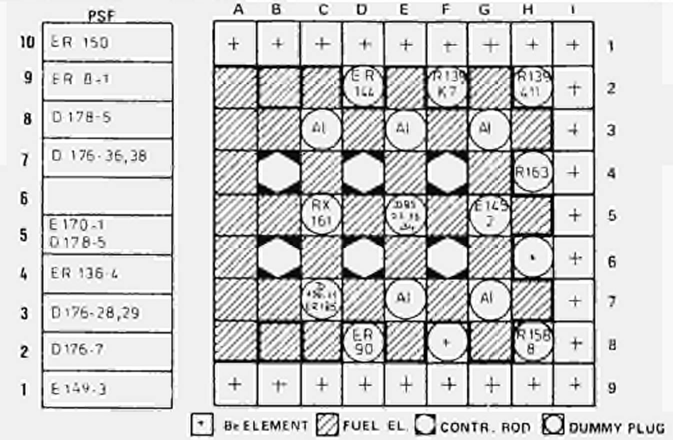
EXP. No.	DESCRIPTION	POSITION	*
D85-25	Low Temp. Graphite TRIO 129-18 leg I	C5	
D85-31,32	Intermed.Temp.Graphite TRIO 129-18, leg II, III	C5	
D156-11	Graphite Creep Rig TRIO 129-12 leg I (DISCREET)	C7	
ER185-1	Automatic gas Station TRIO 129-12 leg II (AGATHE)	C7	
ER144	High flux facility for isotope (HIFI)	D2	
ER90	Reloadable Isotope Facility (RIF)	D8	
D85-34	Intermediate temp.graph.TRIO 129-17,leg III	E5	
D85-23,33	Low temp.graphite TRIO 129-17 leg I, II	E5	
R139-K5	Steel irr.in sodium TRIO 129-9,3legs (SINAS)	E7	
E145-2	Austenitic steel irr.insodium TRIO 129-16 (AUSTIN)	G5	
RX161	Gamma Heating Measurements (TRAMP)	G7	
R139-4,11	Steel irr. in sodium REFA 170-11	H2	
R163	Instrumentation test capsules (INTEC)	H4	
E177-4,5,6	Austenitic steel and tensile samples irr. in sodium TRIO 129-13 (FANTASIA)	H8	
E149-03	Borosilicate glasses irr. (BONI) psf supp. 31	psf 1	
E188-G01	BWFC Gamma Test BWFC supp. 22	psf 3	
E188-G2	BWFC Gamma Test BWFC supp. 22	psf 4,8	
E170-01	Power cycling exp. (POCY) spec. supp.	psf 5	
D176-07	Power ramp exp. BWFC supp. 14	psf 6,4	
ER150	Neutrography camera	psf 9,10	
E121-04	Develop. LWR irr.	psf 8	
ER8-1	High Flus Isotope facility (HF-PIF)	psf 8,6	
D178-15	Power Ramp Experiment	psf 3, 7	
D176-29	Power Ramp Experiment	psf 3	
D176-38	Power Ramp Experiment	psf 7,8	
ER136-04	Fissile Isotope Target (FIT)	psf 6.	

*REMARKS. F: first loading P: plug type exp. no vessel entry S: short irradiation

Fig. 5 HFR reactor loading for cycle 80.03

H.F.R. REACTOR LOADING FOR CYCLE : 80-04

START UP DATE : 10-4-1980
END OF CYCLE : 5-5-1980
REACTOR POWER : 45 MW



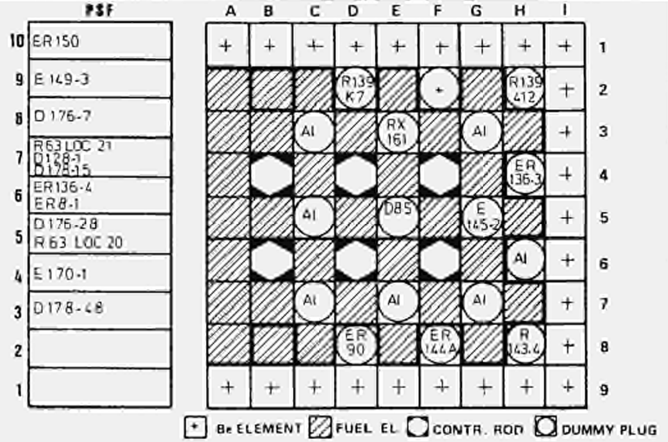
EXP. No.	DESCRIPTION	POSITION	*
RX161	Gamma Heating Measurements (TRAMP)	C5	
D156-11	Graphite Creep Rig-TRIO 129-12 leg I (DISCREET)	C7	
ER185-1	Automatic Gasstation TRIO 129-12 leg II (AGATHE)	C7	
ER144-1	High Flux Facility for Isotope (HIFI)	D2	
ER90	Reloadable Isotope Facility (RIF)	D8	
D85-23,33	Low Temp. Graphite TRIO 129-17 leg I, II	E5	
D85-34	Intermed.Temp.Graphite TRIO 129-17 leg III	E5	
R139-K7	Steel Irr.in Sodium TRIO 129-26,3 legs (SINAS)	F2	
E145-2	Austenitic Steel Irr. in Sodium TRIO 129-16, 3 legs (AUSTIN)	G5	
ER136-3	Fissile Isotope Target (FIT) spec. rig	G7	
R139-411	Steel Irr. in Sodium, REFA 170=11 (SINAS)	H2	
R163	Instrumentation Test Capsules (INTEC)	H4	
R158-8	Zircaloy Creep Irradiation (HOBBIE)	H8	
E149-3	Borosilicate Glasses Irr.(BONI) psf supp. 31	psf 1	
E188-G1	BWFC gamma test BWFC supp. 22	psf 1	
D176-7	Power ramp exp. BWFC supp. 14	psf 2	
D176-29	" " " BWFC supp. 19	psf 3	
D176-28	" " " BWFC supp. 19	psf 3	
D178-16	" " " BWFC supp. 9	psf 4	
D176-38	" " " BWFC supp. 21	psf 7	
D176-36	" " " BWFC supp. 21	psf 7	
D178- 5	" " " BWFC supp. 15	psf 8	
ER8-1	High Flux Poolside Isotope Facility (HFPIF)	psf 9	
ER150	Neutrography Camera	psf 10	

*REMARKS. F: first loading P: plug type exp. no vessel entry S: short irradiation

Fig. 6 HFR reactor loading for cycle 80.04

H.F.R. REACTOR LOADING FOR CYCLE : 80-05

START UP DATE : 6-5-1980
END OF CYCLE : 2-6-1980
REACTOR POWER : 45 MW



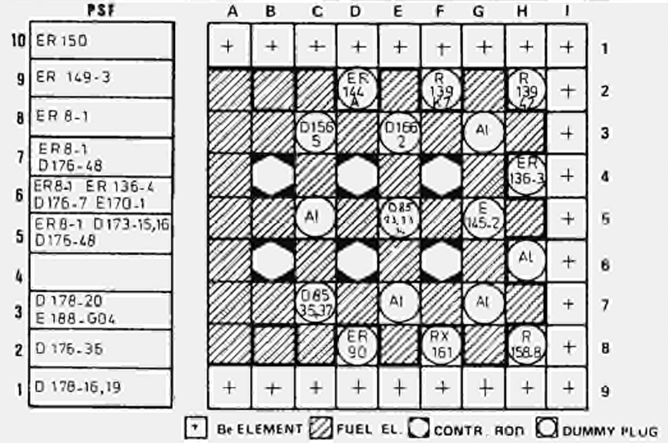
EXP. No.	DESCRIPTION	POSITION	*
R139-K7	Steel irr. in Sodium, TRIO 129-7 (SINAS)	D2	
ER90	Reloadable isotope facility (RIF)	D8	
RX161	Gamma heating measurements (TRAMP)	E3	
D85-23,33	Low temp. graphite irr. TRIO 129-17 legs I,II	E5	
D85-34	Intermed.temp. graphite irr. TRIO 129-17 leg III	E5	
ER144A	High Flux facility for Isotopes (HIFI)	F8	
E145-2	Austenitic steel irr. (AUSTIN)	G5	
R139-412	Steel irr. in Sodium	H2	
ER136-3	Fissile isotope target (incore FIT)	H4	
R143-4	Natrium steel irr. (MONA)	H8	
R158-8	Zircaloy creep irr. (HOBBIE)	H8	
D178-48	Power ramp exp.	psf 3	
E170-1	Power cycling exp. (POCY)	psf 4	
D176-28	Power ramp exp.	psf 5	
R63LOC20	Loss of cooling exp. (LOC)	psf 5	
R63LOC21	Loss of cooling exp. (LOC)	psf 7	
D128-1	Fuel stack displacement	psf 7	
D178-15	Power ramp exp.	psf 7	
D176-7	Power ramp exp.	psf 8	
E149-3	Borosilicate Glass (BONI)	psf 9	
ER8-1	High Flux Poolside Isotope Facility (HF.PIF)	psf 6	
ER150	Neutrography camera	psf 10	
ER136-4	Fissile isotope target (FIT)	psf 6	

*REMARKS. F: first loading P: plug type exp. no vessel entry S: short irradiation

Fig. 7 HFR reactor loading for cycle 80.05

H.F.R. REACTOR LOADING FOR CYCLE : 80-06

START UP DATE : 3-6 -1980
END OF CYCLE : 30-6 -1980
REACTOR POWER : 45 MW



EXP. No.	DESCRIPTION	POSITION	*
D156-5	Graphite Creep Rig(DISCREET) TRIO 129-24 3 legs	C3	F
D85-36	Intermed.temp.graph.TRIO 129-23 leg III	C7	F
D85-37	Low Temp.Graphite TRIO 129-13 leg II	C7	F
ER144A	High Flux Facility for Isotopes (RIF)	D2	
ER90	Reloadable Isotope Facility (RIF)	D8	
D166-2	Graphite Creep Irradiation REFA 140-15(CRIMP)	E3	F
D85-23,33	Low Temp.Graphite TRIO 129-17, leg I, II	E5	
D85-34	Intermed.Temp. Graphite TRIO 129-17, leg III	E5	
R139-K7	Steel irr. in Sodium (SINAS), TRIO 129-7,3 legs	F2	
RX161	Gamma Heating Measurement (TRAMP)	F8	
E145-2	Austenitic Steel Irr.in Sodium TRIO 129-17, (AUSTIN) 3 legs	G5	
R139-412	Steel irr.in Sodium (SINAS) REFA 170-9	H2	
ER136-3	Fissile Isotope Target (in-pile) spec.rig	H4	
R158-8	Zircaloy Creep Irr.(HOBBIE) spec. rig	H8	
D178-16	Power Ramp Exp. BWFC supp. 9	psf 1	
D178-19	" " " " " "	psf 1	
D176-36	" " " " " 19	psf 2	
D178-20	" " " " " 15	psf 3	
D173-15,16	FBR fuel pin irr. (DUELL) psf supp. 30	psf 5	
D176-7	Power Ramp Exp. BWFC supp. 14	psf 6	
E170-1	Power Cycling Exp. (POCY) Spec. support	psf 6	
D176-48	Power Ramp Exp. BWFC supp. 21	psf 5-7	
ER136-4	Fissile Isotope Target (FIT) Spec. supp.	psf 6	
E188-G04	γ- measurements BWFC supp. 15	psf 3	
ER8-1	High Flux PIF (HF-PIF)	psf 5, 6, 7, 8	
E149-3	Borosilicate glass (BONI) Spec. supp.	psf 9	
ER150	Neutrography Camera	psf 10	

*REMARKS F: first loading P: plug type exp. no vessel entry S: short irradiation

Fig. 8 HFR reactor loading for cycle 80.06

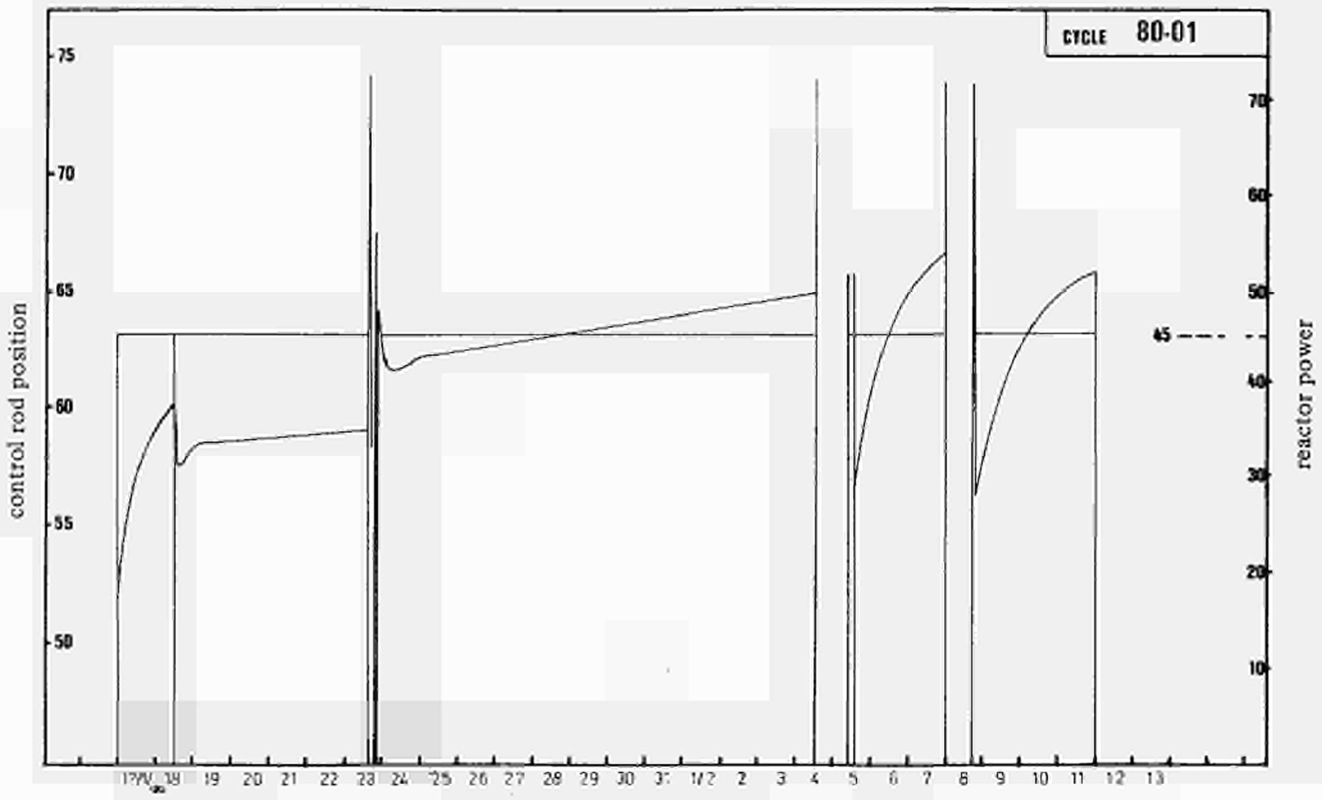


Fig. 9 HFR reactor power and control rod position vs. time, cycle 80-01.

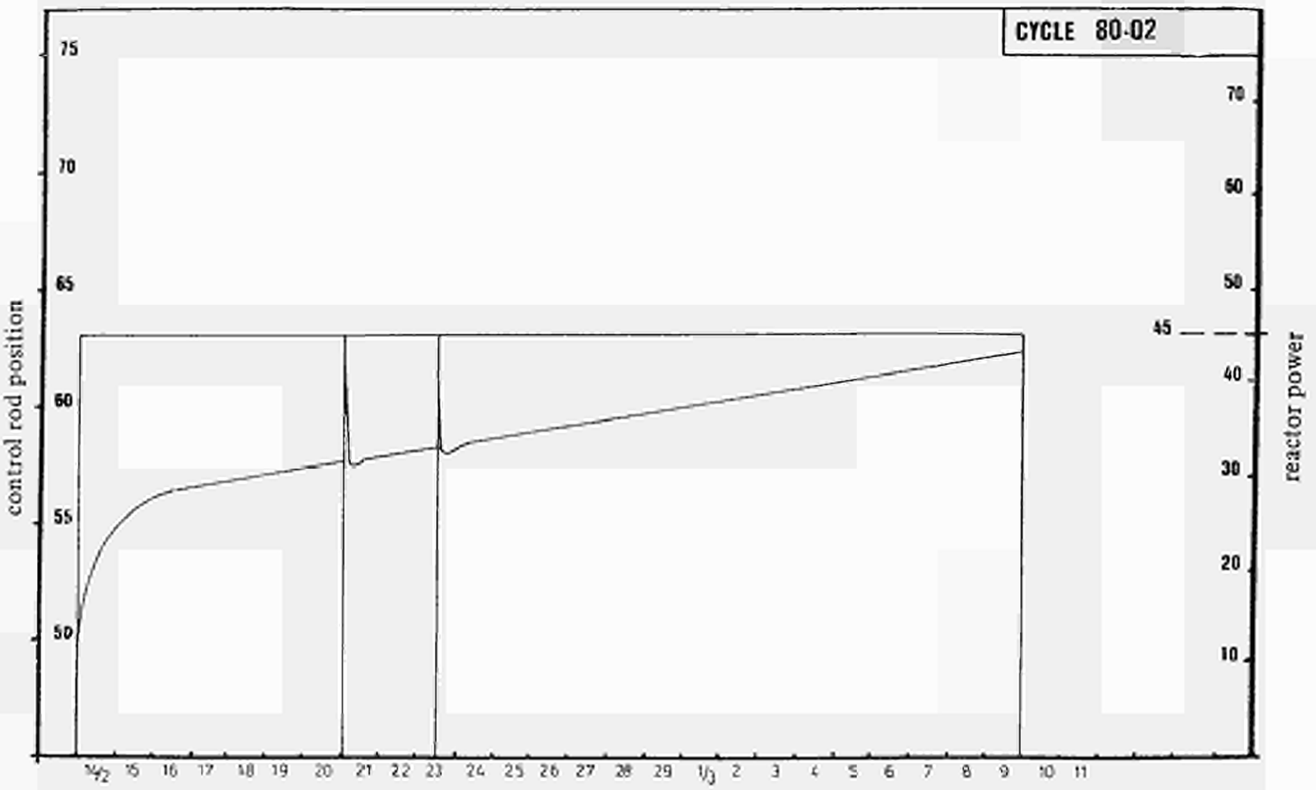


Fig. 10 HFR reactor power and control rod position vs. time, cycle 80-02.

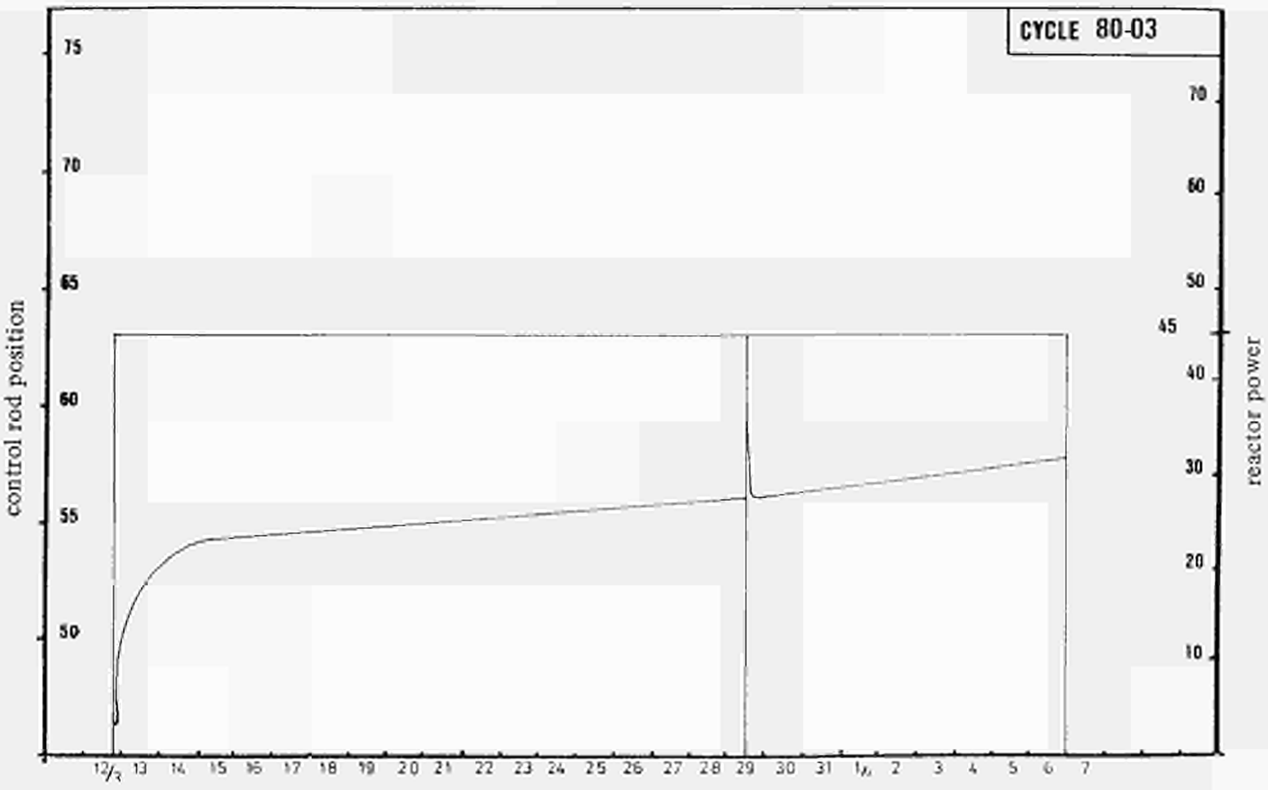


Fig. 11 HFR reactor power and control rod position vs. time, cycle 80-03.

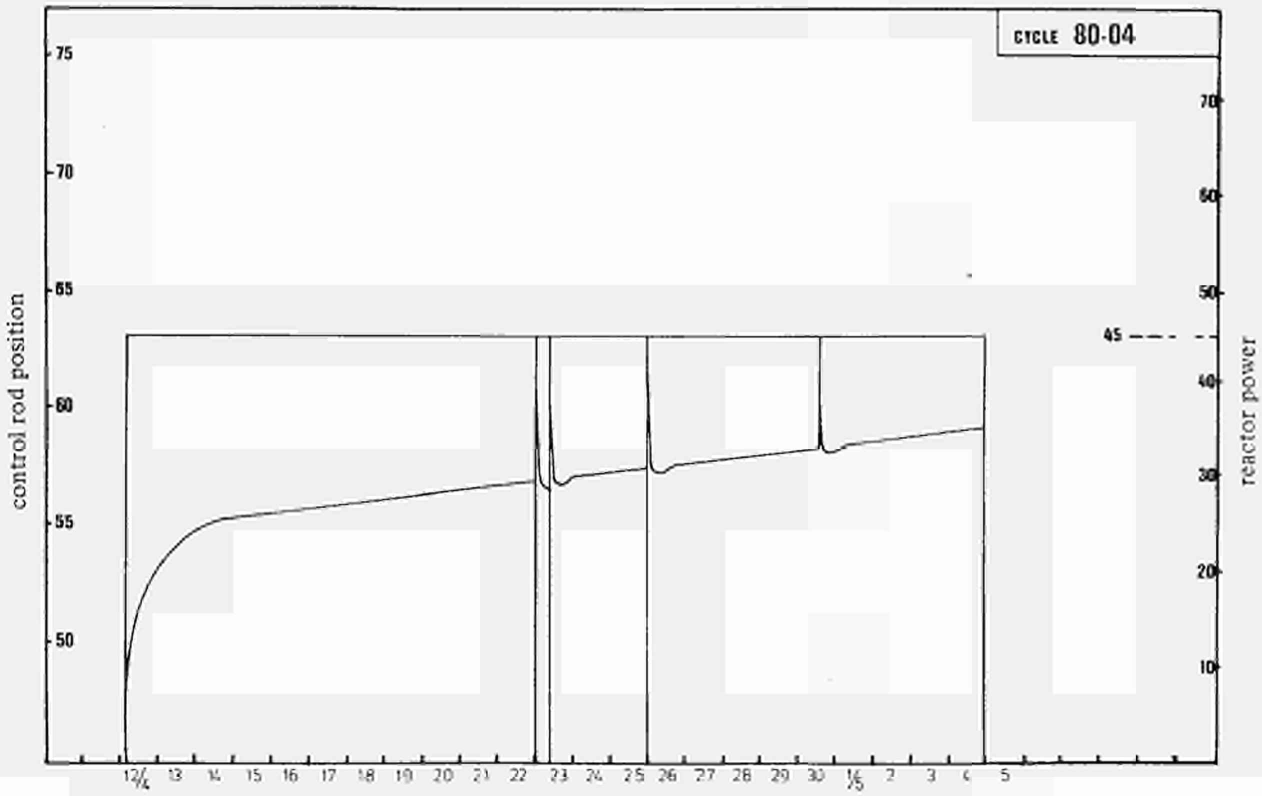


Fig. 12 HFR reactor power and control rod position vs. time, cycle 80-04.

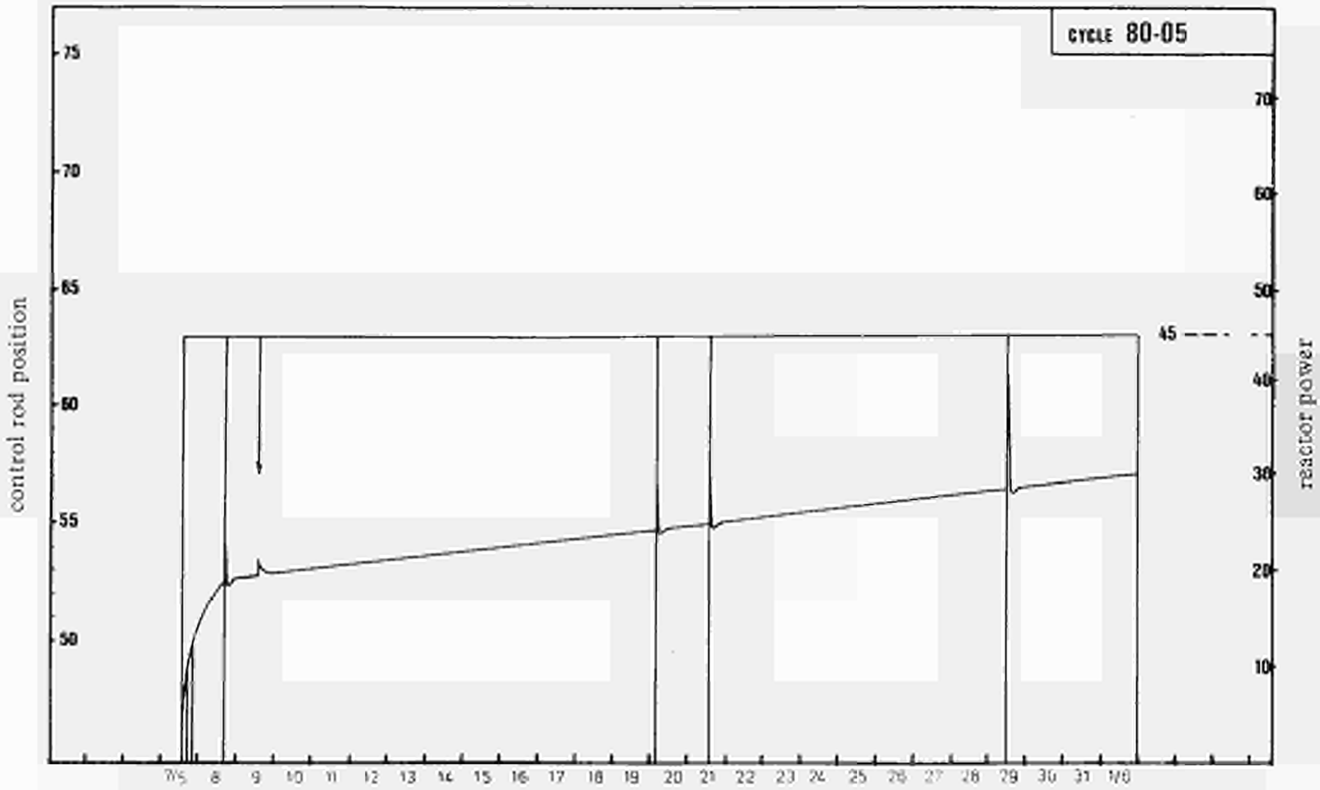


Fig. 13 HFR reactor power and control rod position vs. time, cycle 80-05.

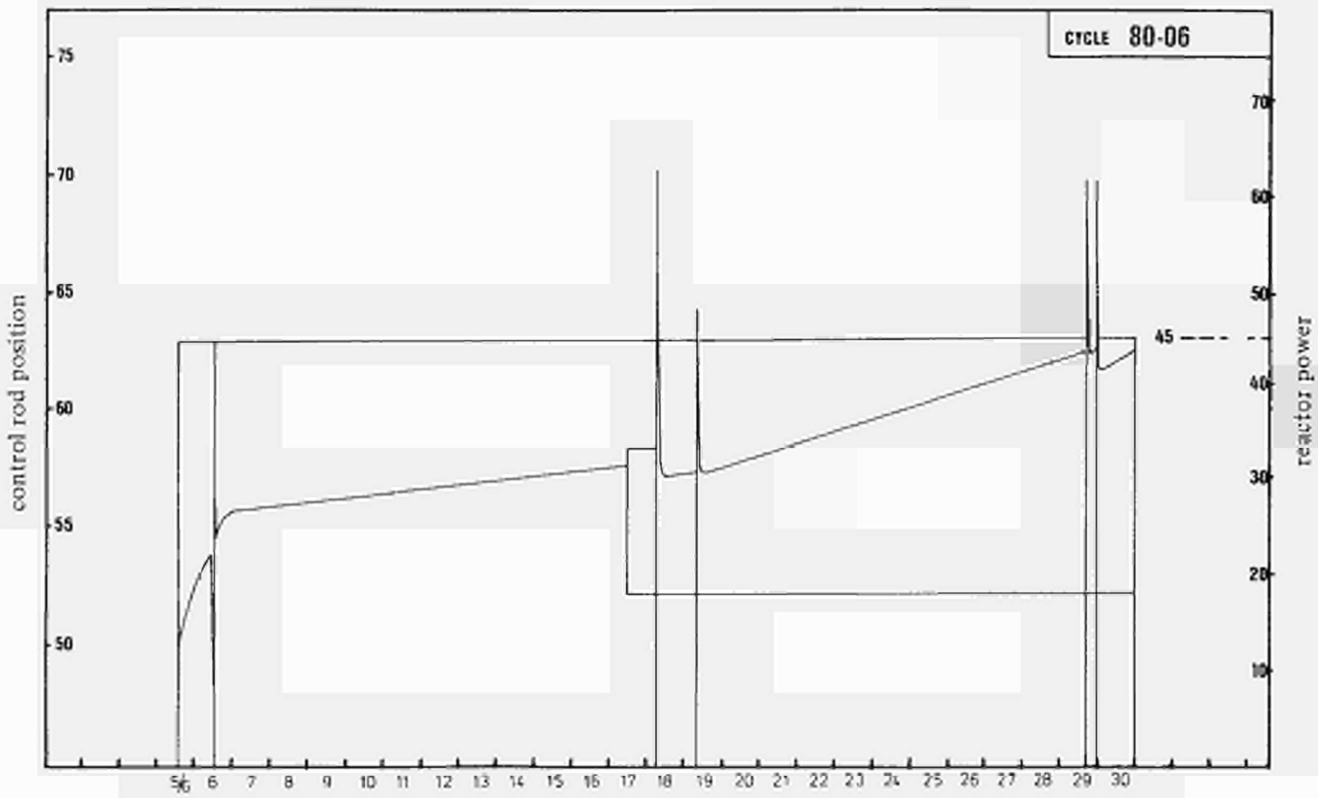


Fig. 14 HFR reactor power and control rod position vs. time, cycle 80-06.

In Table 2 some basic operation data during this half year are compared with the data of the previous half year. The table reveals that reactor operation was interrupted by 21 unscheduled scrams, two unscheduled power decreases, while five scrams and five power decreases were scheduled in accordance with various experimental requirements.

Two scrams, one due to a bent cooling water hose of the BWFC installation and one caused by a blown fuse of the reactor interlock system, resulted in xenon poisoning of the core, thus necessitating core reloading.
More detailed information is given in Table 3 on the intended and in Table 4 on the unscheduled reactor power deviations.

Table 2 Basic operating data.

Subject		2nd half 1979	1st half 1980
Integrated reactor power	MWd	5597	6738
Average reactor power	MW	44.9	44.8
Operating time	%	65.9	82.6
Unscheduled shutdown time	%	1	0.8
Unscheduled scrams	–	7	21
Unscheduled power decreases	–	–	2
Planned scrams	–	–	5
Planned power decreases	–	3	5
Fuel consumption	grs ^{235}U	6990	8416
Released through stack	Ci(^{41}Ar)	241.3	275.7

Table 3 Intended, mostly scheduled, manual reactor shut-downs and power decreases.

Date	Hour	Duration hrs./min.	Disturb. type	Reactor system or experiment	Cause	Code	Core nr.
18/1	12.20	00.21	MS	Exp. D85-25, 31, 32	Removal of experiment because of stuck displacement unit.	ME	80.01.1
23/1	09.05	00.12	MS	Exp. R139-K5	For improvement of irradiation conditions experiment replaced from position F8 to E7.	ME	80.01.2
23/1	13.25	00.12	MS	Exp. D85-25, 31, 32	Reloading of experiment.	ME	80.01.3
23/1	16.44	00.24	PD (20MW)	Exp. E170-01	Installation of experiment in p.s.f.	ME	80.01.4
18/4	15.00	00.02	PD (40MW)	Exp. RX161	Installation of experiment	ME	80.04.2
30/4	18.47	00.20	PD (25MW)	Exp. E170-01	Unloading of experiment from p:s.f.	ME	80.04.2
07/5	22.18	00.09	AS	Exp. R63 LOC 20	Scheduled transient	ME	80.05.1
08/5	15.08	00.04	AS	Exp. R63 LOC 21	Scheduled transient	ME	80.05.1
09/5	14.32	00.43	PD (30MW)	Exp. R158-8	Loading of experiment	ME	80.05.1
29/6	18.35	00.15	PD (30MW)	Exp. E170-01	Unloading of experiment from p.s.f.	ME	80.06.1

AS : Automatic scram
APD : Automatic power decrease
PD : Manual power decrease
MS : Manual shut-down
F : Failure system or operating condition

I : Instrument failure
O : Operating error
PS : Power supply failure
M : Intended stop or power decrease
R : Reactor
E : Experiment

Table 4 Unintended, unscheduled automatic power changes and scrams.

Date	Hour	Duration hrs./min.	Disturb. type	Reactor system or experiment	Cause	Code	Core nr.
04/2	13.05	18.30	AS	BWFC installation	Flow scram caused by accidently bending of cooling water hose, during in-pool manipulations. Xenon poisoning neccessitated core reloading.	OE	80.01.4
05/2	07.47	00.07	AS	Interlock	Blown fuse in interlock power supply system	FR	80.01.5
05/2	10.16	00.35	AS	Ibid	Ibid	FR	80.01.5
06/2	24.00	16.45	AS	Ibid	Ibid Fault was traced to defective fuse holder, which was replaced. Xenon poisoning necessitated core reloading.	FR	80.01.5
20/2	01.10	00.20	AS	Control rods	Spontaneous drop of control rods 5 and 6.	FR	80.02.1
23/2	11.29	00.20	AS	Exp. D173	High temperature limit exceeded during adjustment of gas content of experiment.	OE	80.02.1
10/3	08.14	00.16	AS	Flux channel	Unidentified disturbance.	IR	80.02.1
29/3	11.36	00.17	AS	Flux channel	Unidentified disturbance.	IR	80.02.1
12/4	09.26	00.03	APD (39MW)	Thermal column	Flow decrease shield cooling during manipulation with hydraulic rabbit system	OR	80.04.2
23/4	00.00	00.05	AS	Exp. E136-04	"High cooling pressure" scram due to failure of flow restrictor	FE	80.04.2
23/4	05.55	00.05	AS	BWFC-A	Pressure peak during filling of primary system of BWFC system A.	OE	80.04.2
26/4	00.32	00.05	AS	Control rod magnet unit	Faulty adjustment of control rod magnet current.	OR	80.04.2
07/5	20.09	00.03	AS	Exp. R63 LOC 20	High temperature scram due to low NaK level during preparation of transient.	FE	80.05.1
20/5	10.37	00.23	AS	Exp.ER136-03	Reactor power transient during unloading.	OE	80.05.1
21/5	15.48	00.20	AS	Exp.ER136-04	High pressure in cooling system caused by a loose flow restrictor.	FE	80.05.1
29/5	11.45	00.04	AS	BWFC-A	Short circuiting during instrumentation check.	OE	80.05.1
05/6	09.54	00.01	APD (41MW)	Exp. E149-3	"Low flow" caused by in-pool manipulations.	OE	80.06.1
05/6	10.02	00.18	AS	BWFC-A	Instrumentation switched off inadvertently	OE	80.06.1
18/6	10.15	00.22	AS	Exp.E136-03	Reactor power transient during loading of incore FIT.	OE	80.06.1
18/6	10.43	00.18	AS	Flux channel	Reactor power transient during equalization of control rods.	OR	80.06.1
19/6	11.24	00.23	AS	Exp.E170-01	During in-pool manipulations disadjustment of flow restriction.	OE	80.06.1
29/6	18.55	01.10	AS	Flux channel	Power overshoot during recovery from PD	OR	80.06.1
29/6	23.21	01.01	MS	Exp. ER136-03	Target holder dropped during unloading. Reactor shut-down manually.	FE	80.06.1

AS : Automatic scram
APD : Automatic power decrease
PD : Manual power decrease
MS : Manual shut-down
F : Failure system or operating condition

I : Instrument failure
O : Operating error
PS : Power supply failure
M : Intended stop or power decrease
R : Reactor
E : Experiment

2.1.3.2 Maintenance and Modifications

Major activities carried out during the winter maintenance period are summarized hereafter.

a) Mechanical Installation

Reactor vessel and associated structure

- Dimensional check on core box walls (deviations from design dimensions < 0.45 mm), lower grid plate (no deviations), and grid bars (0,40 mm play in grid bar nr. 1, other grid bars without deviations).
  Visual inspection of corebox walls (no remarks).

- Dimensional check of all top lid penetrations (some burrs removed), inspection of top lid fixations clamps and bolts.

- Replacement of control rod drive tubes nrs. 1, 2 and 3. Difficulties with the coupling of the hoisting tool and drive tube nr. 3 were traced to a failure of the bolts (diam. 1/4", length 30mm) by which the small top cover plate of the drive tube is connected to the drive tube proper. Due to this failure the cover plate got wedged into the control rod piston. This was discovered after an extended search action for the missing cover had been conducted inside the reactor vessel. No clear cause for the failure has been found. As it never happened before it will be considered as an isolated incident. Though the loosening of the cover plate has no direct safety implication, a repetition might cause problems during control rod installation and therefore special preloading inspection procedures have been implemented for future operations. Apart from a minor water leakage along one of the drive tubes (cause not yet identified) the inspection and overhaul did not yield any abnormal observations. The drive motors and associated parts have been inspected and overhauled where necessary.

Cooling systems

- Leakage check on main outlet valve of the secondary cooling system (360 l/hr at one bar overpressure).
- Hartford loops inspected, vacuum valves replaced.

- Primary cooling pump nr. 1 revised.

- Check valves on primary pump outlet inspected and overhauled. Minor wear observed.

- Motor driven outlet valves of primary pumps dismantled, inspected and overhauled.

- Technical preparations and execution of containment building leakage tests (April and June/July, 1980).

- Various hoisting, handling and storage provisions installed in and around reactor pools.

- Inspection of mechanical filters in primary cooling circuit. Slight opening between filter insert and housing corrected which should increase filtering efficiency. The correction was necessary as small solid matter (a.o. pieces of rubber gaskets) had been observed to bypass the filter.

- Tap water fed emergency core cooling system flushed and checked. Measured capacity at normally available pressure (6 bar): 24 m^3/hr.

- Main heat exchanger nr. 2 inspected on primary side. Heat exchangers nr. 3 and 4 inspected and cleaned on secondary side. Presence of a piece of concrete (20 x 10 x 4 cm) necessitates internal inspection of secondary cooling section upflow from heat exchanger.

Offgass and ventilation systems

- Outlet valves main ventilation system ("containment" valves) inspected; gaskets and air cylinders replaced.

- Active charcoal filters between Dismantling Cell and offgas system calibrated for iodine retention efficiency. Low values (65 to 75°/o necessitate further testing and possibly early replacement and recalibration.

- Special air venting system (to be used under manual control during"containment" situations) checked for proper control and flow measuring functions. Special procedures written and implemented for operation of the system under reactor hall overpressure situations (e.g. during leakage tests).

Experimental Systems

- Vacuum pumps of beam tubes inspected.

- BWFC high pressure water supply pumps inspected and overhauled.

Miscellaneous Systems

- Leakage check carried out on CO_2 cooling system HB-0 facility (6 l/min. without- and 22 l/min. with CO_2 circulation).
- Renewal of main water supply lines partially completed.
- Check valves in liquid waste discharge systems inspected and replaced where necessary.

b) Instrumentation

The main activities consisted of:

- Flux channels
  Replacement of signal cable and protection hoses on start-up channel 1, period channel 2, safety channels 1 and 3 and on the linear channel. Inspection and

calibration, as far as neceaary of all nuclear channel instrumentation chains.

- Reactor Interlock Systems
  Inspection of interlock relais and switching devices. The main interlock power switch had to be replaced because of mechanical damage of the lock.
  Replacement of the fuse holders of the interlock power supply unit. Damaged contacts in one of the fuse holders had caused interlock power interruptions which led to three reactor scrams.

- Communications Systems
  A start has been made with the installation of signal cables for the extension of the CCTV camera system in the reactor building.

- Process Instrumentation
  - Installation of a digital ΔT measuring instrument which measures the ΔT with an accuracy better than ± 0.5 °/o of the measured values.
  - Installation of new beam tube leak detection instrument.
  - Periodical maintenance and calibration of the process instrumentation.

c) Electrical Installation

The main activities consisted of:

Building and general installations

- Modification of electrical power supply and connections to the movable reactor pool bridge in order to improve drive motor control and to provide power to the two bridge-mounted hoisting tackles.

- Installation of power supply and control provisions for new hoisting tackle located near irradiation capsule servicing platform.

- Adaptation of power supply and interlock systems to reactor hall overhead crane in connection with shortening of crane hut.

- Maintenance and modification of intercommunic-action systems and wall sockets around reactor pool walls.

- Installation of signal leads for new temperature and humidity sensors as required for reactor hall leakage tests.

- Provision of appropriate power supply to
  - new sliding door of the "Hoge Montage Hal",
  - feed pump for reactor hall containment water flushing system,
  - heater unit for reactor hall cooling unit compressor,
  - irradiation facility assembly room.

Experimental facilities

- Improvements to power supplies for BWFC installation,
- Electrical preparations for new HFR-TOP out-pile systems,
- Miscellaneous services to new irradiation experi-ments,
- Installation and adjustment of new valve cabin for pneumatic rabbit system.

d) Standard Experiment Service Installations

The main activities have been centred around the replacement of the obsolete data loggers by new microprocessor-controlled units (Fig. 15) and around the first tests with the central processor "DACOS" (Fig. 16).

A detailed description of "DACOS" has been given in the July-December 1979 Programme Progress Report (Nr. 3731). The system is now capable of handling on-line data. It has extensively been used in preparing the reactor containment building leak tests (see paragr. 2.1.3.5).

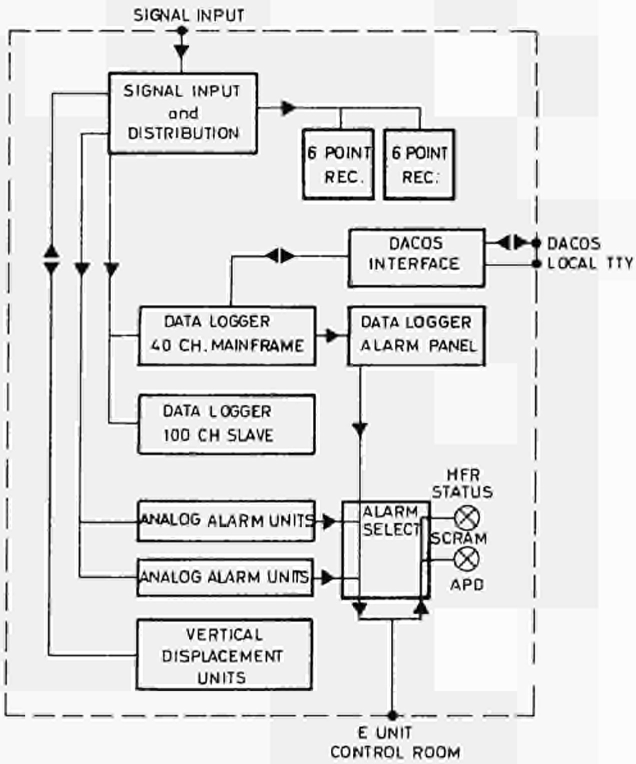


Fig. 15 Micro-processor based data acquisition. Block scheme of the new data logging cabinet.

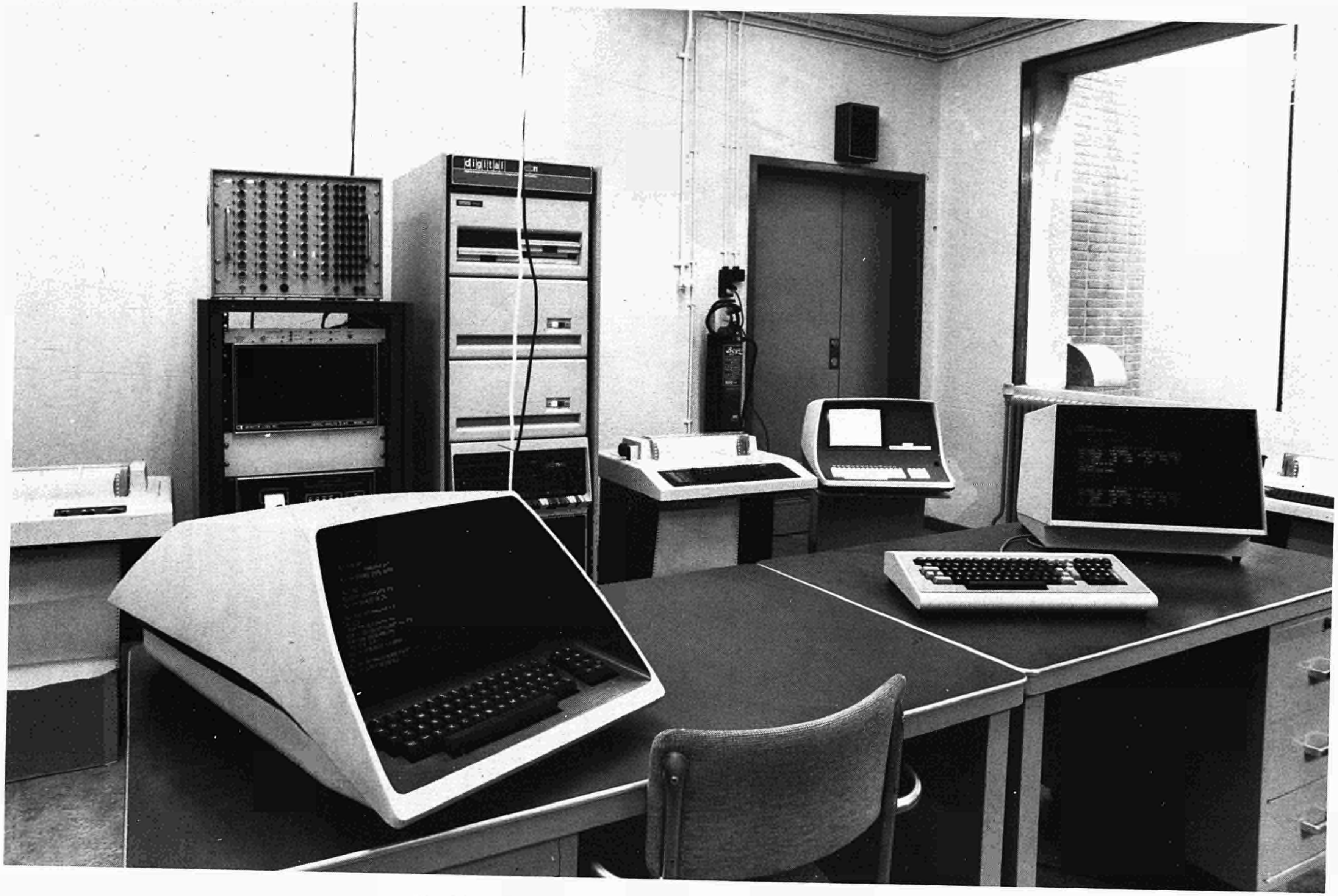


Fig. 16 Data acquisition system (DACOS). Control room.

2.1.3.3 Nuclear and Technical Support and Development

a) Nuclear heating characteristics of HFR core.

In order to identify the defective calorimeters in the "TRAMP" nuclear heating rig, and to investigate possible causes for the erratic behaviour (ref. [5], p. 27) two actions have been taken, i.e.

- design, construction and application of a simple manually operated nuclear heating sensor ("TRAMP-STICK"), by which the TRAMP-results can be checked and qualified,
- neutron radiography of the actual TRAMP calorimeters.

The simplified TRAMPSTICK consists of a long support tube, mounted with two sets of three calorimeters containing heatings sensors of Al, SS and graphite respectively (Figs. 17 and 18). As the calorimeters were of the same design and, as experimentally verified, had the same thermodynamic characteristics as those applied in TRAMP I, the TRAMPSTICK measurements could be used to qualify - or disqualify - the results obtained with TRAMP and thus identify the faulty sensors.
The application of TRAMPSTICK has been a success in the sense that good nuclear heating values have been obtained for several in-core positions. Comparison with earlier "CADO" measurements (graphite samples) show good agreement, while the results obtained for stainless steel and aluminium are the first reliable data for these materials.
In Fig. 19 the vertical distribution of the nuclear heating is given for core positions C5, F8, E3 and G7 at the beginning of a reactor cycle. The primary objective of TRAMPSTICK, i.e. the qualification of the TRAMP data, has nòt been achieved as the agreement between both measurement series is very poor. The preliminary conclusion has been reached that the TRAMP data are - for reasons not yet fully understood - so inconsistent and of such poor quality that they cannot be utilized for the nuclear heating data basis.
A partial explanation of the erratic behaviour of TRAMP has been obtained through neutron radiography of the TRAMP in-pile section. The neutrographs reveal that at least one sensor is filled with water while at least one other is suspect in this respect. A new TRAMP-II rig will now be designed and fabricated applying slightly larger samples and utilizing the same automatic repositioning unit as its predecessor.

b) Accuracy of calculated power distributions and control rod settings in HFR cores

Before the start of each operating cycle the HIP-TEDDI code package is used to provide information on the nuclear conditions in all fuel, reflector and experiment positions. This code package is based upon the diffusion calculation of the neutron flux densities in four broad energy groups in X, Y geometry averaged over the fuel height.
The information is important for the reactor operations team as it provides data, such as control rod insertion depth, shutdown time margin, power distribution and thermal safety margin, relevant to the safe operation of the reactor.
The calculated thermal safety of the reactor is especially dependent on the power distribution and the control rod insertion depth. In order to get an impression of the reliability of the calculated thermal safety of the reactor comparisons have been made between calculated and measured power distributions and control rod insertion depths [6].

As it appears from Table 7 the agreement between the calculated and measured flux density distribution - which is a good measure for the power distribution - is quite satisfactory. The uncertainty of 5°/o in the calculated power density, which is used for the evaluation of the thermal safety margin, seems justified. The control rod settings predicted by HIP-TEDDI deviate only slightly, in most cases 1 to 2 cm (500 pcm) and occasionally 3 to 4 cm (1000 pcm), from the measured ones, see Fig. 20 and Table 5 and 6.

Table 5 Average values and standard deviation of the ratios between calculated and measured thermal flux densities.

core number	averaged over *)	$\left(\frac{\bar{\phi}_c}{\bar{\phi}_m}\right)_{av}$	s **)
524	33 FE	1.008	0.037
524	7 CR	0.983	0.070
524	33 FE + 7 CR	1.004	0.044
525	33 FE	0.994	0.030
525	7 CR	1.031	0.049
525	33 FE + 7 CR	0.999	0.035

*) FE = Fuel Element, CR = Control Rod.
**) Standard deviation in measured thermal flux density, ϕ_m, is 3°/o.

Table 6 Calculated and measured control rod setting in measuring cores 524 and 525.

core number	stage	control rod setting	
		meas.	calc.
524	T = 0, in. Xe, Sm	51.9	51.1
525	T = 0, in. Xe, Sm	48.0	48.1
	T = 0, eq. Xe, in. Sm	60.8	60.6

Fig. 17 In-pile calorimeter TRAMPSTICK. General assembly.

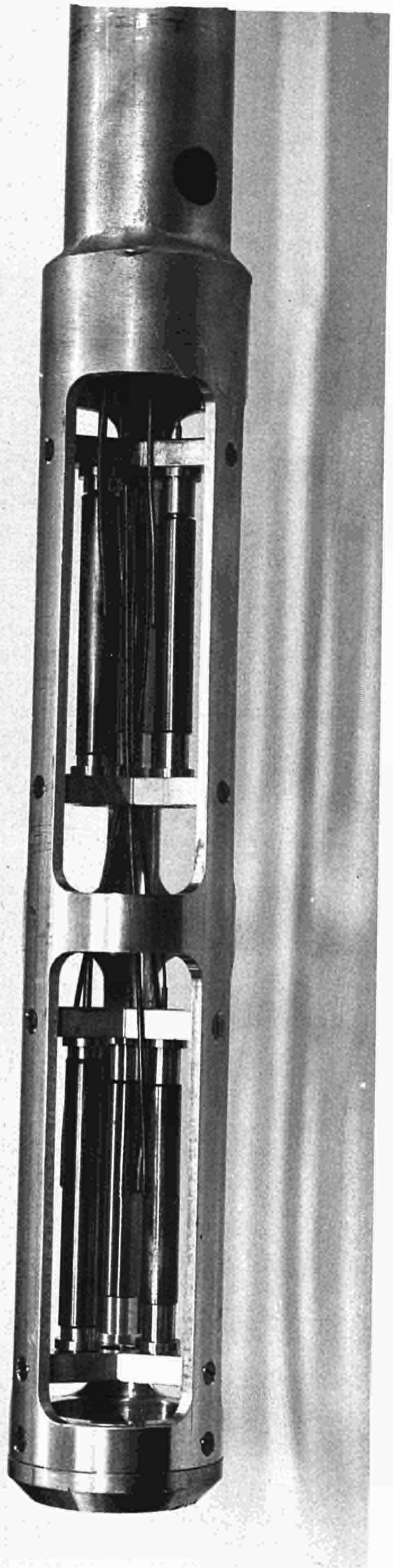


Fig. 18 In-pile calorimeter TRAMPSTICK. Details of the calorimeters.

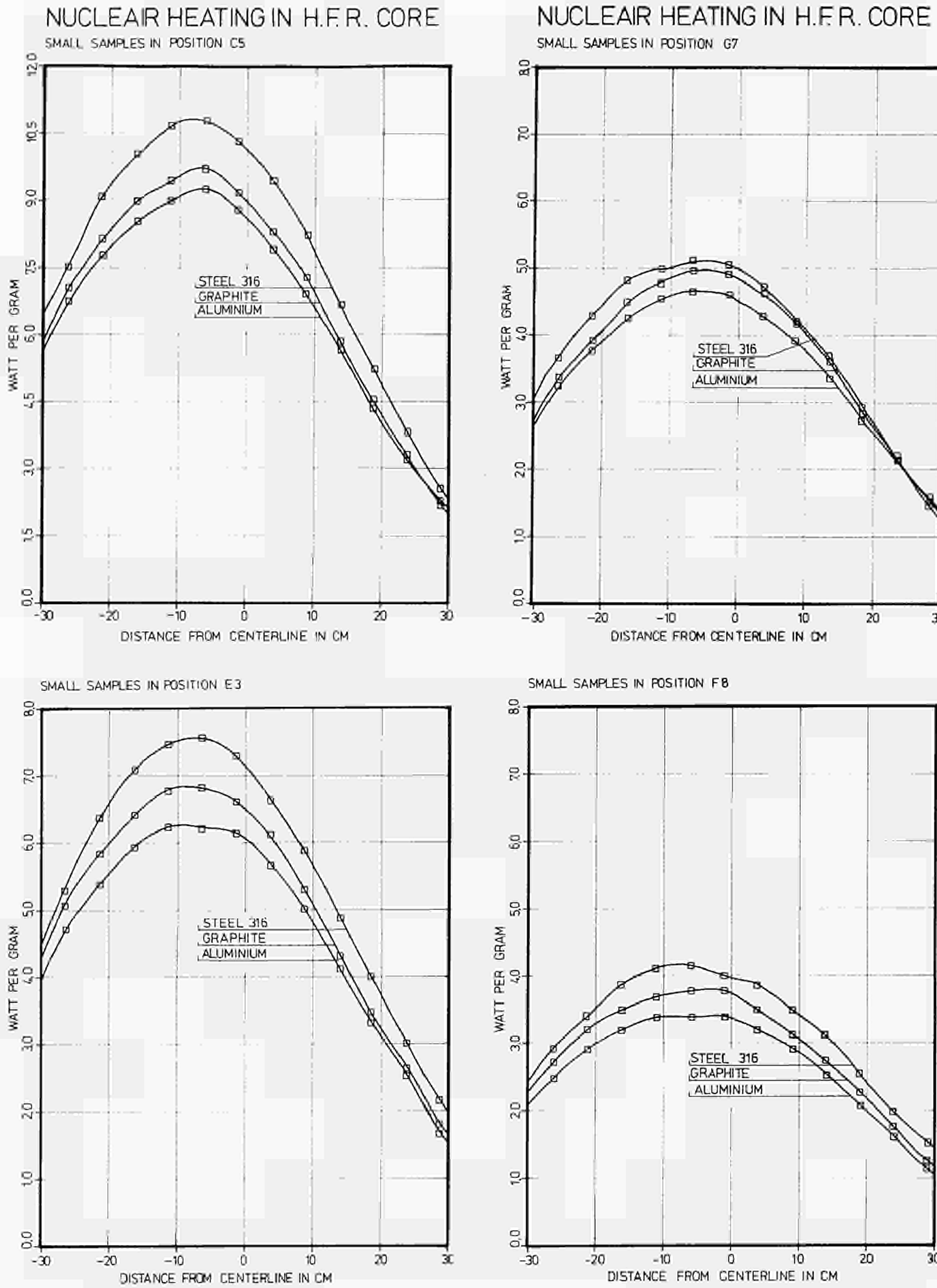


Fig. 19 Nuclear heating measurements in HFR core positions using a small-sample calorimeter ("TRAMPSTICK").

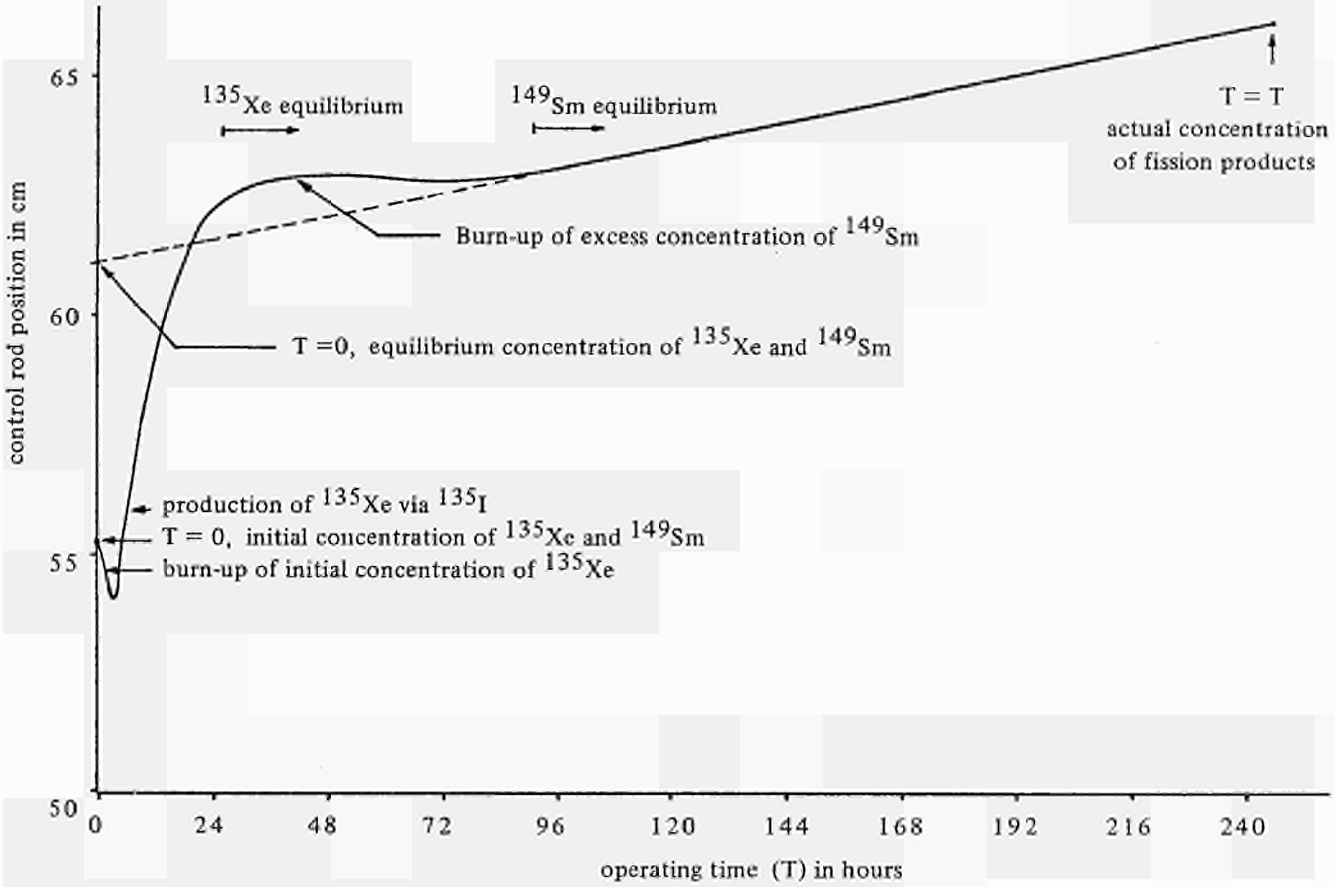


Fig. 20 Typical control rod displacement during the first part of a reactor cycle.

Table 7 Measured and calculated control rod settings in actual cycle cores.

HFR Operating Cycle	operating time in days at 45 MW	measured			calculated		
		T = 0 in. Xe, Sm	T = 0 eq. Xe, Sm	T = end of cycle	T = 0 in. Xe, Sm	T = 0 eq. Xe, Sm	T = end of cycle
78.08.01	6.67	51.7	59.1	60.8	50.9	58.3	59.0
78.08.02	18.53	54.5	56.4	62.8	52.5	55.6	59.0
78.09.01	26.24	48.2	54.2	62.0	46.5	53.0	58.9
78.10.01	0.72	49.5	--	56.2	48.0	55.3	55.3
78.10.02	24.24	50.2	53.9	62.3	49.1	53.9	59.2
78.11.01	26.56	~ 47	55.8	62.4	49.5	58.6	64.8
79.01.01	25.03	49.1	56.5	62.9	49.5	58.4	63.5
79.02.01	24.22	49.9	57.7	65.8	48.2	57.2	61.8
79.03.01/02	4.60	49.5	55.0	55.8	47.7	58.7	59.4
79.03.03/04	18.85	50.5	57.3	60.8	50.6	57.8	60.7
79.04.01	26.06	51.0	58.6	71.3	52.7	58.7	73.7
79.05.01	15.65	51.2	58.6	61.5	52.6	60.4	62.6
79.05.02	9.48	63.0	61.1	65.7	61.9	62.4	64.7
79.06.01	23.61	51.3	57.1	64.9	53.4	60.3	65.8
79.07.01	20.86	53.5	60.0	65.1	50.5	59.3	63.1
79.07.02	2.52	53.0	57.0	57.0	52.1	56.2	56.5
79.08.01	25.64	55.5	57.1	64.4	55.6	57.2	63.0
79.09.01	25.44	50.0	58.4	64.3	49.0	58.0	62.5
79.10.01	7.33	52.0	60.0	61.8	51.3	59.3	60.3
79.10.02	17.41	61.5	61.0	65.5	60.1	59.9	63.2
79.11.01	25.04	51.4	59.4	65.0	51.7	59.4	63.8

In Table 6 a comparison between calculated and measured control rod settings for the two special cores 524 and 525 is shown, while Table 7 gives such a comparison of a series of actual operating cores. The three stages for which the comparisons have been made are graphically indicated in Fig. 20, where a typical control rod moving pattern for an operating cycle is shown.
The comparisons have shown that calculated core reactivity values and control rod positions deviate only slightly from actual values, the largest deviation being in the order of 1000 pcm, corresponding to a difference in control rod position (six rods) of 3 to 4 centimeters.

2.1.3.4 Fuel Cycle

a) Uranium supply

Two export licences have been granted in May, covering 36,6 kg ^{235}U for the 1981 fuel element fabrication. Two new requests have been introduced towards the end of the reporting period.

b) Local fuel element management

During the first half year of 1980, 39 new fuel elements and 14 new control rods have been delivered (see Table 8). Several orders for the manufacture of fuel elements and control rods have been prepared. They all specify the increased uranium contents of 405 g ^{235}U (fuel element) and 280 g ^{235}U (control rod).

c) Spent fuel transports

32 depleted fuel elements and 10 depleted control rods have been transferred for reprocessing. The decay power of this transport was 2093 W. Confirmation has been received from the Savannah River reprocessing plant that blister formation on the side plates has not been observed.
One of the future transports to Savannah River will include 13 spent elements from the HOR research reactor in Delft, The Netherlands. These elements have been shipped to Petten on June 24, 1980, under an agreement with the Delft University (I.R.I.).

d) Fuel element storage

Storage of fuel elements containing 405 g ^{235}U

To ensure safe storage in and transport to the pool of fuel elements with a higher ^{235}U content (405 instead of 390g) a safety analysis has been performed [7]. The cases considered are specified in Table 9.
An infinite double row of racks represents the most reactive in-pool storage pattern and a double row of six elements is representative for the transport lorry, loaded with 12 elements, submerged in water.
The safety analysis has revealed that safe storage (i.e. $k_{eff} < 0.9$) is feasible for the high mass fuel elements provided ^{10}B is present in the side plates. Calculations for slightly burned-up fuel indicate that also when part of, or all of the ^{10}B has been converted the required safety margin against criticality will be maintained in the most reactive storage configuration. Maximum reactivity for irradiated fuel elements is reached after four burn-up steps of 26 irradiation days.

Table 8 Fuel element and control rod movements, 1979/80.

	First half 1979	Second half 1979	First half 1980
Transfer of depleted fuel elements	–	73	32
Transfer of depleted control rods	–	11	10
Average burn-up of transferred fuel elements	–	51°/o	52°/o
Average burn-up of transferred control rods	–	55°/o	51°/o
Delivery of new fuel elements	11	24	39
Delivery of new control rods	–	–	14
New fuel elements available for use at end of half year	26	18	27
New control rods available for use at end of half year	16	8	14
New fuel elements charged to core	36	32	30
New control rods charged to core	8	8	8
Fuel elements depleted	34	22	31
Average burn-up of depleted fuel elements	53°/o	51°/o	52°/o
Control rods depleted	6	7	10
Average burn-up of depleted control rods	52°/o	49°/o	49°/o

Table 9 Values of k_{eff} for various fuel storage patterns.

Case	Initial ^{235}U mass in g	k_{eff} without ^{10}B	k_{eff} with ^{10}B
Infinite double row of racks	400	0.982	0.81
Ibid	450	0.999	0.83
Double row of six elements	450	0.951	0.79

Compact fuel storage

Calculations [8] have been performed to study the feasibility of a compact fuel storage rack for the HFR fuel elements, using BORAL plates as a shielding material to ensure safe storage. The computer codes MICROFLUX, GAM and TEDDI have been applied using Fig. 21 as the calculation model. The reactivity of an infinite storage rack using various BORAL plate thicknesses is shown in Fig. 22. The construction of a compact storage rack is being considered in order to create more space in the storage pool for experimental equipment and container manipulations.

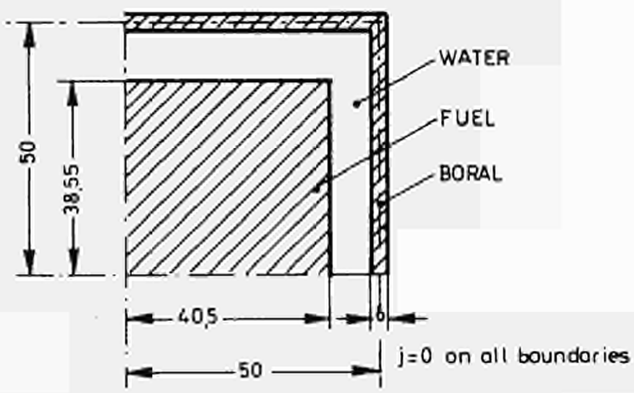


Fig. 21 Compact storage calculation model.

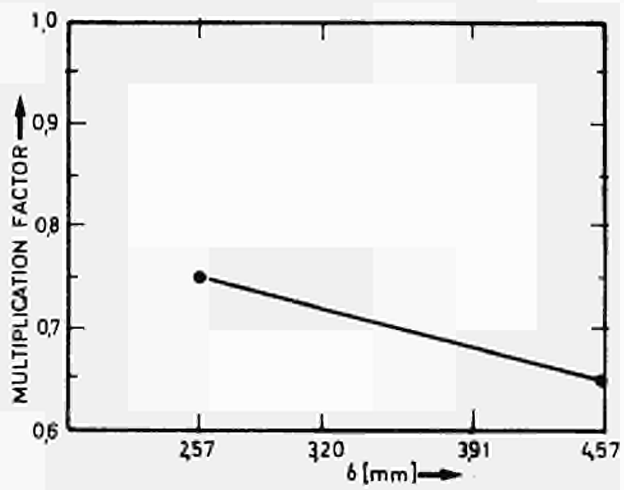


Fig. 22 Compact fuel storage. Reactivity vs. boral plate thickness

e) Fuel development

"LOUISE" - project

Preparation for the test irradiation of low enriched (LEU) MTR type fuel elements continued. Discussions with the fuel manufacturers, who will make LEU fuel elements available for test irradiation in the HFR, lead to final design proposals by the end of this half year. The test elements will be adapted for incorporation of various flux monitoring devices and for improved accessibility for intermediate dimensional checks. Pending delivery by the US of the low-enriched uranium, test element fabrication is now expected to start during the last quarter of 1980. Irradiation in the HFR is foreseen from cycle 81.04 onwards. A draft design and safety report for the test irradiation has almost been completed and will be ready for safety committee reviewal by September 1980.

2.1.3.5 Miscellaneous Tasks

Reactor Containment Building Leakage Rate Tests

- Low pressure leakage test
  On April 9, 1980 the half-yearly leakage rate test of the HFR containment building has been carried out. The overpressure during the measurements was 0.1 bar. Preceeding the actual measurements soap tests were performed at most penetrations. No major leaks were detected. The measurements started at 03.18 h, on April 9 and were continued untill 03.18 h, the next morning.
  The gross leakage rate which could be derived from the measuring data during this 24 h. period was - 33.9 Nm^3/day (see Fig. 23).
  This apparent inflow is caused by various sources (such as cryostats, sweep systems, etc.) inside the containment building. Total gas production has been measured to be 34.4 Nm^3/day. Corrected for this outflow the actual leakage is 0.5 Nm^3/day at 0.1 bar overpressure.
  Extrapolating to the Design Base Accident overpressure of 0.5 bar and accounting for all errors involved in the measurements a total leakage value of 3.0 Nm^3/day with a standard deviation of

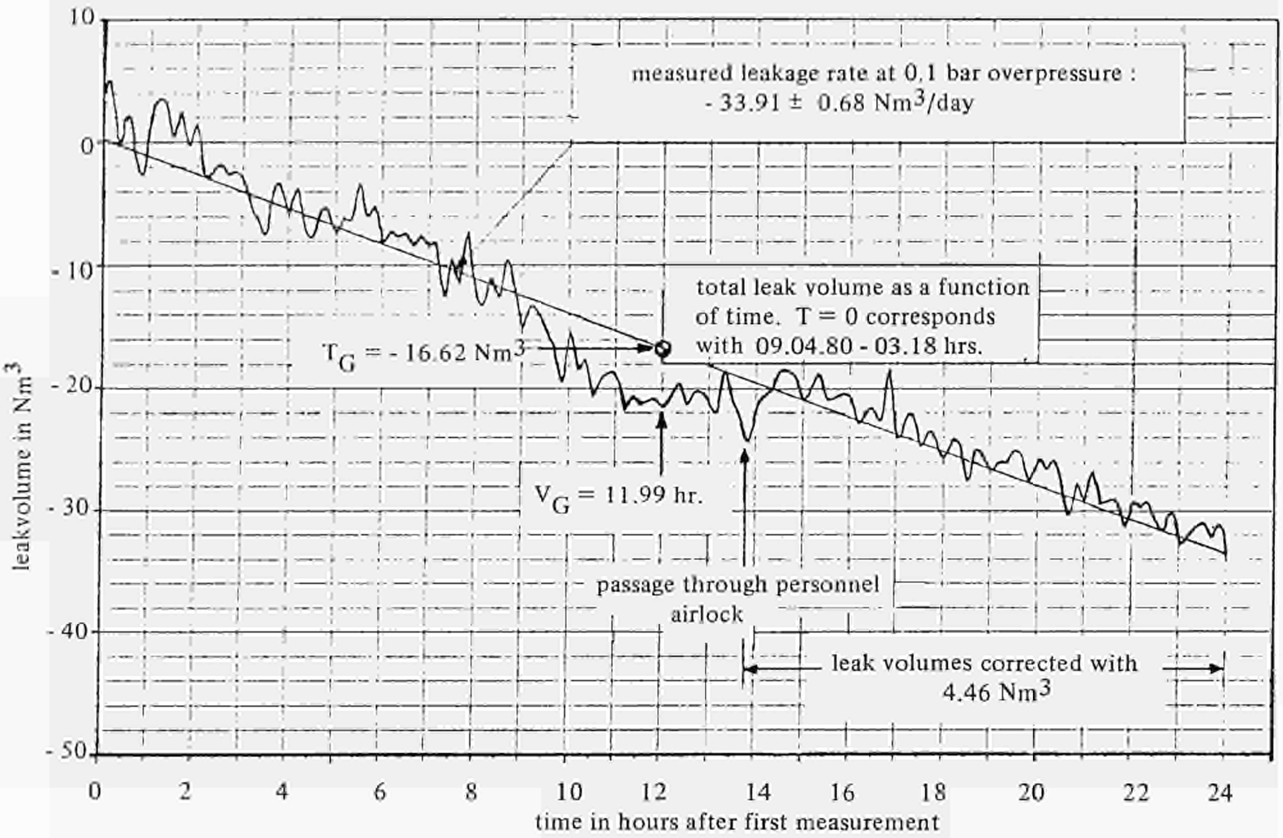


Fig. 23 Reactor containment building, low pressure leak test, April 1980. Leak volume vs. time.

5 Nm^3/day is found. This value is well below the maximum allowable value of 18 Nm^3/day.
During this test a complete updated measuring equipment was applied. In addition the data sampling and evaluation was fully automated. For this purpose the local PDP 11/40 computer was applied (see paragr. 2.1.3.2 "DACOS").

- High pressure leakage test

The measurements described above could be considered as a useful "rehearsal" for the high pressure (0.5 bar) leakage test which was to be carried out during the 1980 summer shutdown period. This test, which is required by the competent licensing authorities to be carried out every four years, was performed already in 1979. Though the measured leakage rates at that time were well within the allowable limits, some doubt was cast upon the reliability of the calibration run (which had a preset leakage rate) which did not give sufficiently consistent results.
Several technical improvements both with respect to the measuring set-up and to the evaluation and assessment methods have been implemented.
Just after the end of this half year (July 1st through 3rd, 1980) a new high-pressure leakage test has been carried out. Preliminary results indicate very low leakage values, generally in agreement with the 1979 results.

2.1.3.6 Users' Services

a) Nuclear feasibility of Al N irradiation

The nuclear feasibility of utilizing the cylindrical plugs in the beryllium reflector elements in the HFR core for the production of ^{14}C by the irradiation of Al N has been studied [9].
Four cases have been analyzed, i.e.
1) utilizing all 24 Be-positions with maximum Al N loading in the plugs,
2) as above, with 50°/o H_2O content of the target plugs,
3) utilizing only eight positions,
4) utilizing non-used in-core irradiation positions.

The results, in terms of achievable ^{14}C activity, for cases 1 and 3 are compiled in Tables 10 and 11. Case 2 yields roughly half the specific activity values as those of case 1, while the analysis of case 4 has revealed that the production capacity of the in-core positions is considerably greater (e.g. position E5 yields 20 Ci per year) than that of the reflector positions.
As far as reactivity penalties are concerned all four options seem acceptable (550 pcm negative reactivity for case 1). Considering the neutron flux density reduction at the beam tube ports only configurations 3 and 4 seem acceptable, as case 1 results in a 34 - 41°/o and case 2 in a 22 - 29°/o decrease.

Table 10 Specific activity of ^{14}C after 290 days irradiation of AlN target plugs in 25 reflector positions of the HFR core.

Core position	Spec. activ. Ci/kg AlN	Core position	Spec. activ. Ci/kg AlN	Core position	Spec. activ. Ci/kg AlN	Core position	Spec. activ. Ci/kg AlN
A1	3.5	B1	4.4	C1	4.8	D1	4.7
E1	4.3	F1	3.7	G1	2.9	H1	2.2
I1	1.4	I2	2.3	I3	3.0	I4	3.5
I5	3.6	I6	3.5	I7	3.0	I8	2.3
I9	1.5	H9	2.3	G9	3.1	F9	3.9
E9	4.5	D9	5.0	C9	5.0	B9	4.6
A9	3.6						

averaged value over positions: 3.5 Ci/kg AlN
total activity: 208.7 Ci

Table 11 Specific activity of ^{14}C after 290 days irradiation of AlN target plugs in 8 reflector positions of the HFR core.

Core position	A1	B1	C1	D1	D9	C9	B9	A9	averaged over positions	total activity
Specific activity Ci/kg AlN	3.4	4.4	4.8	5.2	5.3	4.9	4.5	3.5	4.5	85.9 Ci

As a preliminary conclusion the utilization of non-used in-core positions seems to be the best choice for C^{14} production.

b) Reducing neutron flux gradients in the DUELL experiments (see paragr. 2.2.3.5)

After the first irradiation series had revealed that unacceptable neutron flux gradients existed across the mixed oxide fuel pins, various options to flatten the flux distribution have been analyzed. For the option which has been selected as the best solution to the problem, i.e. the installation of cobalt-steel absorber blocks in the corners facing the core box wall, the results have been presented in Fig. 24 together with the original version of DUELL.

c) Miscellaneous jobs

- In order to evaluate the safety aspects of the vertical repositioning of the in-core sample holder of the D190 "STORY" project (see paragr. 2.2.3.3), reactivity calculations have been carried out for various material compositions of the irradiation insert.
  The study has a.o. led to the conclusion that the replacement of the insert loaded with stainless steel targets with water will lead to a 340 pcm reactivity increase.
- Fission power calculations for the E172 ("CORROX") mixed oxide fuel pins have been carried out with both more and less detailed geometries of the in-pile cross section of the facility as an input. The earlier observed effect that the HIP-TEDDI calculation code will compute higher powers when geometries with small areas and large absorption cross sections are used has been confirmed by this study. The most detailed geometry gave 18°/o to 25°/o higher power ratings than those measured during irradiation.

2.1.3.7 Neutron Metrology Methods Development

a) External dosimetry

The irradiation of the TRIO assembly with experiment E177 and fluence detector sets in vertical channels in the corner of the assembly has been performed during cycle 80.02. During the loading of the external detector sets it turned out that one corner could not be used. A start has been made with counting of the detectors.
The internal detector sets from E177 will arrive in the second half of this year. Then a start can be made with the comparison of the metrology results of internal and external detectors. A detector set comprises: iron, nickel, cobalt and niobium detectors.

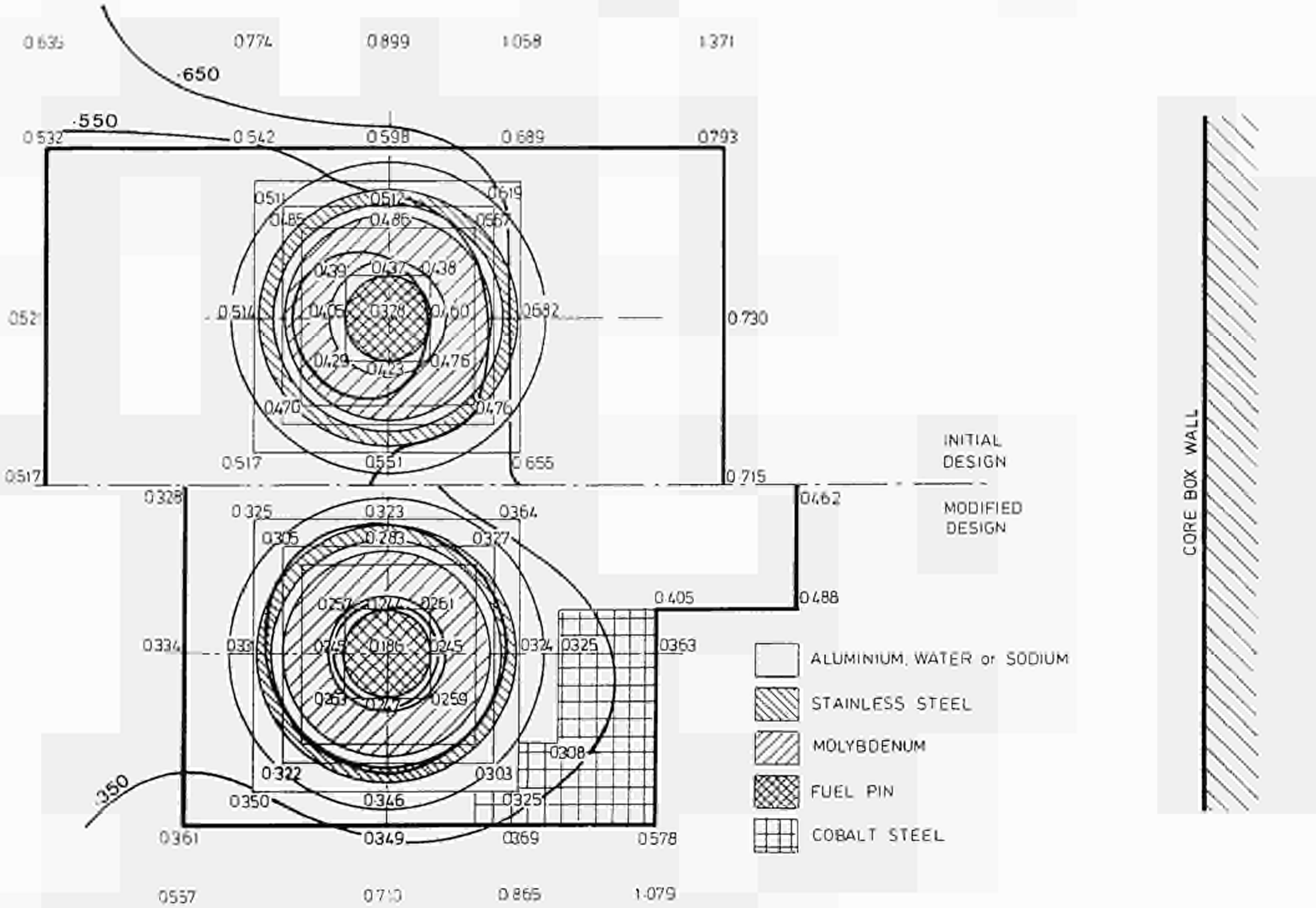


Fig. 24 D173 DUELL. Improved distribution of neutron flux density.

b) Gamma ray spectrometry

For a correct determination of the counting rates of a gamma ray source, the counts lost by the dead time of the total counting equipment have to be taken into account. In a counting arrangement including a multichannel analyser, the counts are corrected only for the dead time by the multichannel analyser.

To investigate the influence of the dead time of the other parts of the counting arrangement, dead time values (τ) have been determined for all local gamma ray spectrometry arrangements.

It is the intention to incorporate a correction for dead time losses in the computer program MBSPROG. This program calculates the activity values at reference time from the counting results obtained from multichannel gamma ray spectrum measurements.

2.1.3.8 HFR Vessel Replacement

a) Reactor Vessel Design ("MARK I" design)

The design work on the new vessel is now well advanced and completion is expected in late summer this year. Contacts have been established with the safety authorities through a series of formal discussion meetings. Agreement has been reached on the classification of the vessel under the ASME III code class 2, although critical areas in the core box region will be treated as typical of a class 1 component.

Draft copies of some sections of the design report and general arrangement and fabrication drawings have been submitted to the safety authorities for comment. Preliminary proposals for in-service inspection have also been submitted.

An extensive international survey has been made into the properties of irradiated aluminium, particularly its embrittlement because this is a key factor in the input to the design calculations. On the question of the choice of material, there is at present a consensus of opinion that the present vessel material ASTM 5154 should be used, mainly because it is felt that any benefit in terms of strength or ease of fabrication to be gained from choosing an alternative are outweighed by the advantage of having the irradiated material of the present vessel as a source of test specimens.

b) Beam Tube Design

A contract has been awarded for the design of the beam tubes and the external Be reflector. Preliminary calculations have shown that the optimum gap between the core box and the beam tube ends is 7,5 mm.

Sufficient cooling for the beam tubes will be provided by enclosing the ends of the tubes with a baffle running parallel to the core box wall and passing pool cooling water through the intervening space.

c) Hydraulic studies

Two dimensional sorting tests have been performed on various inlet configurations of the primary cooling water system. The results of these tests [10] indicate that the preferred design should take the form of a number of full width vanes which will turn the flow from the horizontal direction of the inlet pipes to a vertical flow down through the inlet plenum. A contract has been signed to perform half scale model tests on variations of this baffle design. Tests are expected to start in September/October this year.

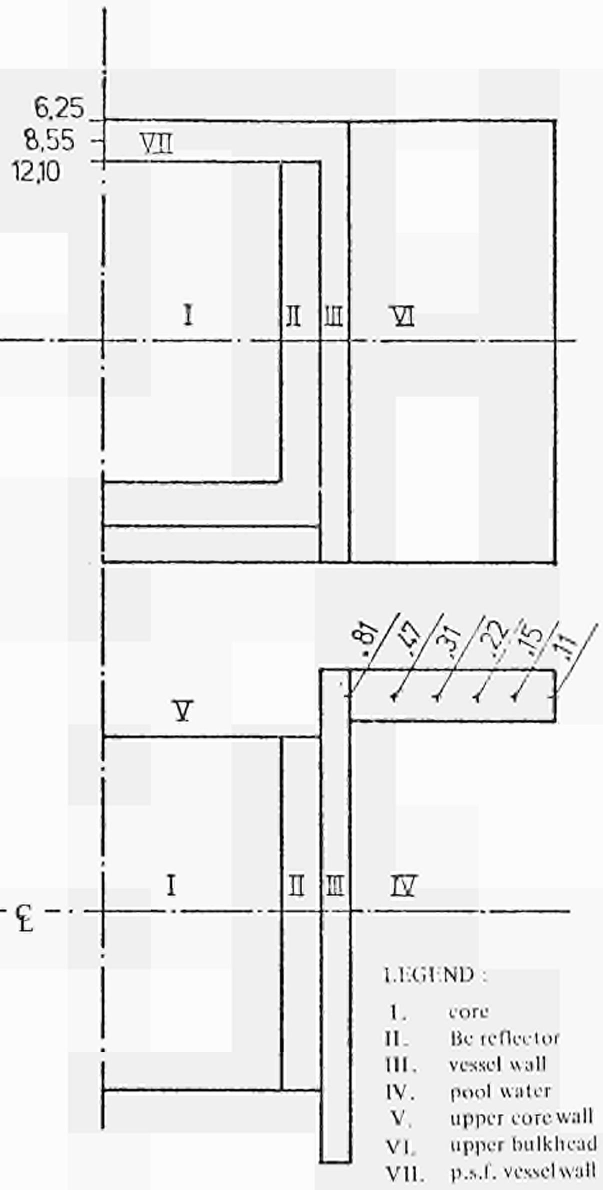


Fig. 25 Gamma heating rates (Wg^{-1}) at 60 MW in HFR vessel walls and upper bulk head.

d) Nuclear heating analysis

An important input to the design calculations of the new reactor vessel consists of the anticipated nuclear heating rates in the core box region. The distribution of the gamma heating has been calculated with the computer code MERCURE-3. The results which refer to a reactor power of 60 MW are presented in Fig. 25 and ref.[11].

e) Vessel replacement scenario

A network planning scheme has been specified and implemented on the CYBER 175 computer [12] for the reactor vessel replacement job. The activities can be devided into three main groups.

The preparation activities, which will take about 310 days, comprise amongst others an inventory of material or equipment to be removed, transported, stored, re-used, the type and size of storage needed, and arrangements with other sites for temperary storage.
In this first issue of the replacement job it is assumed that all beam tube experiment equipment will be removed for ease of work during the last reactor cycle before the actual replacement will take place. This last cycle is also incorporated in the preparation activities.

The removal activities - estimated to last 54 days - can be subdivided into the following smaller groups: removal of internal parts of the vessel, removal of external parts of the vessel, beam tube activities, removal of thermal column, removal of vessel, and mounting preparations. As a result of discussions an alternative handling procedure for the removal of the thermal column will be tested on its feasibility using a real scale test mock-up. The testing will comprise the use of a hardening foam to be inserted into the nose of the thermal column which should enable cutting of this nose while the complete assembly is still in its original position. When this procedure can be applied it will considerably reduce radiation doses during handling. Based upon comments received on the planning report the handling of the beam tube experiments will be subject to an extensive analysis of alternative working methods and their associated influences on the total project.

The installation activities - estimated to last 63 days - comprise the installation of the new vessel, alignment of the vessel to the existing control rod drive mechanisms (which will be retained without any change), mounting and alignment of internal parts of the vessel, installation of external parts of the vessel, mounting of new beam tubes and nuclear channels, flow testing of the primary system using dummy fuel elements and experiments, flow testing of the pool circulation, installation of the first new core and testing at low reactor power, measurements of all characteristics of interest, installation of experimental facilities. Once the operational tests have been completed the beam tube experiment plugs can be reinserted and the reactor is ready for actual start-up at full power. During the first reactor cycle the build-up of the beam tube experimental equipment will take place.

f) Evaluation of an alternative ("MARK II") vessel design

The present ("MARK I") design of the new reactor vessel has been based upon the recommendation of a joint JRC-ECN working group which had combined all operational, technical and experimental requirements into a functional specification report. The basic requirements,

- continued operational reliability,
- improved versatility for irradiation experiments,
- improved operational performance,
- improved safety,
- acceptable replacement economy,

have been met by the Mk I design. The Mk I vessel has clear safety improvements in comparison to the present vessel, a.o. since all critical welds will be inspectable in service and since the design will conform with a well established code.

It has been realized, however, that the Mk I design has developed into a rather complex structure. This is due mainly to the requirements for flat walls above the east and west pool side facilities and to the requirements to retain the current grid bar design. In view of the fabrication and cost implications of the present design it was considered worthwile to evaluate an alternative approach to the design. Instead of designing the vessel to to a predetermined set of experimental and operation requirements, the most simple core envelope (see Figs. 26, 27, 28) should be chosen, (i.e. a cylinder) and the question to which extent the basic operational and experimental requirements could be met should then be evaluated.

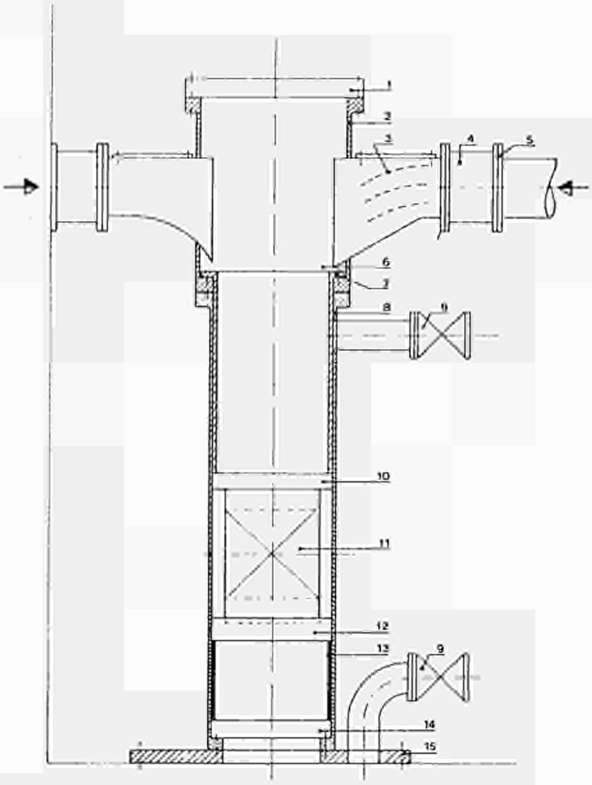


Fig. 26 Cross section through the Mk II vessel

A report [13] has now been submitted for further discussion, in which the relative merits of the two vessel designs are compared mainly on the basis of a technical appraisal.

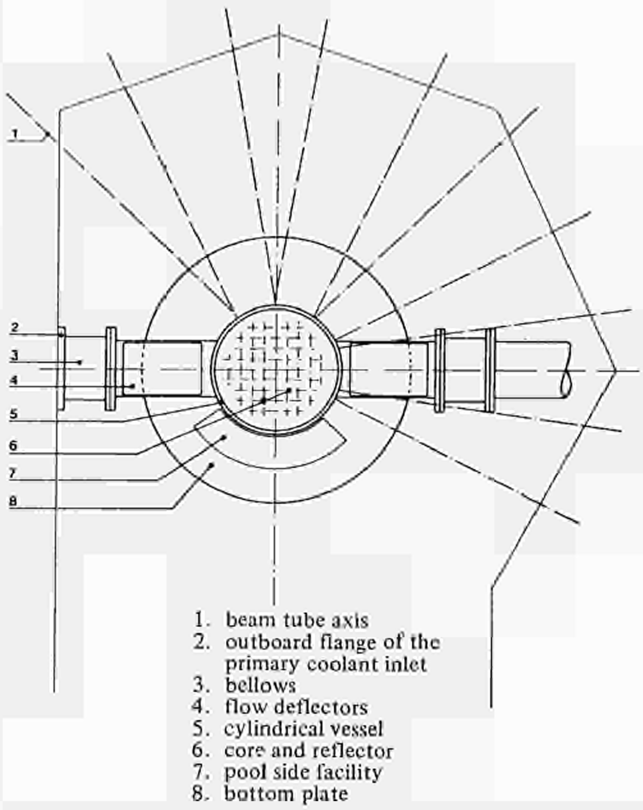


Fig. 27 Cross section through the Mk II vessel and pool walls.

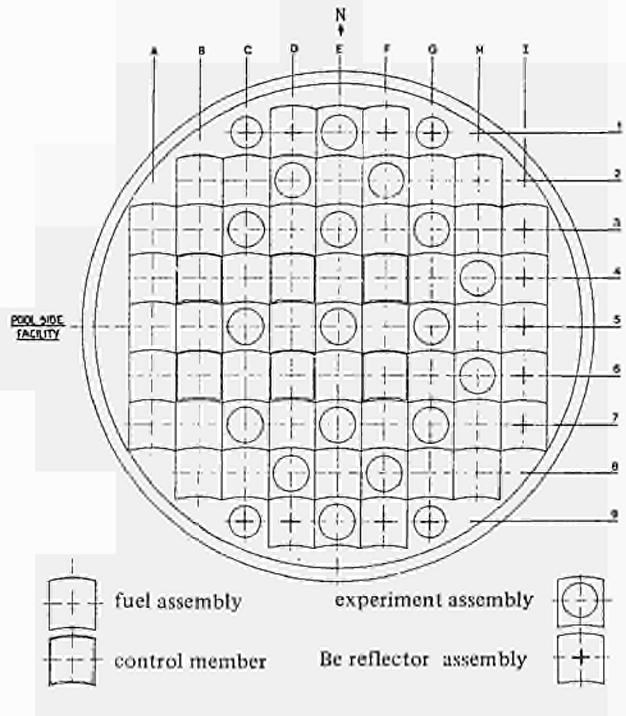


Fig. 28 Possible core configuration in the Mk II vessel.

g) New neutron flux channels

New configurations and new methods for support and guidance of the neutron flux channels around the HFR have been investigated in connection with the new vessel [14]. The present set-up, i.e. all channels being located on the east side of the core, cannot be utilized since that region might be occupied by the second pool side irradiation facility.

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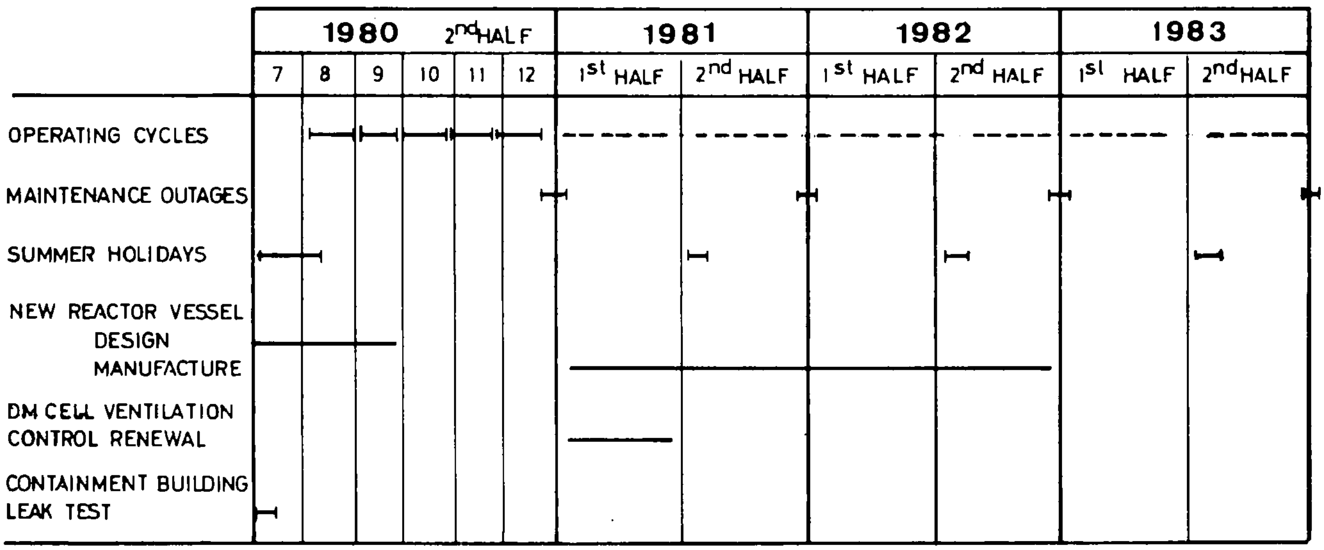
N.B. Only refs. 1 and 3 are available in open literature.

Fig. 29

REACTOR OPERATION
SURVEY OF FUTURE ACTIVITIES



2.2 Reactor Utilization

2.2.1 Objectives

The reactor can be considered as a neutron source for solid state physics (diffraction), nuclear physics (n,γ reactions), radioisotope production and activation analysis (activation), fissile material testing (fissions), and structural material testing (radiation damage).
The work in and around HFR Petten is oriented towards the conception and implementation of experiments using the above-mentioned interactions of neutrons with matter.
The key objectives, expressed by the number of experiments, for the reporting period have been, according to the planning of December 5, 1979:

- start of new irradiations 24
- continued irradiation throughout the six months 3
- unloading after end of irradiation 21
- short term transient irradiations 15
- 8 to 9 horizontal beam tubes in permanent use,
- 3 isotope facilities in permanent use,
- 6 facilities in intermittent operation.

2.2.2 Methods

Technological irradiation projects come into existence by request of experimenters (JRC or external organization). They then pass through the following main stages:

- Design study (feasibility study)
- Detail design and calculations
- Safety analysis and assessment
- Machining, purchase of material, instrumentation . .
- Assembly and testing
- Commissioning, loading and connection in HFR
- Irradiation and surveillance
- Unloading, dismantling
- Post irradiation examens (PIE)
- Reporting.

More detailed project management schemes have been elaborated, featuring about 100 steps per project. Considering that about 30 irradiation projects are handled simultaneously, one can easily judge the work volume involved. It has turned out during the years that the reactor occupation by irradiation experiments is limited by the available staff rather than by experimenters' request.
For non-energetic applications (solid state and nuclear physics experiments, radioisotope production and activation analysis) the project stages are different from above-mentioned scenarios. Usually, fixed installations and long-term facilities are used (which do undergo the complete development cycle) in which the individual target can be irradiated with a minimum amount of technical and administrative preparation.

2.2.3 Results

2.2.3.1 Graphite

A large number of graphite samples has been irradiated since 1962. The HFR graphite irradiation programme supplies the necessary design base for the nuclear process heat and the direct cycle concepts of the High Temperature Reactor (HTR).
The irradiation capsules contain unstressed samples (fundamental properties programme) or creep specimens under tension or compression. They are irradiated in three to four fluence steps, with intermediate measurement of their changed physical properties. For the reflector graphite material, irradiation temperatures range between 300 °C and 1100 °C, the neutron fluences will reach 2×10^{22} cm^{-2} (EDN*) for the most exposed samples.

D 85 Series (Fundamental properties programme), see Table 12

D 85-23 (300 °C): After the withdrawal of the 500 °C carrier in April, 1978 and the 400 °C carrier in September, 1978, the remaining 300 °C TRIO "leg" continued its very stable and accurate performance during the whole reporting period. Its time schedule, originally foreseeing an end of the irradiation after one year, has been changed several times. Actually the experiment is planned until September, 1980. This would be 29 reactor cycles or about 725 days.

D 85-25 (400 °C): After 348 days of successful operation the experiment was finished, according to its schedule, on April 7th. Throughout the whole period the recorded temperatures were within ± 15 °C from the nominal 400 °C.
In April the sample carrier was dismantled and the samples transferred to intermediate measuring. They will continue their irradiation programme as D85-43, later this year.

D 85-26 (300 °C): After the end of the irradiation in December 1979, the sample carrier was unloaded in January, 1980, and the specimens transferred to examination and measurement. These samples have come back in the meantime and are again being irradiated under no. D85-37.

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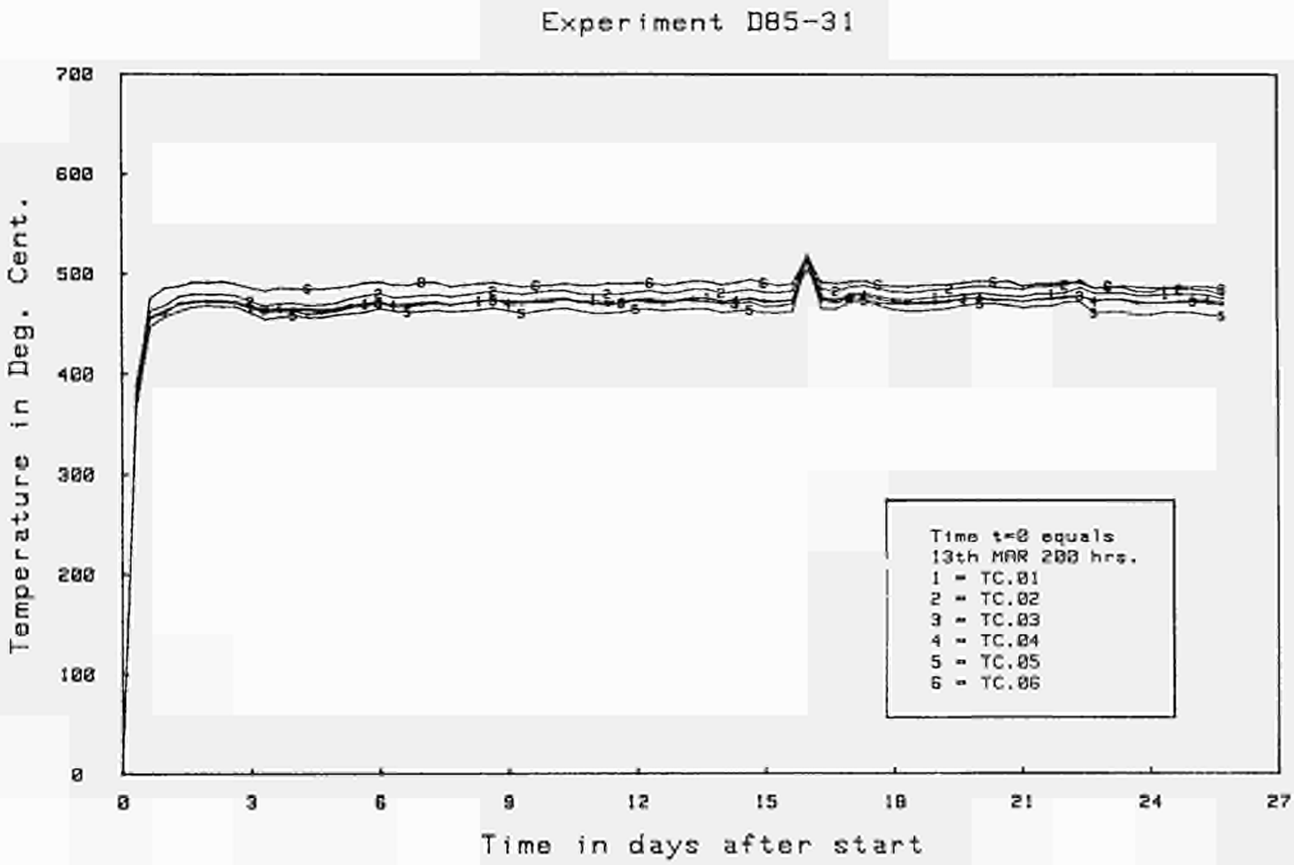
* EDN, Equivalent DIDO Nickel, is a traditional neutron fluence unit for graphite irradiation testing. For HFR core positions, $\phi_{EDN} \approx \phi_{>1MeV}$ (see report EUR 5700, 1979/80 edition, pag. 20 and 50)



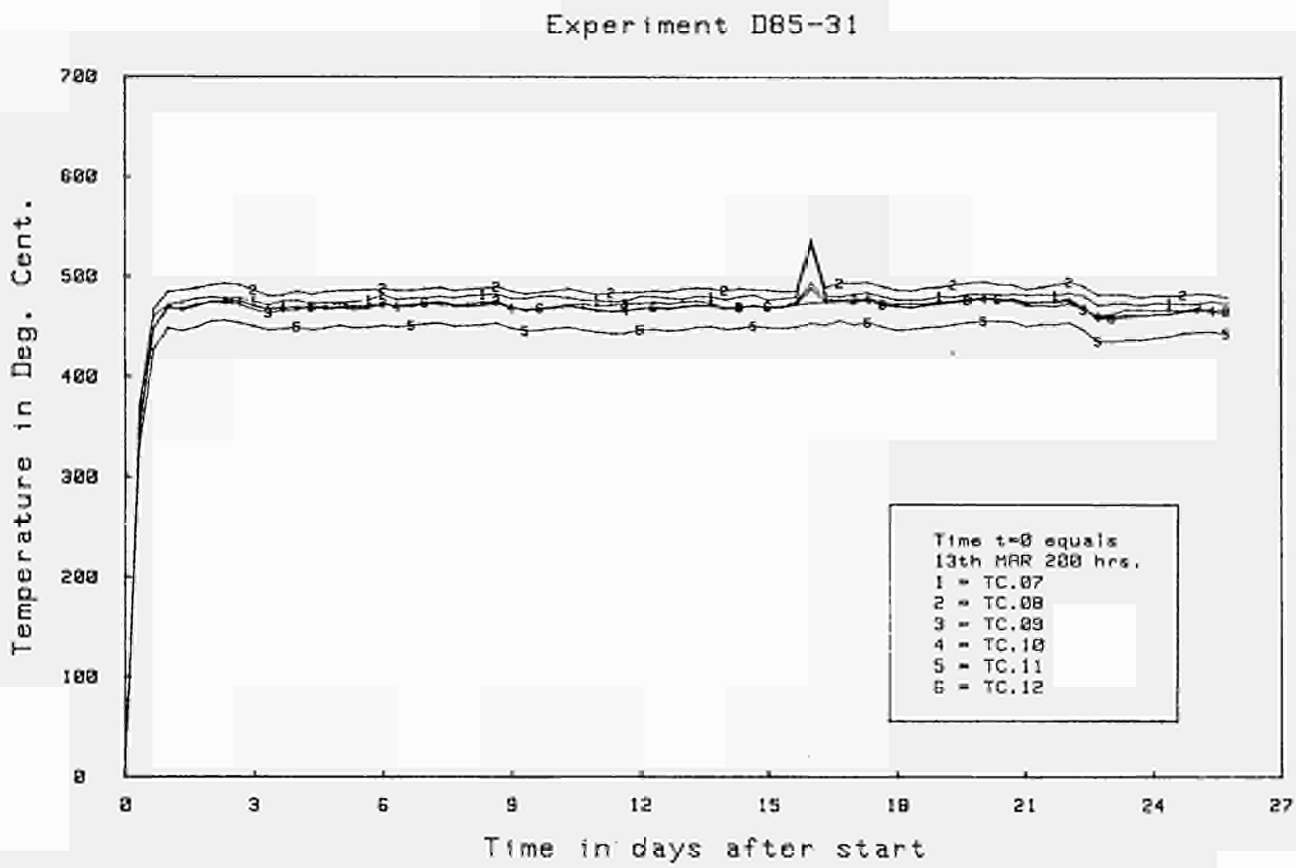


Fig. 30 Graphite irradiation specimen carrier D085-31. Temperature histogram during HFR cycle 80.03.

D85-31 (500°C): The irradiation was finished after 348 days on April 7th, as scheduled. Fig. 30 shows its temperature vs. time behaviour during the last irradiation cycle.
The sample holder was dismantled and the samples transferred for measurements. Later this year, they will continue their irradiation as D85-42, whose start-up is scheduled for October, 1980. Throughout the 14 cycles of reactor operation, none of the 12 thermocouples of D85-31 failed.

D85-32 (500°C): was the follow-up experiment of D85-23/500. It was completely identical to D85-31, concerning design, performed work, start-up and irradiation duration, lasting until cycle 80-03. Its performance during the operation was quite successful, all temperatures were within the specified limits, operation was smooth and unproblematic. It has together with D85-23, accumulated a total fluence of ~ 1.2×10^{22} cm^{-2} EDN.
In April it was dismantled and the samples sent to intermediate measurement, following which they will be irradiated as D85-41, scheduled to start-up in October, 1980.

D85-33 (400°C): continues the irradiation of the samples that have been irradiated in D85-23/400. The design is that of D85-25, i.e. an aluminium rod with holes containing about 130 graphite samples.
After the problems described previously, the new sample holder continued performing very well during the whole reporting period. The time schedule of D85-33 was slightly changed, instead of finishing after cycle 80-06 having completed 13 reactor cycles, it will continue until September 1980, aiming at 14 cycles or 350 irradiation days.
The total neutron fluence (EDN) will then be 8.5×10^{21} cm^{-2} or including D85-23/400, 1.45×10^{22} cm^{-2}.

D85-34 (600°C): is the continuation of D85-24/600. In design it is identical to D85-31/32; its irradiation history is similar to that of D85-33. It started in cycle 79-03.
Probably due to the use of nitrogen as a regulating gas almost all of its twelve thermocouples were lost. The rig was withdrawn and dismantled after 2 cycles. The samples were recovered and continue their irradiation, since cycle 79-07 in a new sample carrier, identical to the previous one.
During the reporting period it operated without any problems and with satisfactory temperature results. It is scheduled to end after cycle 80-07. The total accumulated neutron dose will then be ~ 8×10^{21} cm^{-2}, or including D85-24/600, ~ 1.3×10^{22} cm^{-2} EDN.

D85-35 (750°C) is the replacement of D85-26/750, which did not reach the specifications. It is now a completely new design, based on the principles of D85-31 (isotropic graphite drums, stainless steel carrier tube), but with a different geometry due to other sample sizes and temperature range.
After its start-up in cycle 79-03 and an interruption of two cycles due to HFR overload, it continued its irradiation from cycle 79-07 until the end of the year. At its scheduled termination after cycle 79-11, it had reached a neutron dose of ~ 1.1×10^{22} cm^{-2} EDN on the samples, which previously had been irradiated in D85-30, -29, -27 and -26.
During the reporting period the specimen carrier has been dismantled and the samples sent to KFA Jülich for examination. Since June 6th, they are again under irradiation with the experiment D85-36.

D85-36 (750°C): Its design is slightly changed, but in principle similar to type 33, i.e. D85-35.
The obtained temperature accuracy is very good, see Fig. 31.
The experiment schedule foresees 10 reactor cycles or about 250 days, corresponding to 5×10^{21} cm^{-2} EDN. Including the previous irradiation, the samples will reach a maximum fluence of 1.6×10^{22} cm^{-2} EDN.

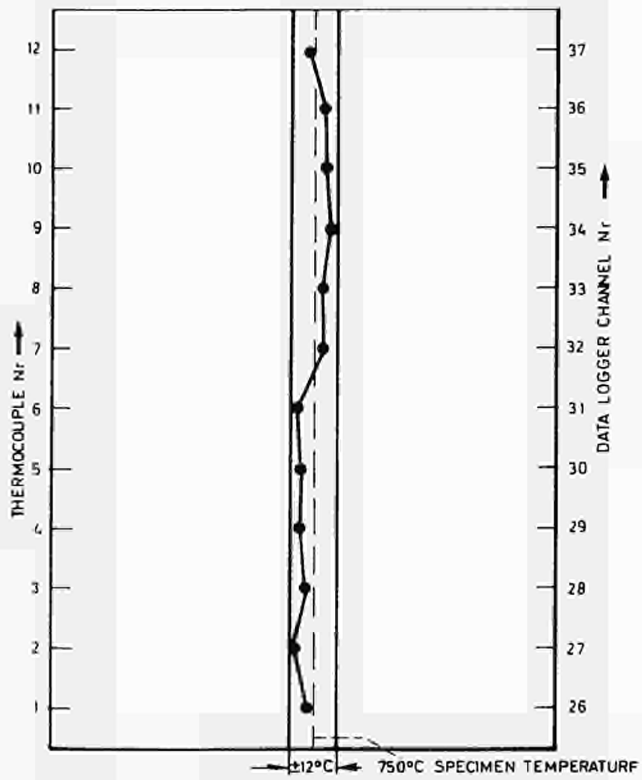


Fig. 31 Graphite irradiation specimen carrier D085-36. Axial temperature distribution on June 19, 1980.

D85-37 (300^{o}C): started together with D85-36 in position C7 on June, 6th. It contains most of the samples irradiated in D85-26 which, after the scheduled 10 cycles, will have accumulated ~ 1.9 x 10^{22} cm^{-2} EDN.
The design of this new sample holder type 15 is very similar to type 12 (= D85-26), the main difference consisting in the partly different sample sizes and the corresponding holes in the all-aluminium carrier.

D85-18/19/20
are the first experiments in a new series of graphite irradiations, in which different types of matrix graphite are irradiated, at higher temperatures, to relatively moderate neutron fluences. 3.5 x 10^{21} cm^{-2} EDN will be accumulated at the end of two irradiation steps.

D85-18 (700^{o}C) will be irradiated during three cycles. For the new Trio capsule with increased diameter (31.5 mm instead of 29 mm, see paragr. 2.3.3.7 and Fig. 83 of this report) a new sample holder was designed, using the existing type 33 (D85-35), but adapting it to the different thimble and sample dimensions. The new standard type was called 331.
During the reporting time, the design work was finished and manufacture started. The present planning schedules start-up of D85-18 in September, 1980.

D85-19 (900^{o}C) is in conception, basic design and work schedule similar to D85-18. The sample holder (type 341) covers the temperature range between 850 and 1050^{o}C.

D85-20 (1100^{o}C) will be irradiated together with D85-18 and -19, in the new Trio 131. The sample carrier (type 351) has been calculated for the temperature range 1050 to 1250^{o}C. It is an all-metal design, made from Nb and TZM. In the 1100^{o}c version it will be equipped with K type thermocouples, sheathed with Nb. At higher temperatures, W/Re thermocouples will be used.
During the reporting period the drawing work was completed, and the fabrication of components started.

Table 12 Fundamental properties graphite irradiation programme (D 85). 1979/81 survey.

Experiment nr. D 85-	Irradiation Period	Irradiation Temperature [^{o}C]	Specimen carrier type
23	Dec. 77 - Sept. 80	300 (400 - 500)	Al - (Al - Cu)
25	Jan. 79 - Apr. 80	400	Al (Type 11)
26	Aug. 78 - Dec. 79	300 - (750)	Al (Type 12) - Cu/S.S.
31	Jan. 79 - Apr. 80	500	Graphite / S.S. (Type 21)
32	Jan. 79 - Apr. 80	500	Type 21
33	Mar. 79 - Sept. 80	400	Type 11
34	Mar. 79 - Sept. 80	600	Type 21
35	Mar. 79 - Dec. 79	750	Graphite / S.S. (Type 33)
36	Jun. 80 - May 81	750	Type 36 (Graphite / S.S.)
37	Jun. 80 - May 81	300	Type 15 (all Al)
38	Nov. 80 - Oct. 81	300	Type 14 (all Al)
39		400	Type 14
40		600	Type 21
41	Oct. 80 - Sept. 81	500	Type 21
42		500	Type 21
43		400	Type 14
18	Sept. 80 - Nov. 80	700	Type 331 (Graphite / S.S.)
19		900	Type 341 (Graphite / TZM)
20		1100	Type 351 (Nb / TZM)

D186 Originally called D85-20, is a new irradiation project on which design studies have been commenced in 1978. Two large graphite samples (120 and 240 mm long, 60 mm Ø), together with a number of compact tensile specimens, shall accumulate in three irradiation stages, a total fast neutron fluence of 2×10^{22} cm^{-2}, simulating reflector working conditions in a PNP reactor. The irradiation temperature is 750°C.
Different design studies and a number of thermal calculations have been made and continued, three irradiation proposals have been written, foreseeing the first irradiation to start in 1981, with the third stage reaching into 1985.
Final decisions concerning the experiment specifications are still pending.

In-Pile Graphite Creep Studies

D156 DISCREET
The top and upper side reflector graphite of the process heat reactor experiences high neutron fluences at relatively low temperatures. Irradiation creep studies on this material are being performed in tension at 300°C and 500°C (also 900°C in the future) and in compression at 500°C. Columns of samples are irradiated in Trio facilities and creep measurements are taken at intervals out-of-pile. The series includes high flux level experiments, low flux level experiments and long term creep studies.

D156.00 Series High Flux 300°C and 500°C tensile, 500°C compressive
Dismantling of the 156-04 rigs was completed in February and dimensional measurements taken on all samples except one 300°C specimen which fractured during hot cell handling. The results of the experiment were presented at the Carbon '80 conference in Baden-Baden [1].
The fluence reached in this irradiation far exceeds that of any other graphite creep experiment, and the principle observation is that the reduction in creep rate at high fluences is much stronger than had been anticipated (Figs. 32 a, b, c).
The irradiation samples were successfully re-encapsulated in new sample holders (D156-05) and irradiation commenced in cycle 80-06 with all temperatures within the specified limits. Components have been ordered for the follow-up experiment D156-06.

D156-10 Series Long Term Creep Studies static and 500°C tensile
D156-11 (unstressed) has been transferred to a new TRIO thimble and will continue irradiation when space is available. It is now proposed to irradiate the specimens up to the zero point, i.e. to the point where the graphite has returned to its unirradiated dimensions, and then to continue as a creep experiment. The zero point dose is in the region of 17×10^{21} cm^{-2} EDN (approx. 1100 irradiation days).
D156-12 (stressed) was dismantled in March and dimensional measurements taken on all samples. The results of the experiment have been included in the paper on 156-04 [1]. Re-irradiation of these specimens will restart in cycle 80-07 as 156-13.
Components have been ordered for the follow-up experiment D156-14.

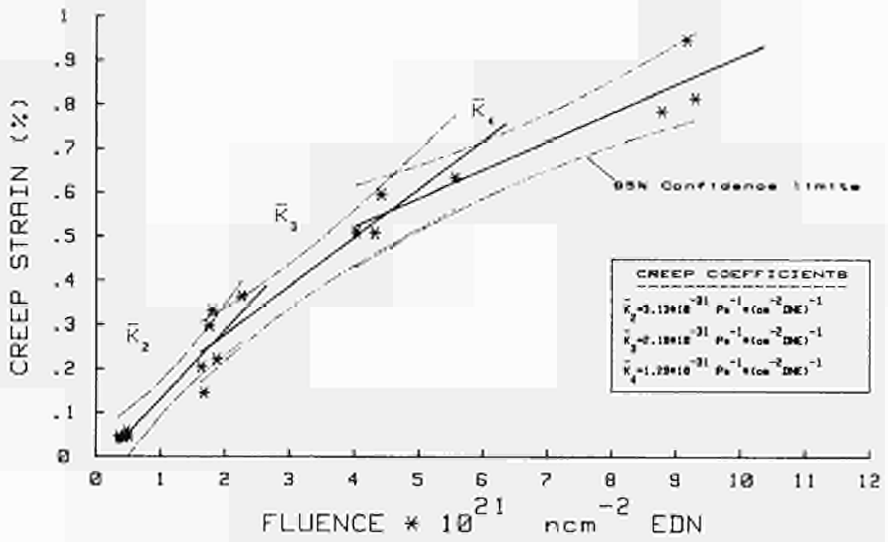
D156-20 Series Low Flux Level, 300°C, 500°C, 500°C tensile
Irradiation of D156-20 was completed after cycle 80-01. The three sample holders have been dismantled and dimensional measurements taken. At this stage of the creep curve it is too early to judge whether or not the flux level has influenced the creep rate.
Re-irradiation of these samples will continue as 156-21 starting in cycle 80-07.
Components have been ordered for the follow-up experiment D156-22.

D156-30 Preliminary specifications have been given for a new creep experiment starting in 1982 on bottom reflector material.
Irradiation is planned for three tensile sample columns operating at 900°C.

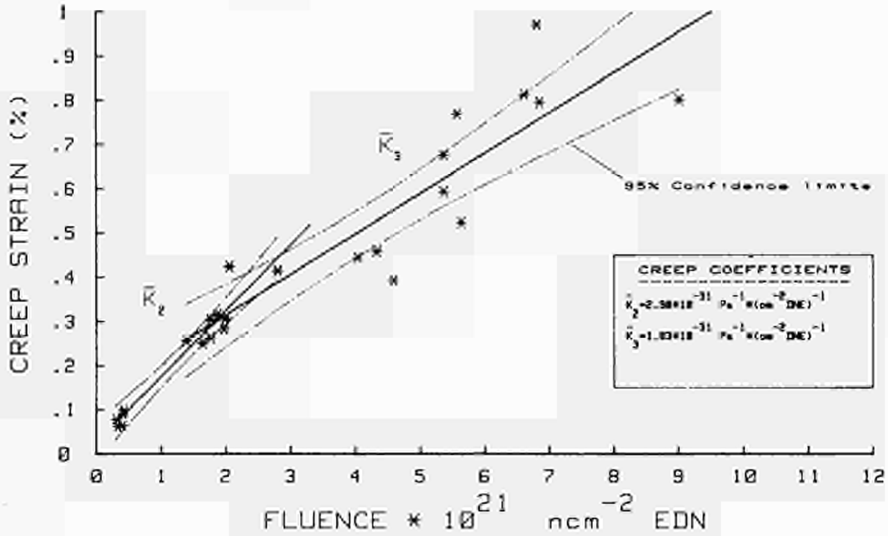
D166 CRIMP Graphite Creep Experiment
An irradiation creep experiment is being performed on samples of H451 graphite in a rig with continuous strain registration. The graphite is a reference material for the fuel blocks and replaceable reflector of the Gulf General Atomic designed HTR.
The target irradiation dose is 7×10^{21} cm^{-2} EDN at 850° - 900°C.

D166-01 Although this experiment was terminated after fracture of the sample, sufficient data was collected to enable the creep constant to be established. The results from this experiment and the results from the previous irradiation of this sample (R135) were incorporated into a paper presented to the Carbon '80 conference at Baden-Baden [2].

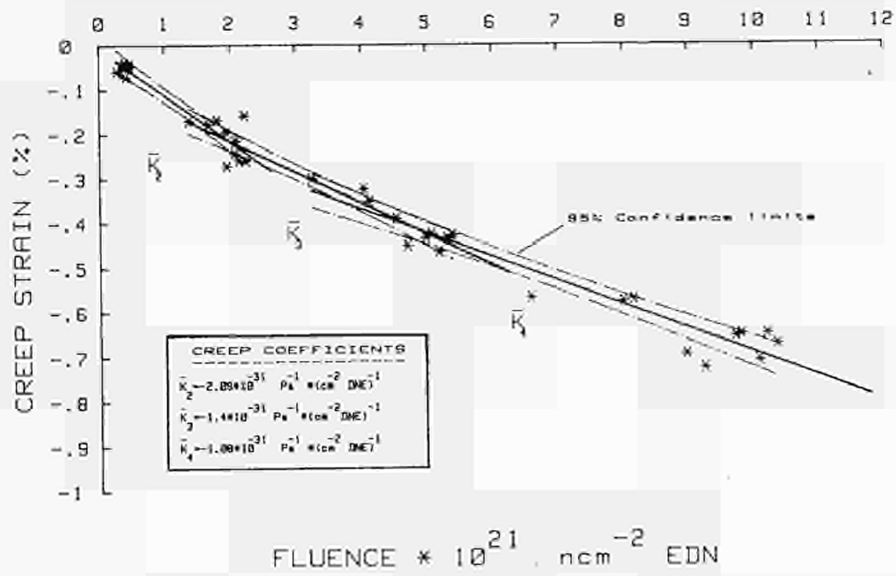
D166-02 Assembly of D166-02 was completed in March (Fig. 33) and reception tests were carried out over the following weeks.
Measurement of the Young's Modulus of the irradiation sample in the creep rig was in good agreement with the value previously found in tests in an "Instron" machine.



a) tensile, 300°C
axial direction



b) tensile, 500°C
axial direction



c) compressive, 500°C
axial direction

Fig. 32 High flux graphite creep series D 156. Creep strain vs. fluence by February 1980.

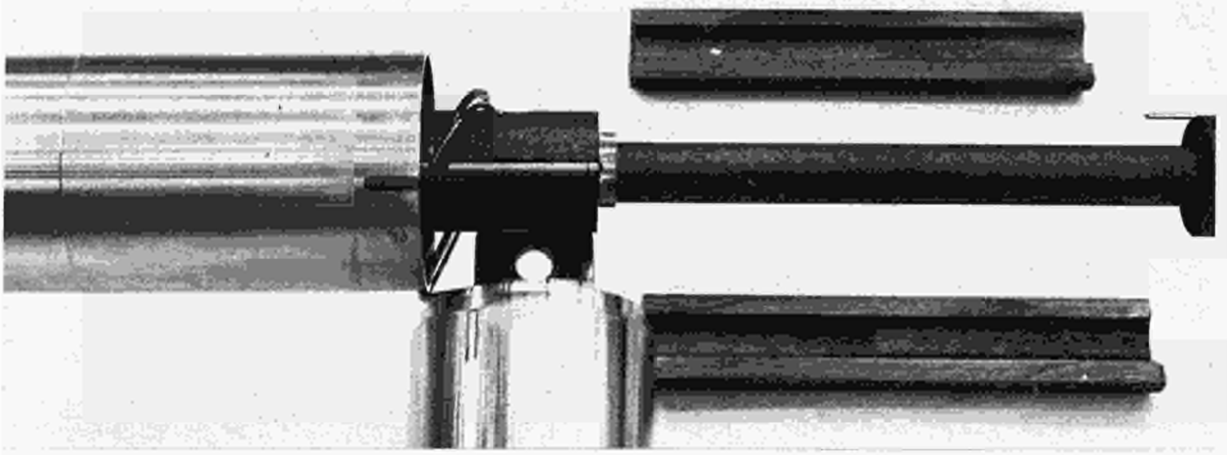


Fig. 33 D166-02 (H 451) Creep experiment. Irradiation sample, half shells and measuring sensor during assembly.

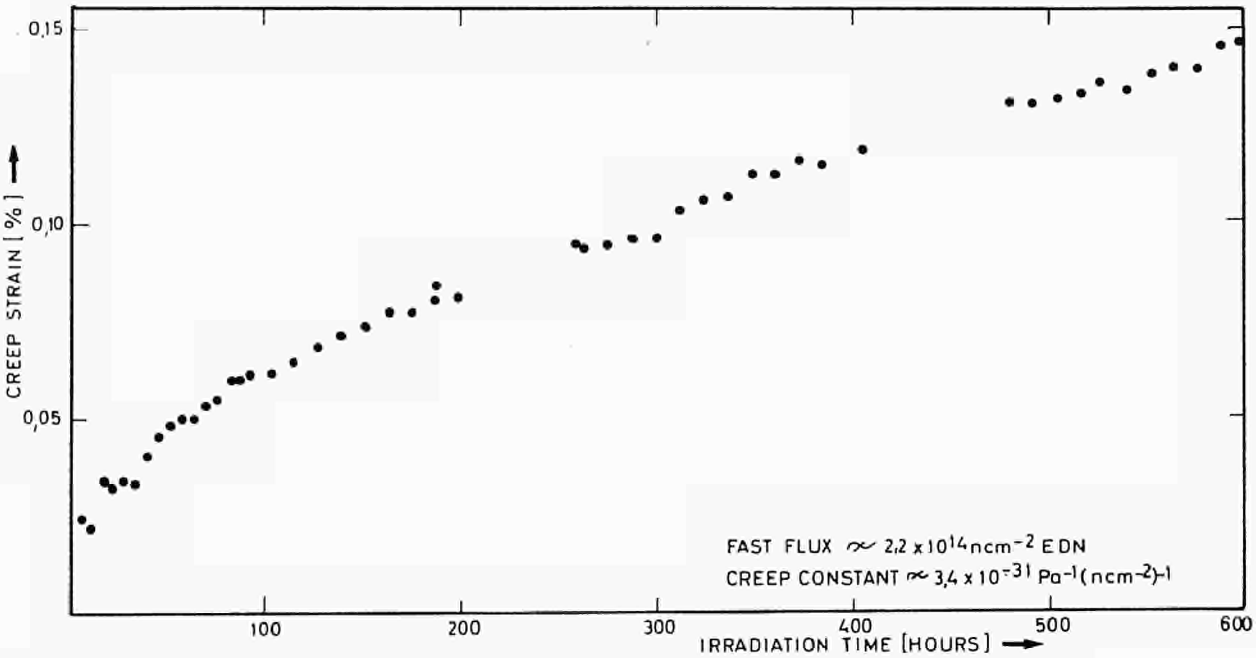


Fig. 34 D166-02 (H451) Creep experiment. Creep strain vs. time during the first irradiation cycle.

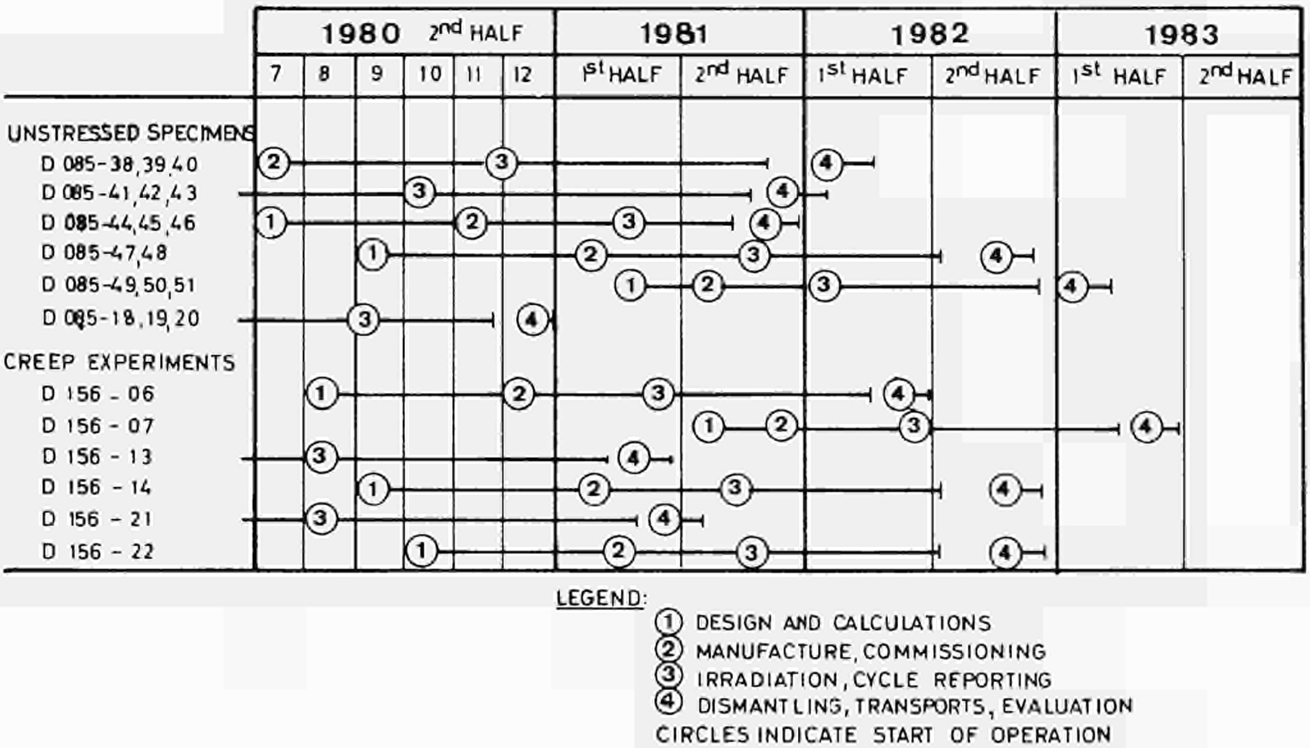
Irradiation started in cycle 80-06 and no problems have been experienced in attaining of maintaining the specified temperature of 850°C ± 20°C.
Primary creep saturated after about 100 hours giving a value of 0,7 of the initial elastic strain (Fig. 34).
Preliminary calculations of the steady state creep constant agree with previous results. Considerable problems have been experienced with hardware failures on the supervising PDP-11 system and this has resulted in the loss of some data. A contributory cause to the failure was probably the poor storage facilities available in the period between 166-01 and 166-02.

References to paragraph 2.2.3.1

1. High Fluence Creep Behaviour of near Isotropic Pitch Coke Graphite.
   M.R. Cundy (Joint Research Centre Petten)
   M.F. O'Connor, G. Kleist (KFA Jülich)
   Paper to the Carbon "80 Conference, Baden-Baden (June 1980).
2. FRG/US-Zusammenarbeit bei der Charakterisierung von Reaktorgraphiten.
   M.F. O'Connor *, G.B. Engle **, L.A. Beavan **, R.D. Burnette **, W.P. Eatherly ***, M.R. Cundy****
   * Kernforschungsanlage Jülich, Jülich, FRG
   ** General Atomic Company, San Diego, USA
   *** Oak Ridge National Lab., Oak Ridge, USA
   **** Joint Research Centre, Petten, the Netherlands

Fig. 35

GRAPHITE IRRADIATIONS
SURVEY OF FUTURE ACTIVITIES



2.2.3.2 HTR Fuel

High Temperature Reactor fuel testing is performed in HFR Petten on new advanced coated particle systems and production fuel elements with special emphasis on Nuclear Process Heat Reactor irradiation parameters.
The activities were concentrated during the reporting period on the evaluation of post-irradiation results of coated particle and fuel element in-pile tests, on the design of new advanced irradiation devices and on final commissioning of new sweep loops (see paragr. 2.3.3.9 of this report) which enable continuous fission gas analysis and temperature control to be carried out.

Coated Particles

R137 BATAVIA, Irradiation of Coated Particle Fuel in an Advanced TRIO facility

This was an experiment on a large variety of prototype and reference coated particle fuel. Ceramographical examinations on 10 fuel varieties of the compacted test specimens has been performed at the Hot Cell Laboratories of ECN. Results, which confirm non-destructive PIE, will be reported in a publication. One of the most significant results has been that the irradiation performance of coated particle fuel can be improved significantly by fission product getters in the fuel kernel (Fig. 36).

D162 ARTEMIS, Irradiation of Coated Particle Fuel for Failure Mechanism Investigation

Re-evaluation of irradiation history with resul results of gamma scanning, neutron dosimetry and dimensional measurements are currently being carried out. Results will be reported in a publication of the final report during the second half of 1980.

D175 PETTICOAT, Irradiation of LEU Coated Particle Fuel at 1000°C

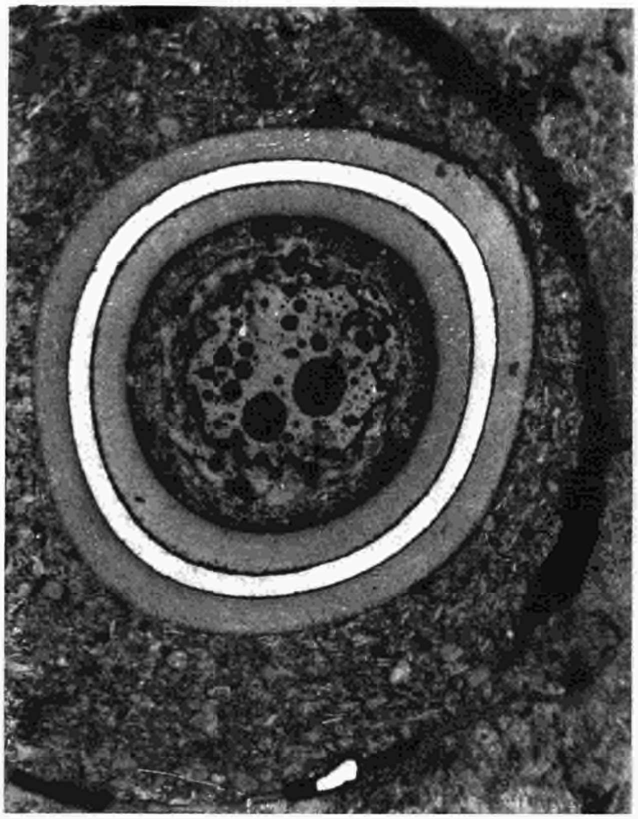
Design of the irradiation facility, TRIO 131 (see ref. [1], paragr. 2.3.3.7 and Fig. 83), and sample holder has been terminated. The isometric drawing on Fig. 37 illustrates the design principles and the connection to the new sweep loops.
Start of irradiation is planned for 2nd half of 1981, when standard quality coated particle fuel of the future low enriched fuel cycle is available.

Fuel Elements

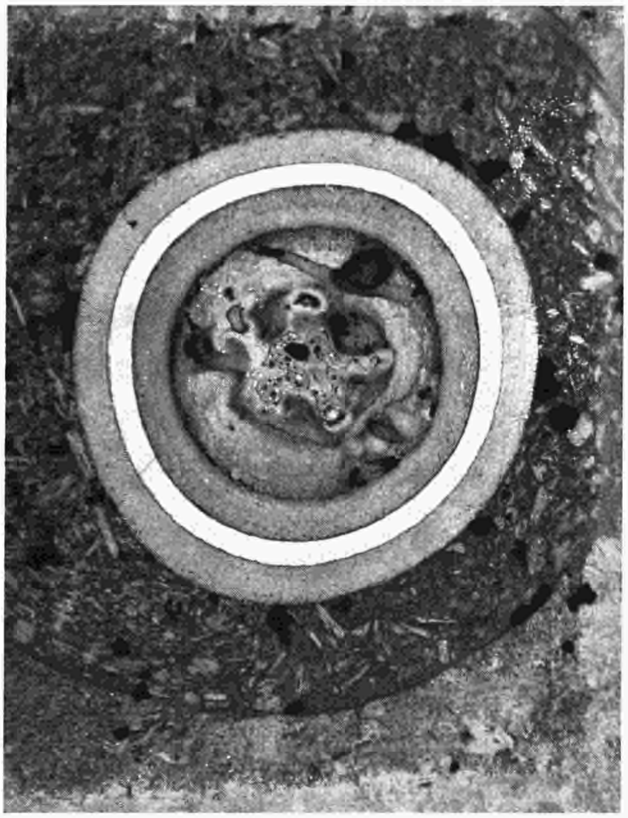
D138 Irradiation of Spherical HTR Fuel Elements for Nuclear Process Heat Reactor Development

E138-2 (2)

A second spherical fuel element test has been proposed by JRC Petten, mainly intended to



1240 ^oC



996 ^oC

1500^oC

Magnification 75 x

Burn-up : 73.7^o/o FIMA

Fluence : 9.42 x 10^{21} cm^{-2}

E > 0.1 MeV

Fig. 36 Irradiation performance of Al_2O_3 gettered UO_2 coated particles. Project BATAVIA. Particle type MA 1.

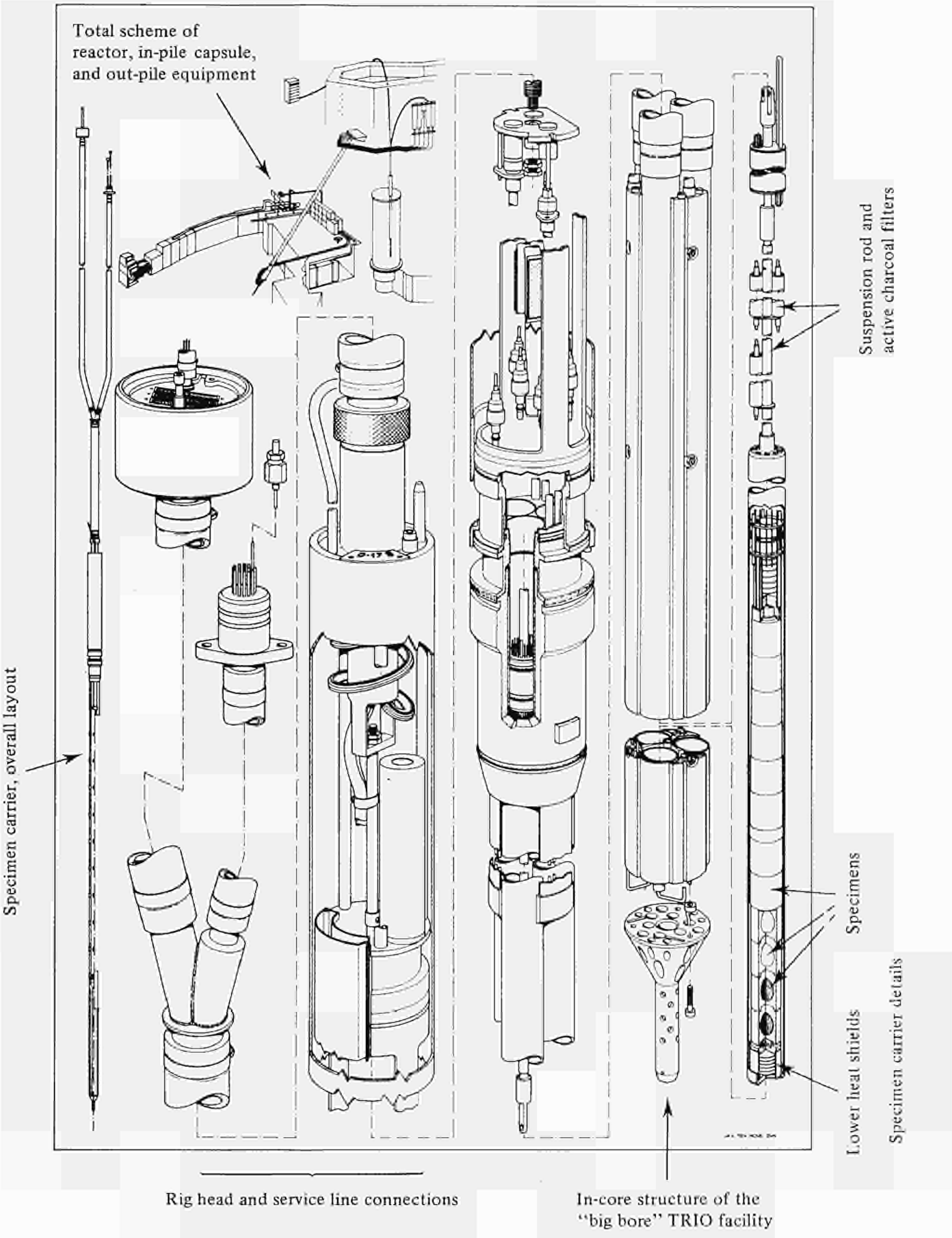


Fig. 37 Irradiation of low-enriched coated fuel particles at 1000°C (D175, "PETTICOAT"). Overall layout.

test the new sweep loops. Irradiation specimens will be UO_2-BISO particles containing fuel elements to investigate failure mechanism, observed during irradiation in the AVR*), and UO_2-TRISO particles containing fuel elements as test material to investigate reprocessing techniques. Both coated particle types contain 15°/o enriched or natural uranium fuel.
Irradiation temperature of "BISO" fuel elements will be cycled between 1300 and 900°C with cycle intervals. "TRISO" fuel elements are irradiated isothermally.

Due to the relatively high fuel mass (1,4 g U-235 and 18,5 g U-238) the experiment will be carried out in position G7 with low thermal flux density.
Nuclear and thermal evaluation is currently being performed. Components of the in-pile part have been fabricated. Irradiation start-up is planned for November 1980.

References to paragraph 2.2.3.2

1. Reloadable Irradiation Facility TRIO-131. Design and Safety Report.
R. Conrad, A. Franzen,
Technical Note P/10/80/15 (April 1980).

*) AVR = Arbeitsgemeinschaft Versuchsreaktor

2.2.3.3 Structural Materials

a) Stainless Steel

Irradiations in HFR Petten are carried out to stringent specifications concerning specimen temperatures and neutron fluences. They have supplied accurate information of material embrittlement by helium formation and fast neutron displacements. The present trend goes to fracture mechanics experiments and in-pile creep studies. The bulk of the irradiation work falls within the scope of the SNR 300 programme, in which the mechanical properties of stainless steel samples are measured after different levels of neutron exposures.
The facilities used in HFR Petten ("NAST", "MONA", "SINAS") feature sodium-filled specimen carriers with a large number of thermocouples, and operating at 550°C or 650°C.

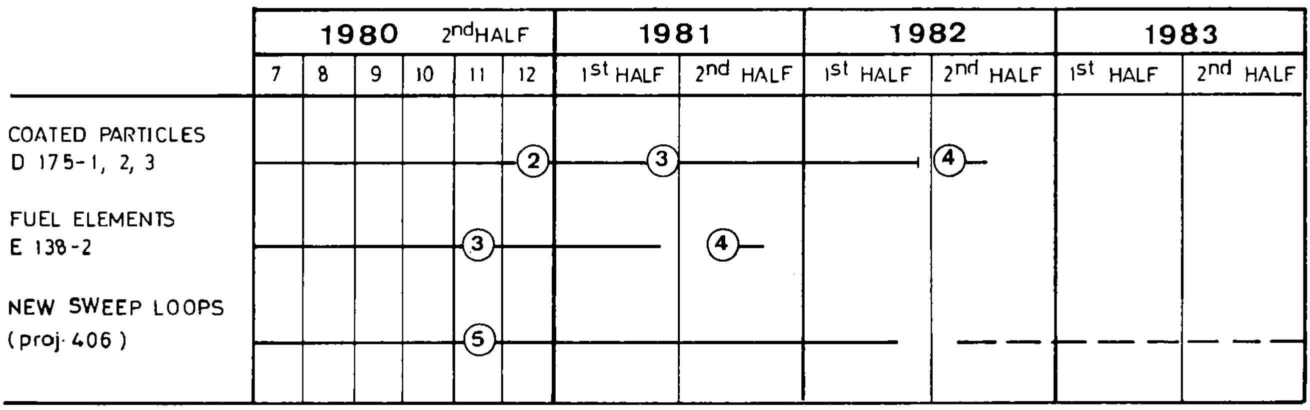
Fast Reactor Steel Irradiations
(see Fig. 44)

R120 ("NAST", ECN project 1.425). Investigation of the influence of irradiation-induced helium on the embrittlement properties of austenitic stainless steel.

R120-14 Fabrication of the R120-14 capsule has been delayed due to difficulties with preparation of the specimens. Reactor loading of NAST-14 is planned for November 1980.
The new thimble, which meets the dimensional requirements for non-interference with adjacent irradiation facilities has been manufactured and tested.

Fig. 38

HTR FUEL IRRADIATION EXPERIMENTS
SURVEY OF FUTURE ACTIVITIES



LEGEND: 1 DESIGN AND CALCULATION
2 MANUFACTURE, COMMISSIONING
3 IRRADIATION, CYCLE REPORTING
4 DISMANTLING, TRANSPORTS, EVALUATION
5 ACTIVE OPERATION SWEEP LOOPS
CIRCLES INDICATE START OF OPERATION

R143 ("MONA", ECN project 1.425)

R143-4 Manufacturing, assembly and testing of the MONA-4 capsule has been completed and irradiation has been carried out as scheduled on May 9, 1980. The achieved characteristics were :

- irradiation position H8,
- neutron fluence ~ 1 x 10^{22} m^{-2} (10^{18} cm^{-2})
- irradiation time 1,5 h at 45 MW
- temperature 550°C.

Evaluation of the thermal behaviour of the capsule shows that all nine specimens in the rig have been in the ± 25°C range, while eight of them were in the ± 20°C range.
The irradiation report of MONA-4 has been prepared [5].
Manufacturing of the MONA-5 and -6 capsules has been started. Reactor loading is planned for November 1980.

R139 TRIO and REFA capsule irradiations ("SINAS")

These experiments belong to the SNR 300 R & D programme, together with R120 and R143. They have grown from the limited original scope of irradiating a few cylindrical tensile specimens into a major series which now includes resilience and fracture mechanics test specimens (Fig. 40).
The following activities have been pursued during the reporting period:

- Post-irradiation evaluation of three terminated experiments have been carried out [1, 2, 3].
- The TK series has been extended into 1982 by the experiments TK8, 9, 10, 11. The irradiation conditions are identical with respect to the preceeding TK experiments. TK 5, 6 and 7 have been irradiated successfully in the period January to June 1980.
- The T20 series has been extended by two experiments, T25 and T26, both for an irradiation of 6 to 20 cycles. The three legs of T25 shall be operated at 450, 500 and 550°C respectively, and those of T26 at 400°C uniformly. One leg of T26 contains a stack of 21 Charpy specimens for which a new sample holder has been designed (see Fig. 39).
- The 40 series includes in the present working schedule one experiment T49 under usual conditions, three experiments with "full size" CT blocks (6 x 6 cm) and three with "half size" CT blocks (3 x 3 cm) The former, C47, C50, C51 (see Fig. 41) are irradiations of 5 blocks each, at 550, 300 and 600°C respectively. The latter, C411, C412, C413 (see Fig. 42) contain 20 blocks each, to be irradiated at 550°C.

The validity of the change to "half size" blocks has been concluded from last years' C410 experiment, where half and full size specimens have been irradiated at ambient temperature (see Fig. 43).
In the period January to June 1980 work has been performed on all 16 irradiation experiments listed in the present working schedule and on one preceeding experiment (TK 4).
The working schedule is shown in Fig. 44. A listing of major activities is given in Fig. 45.
It is worth mentioning that T49 includes the 100th SINAS type capsule in the frame of the R139 project and that none of them has ever failed neither mechanically nor to meet the required temperature and fluence conditions.
Statistical data on the observed thermal in-pile behaviour have been used for a simplified approach to post-irradiation evaluation [4].

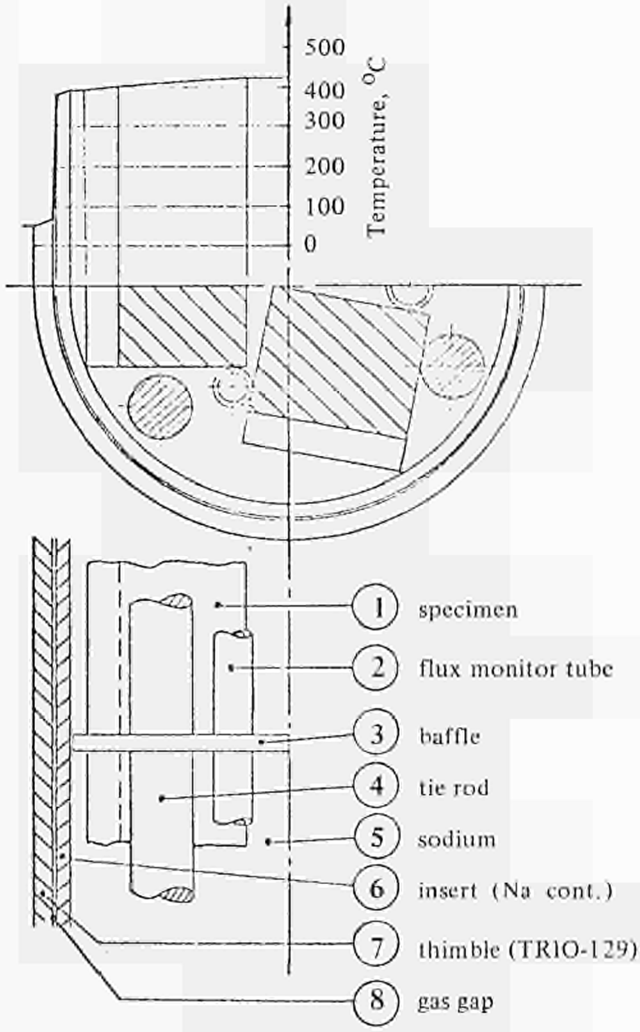


Fig. 39 Design of a new SINAS sample holder for CHARPY specimens (T26). Layout and radial temperature profile.

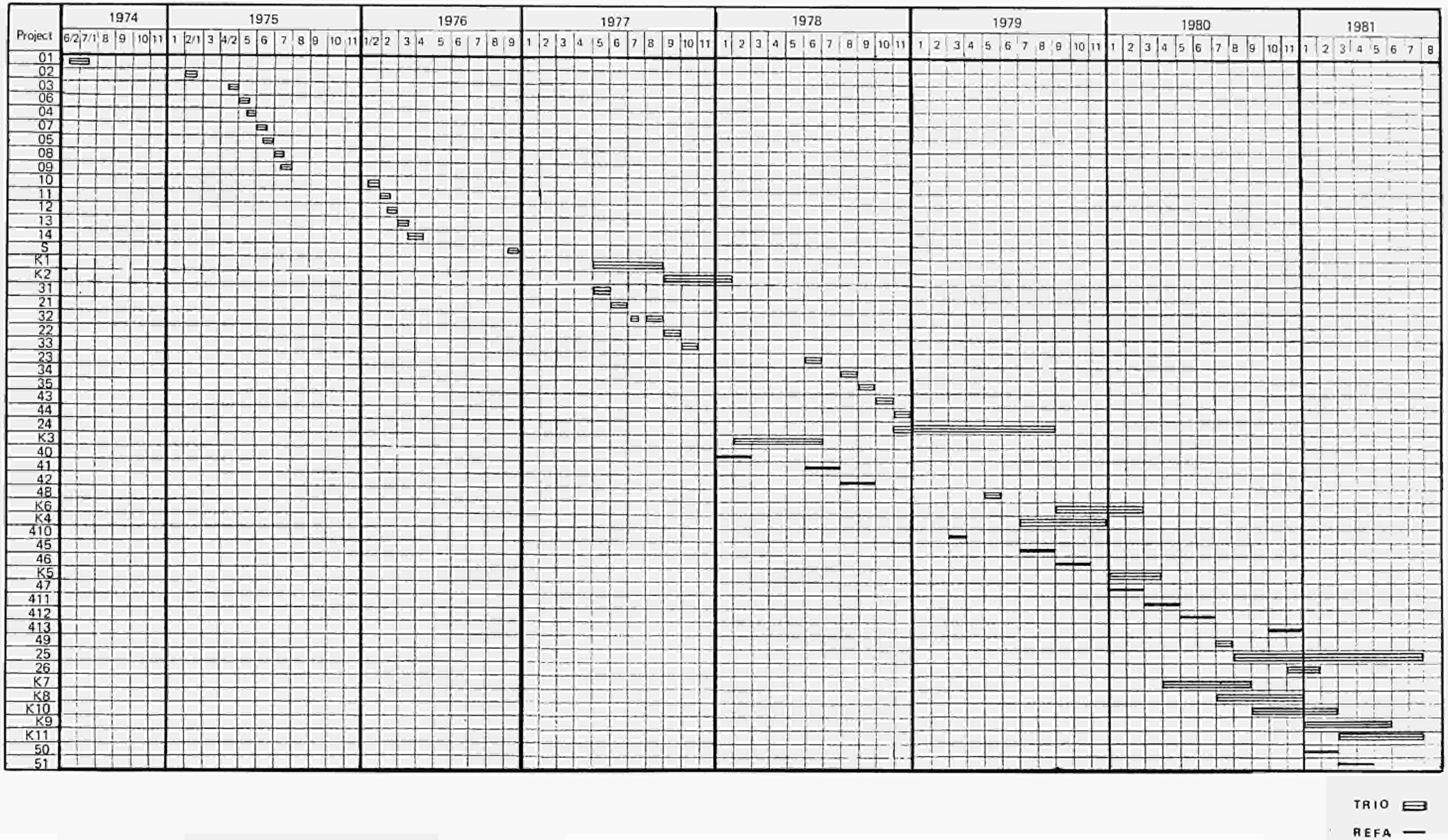


Fig. 40 Project R139 (SINAS). Historical survey.

01.04.1980

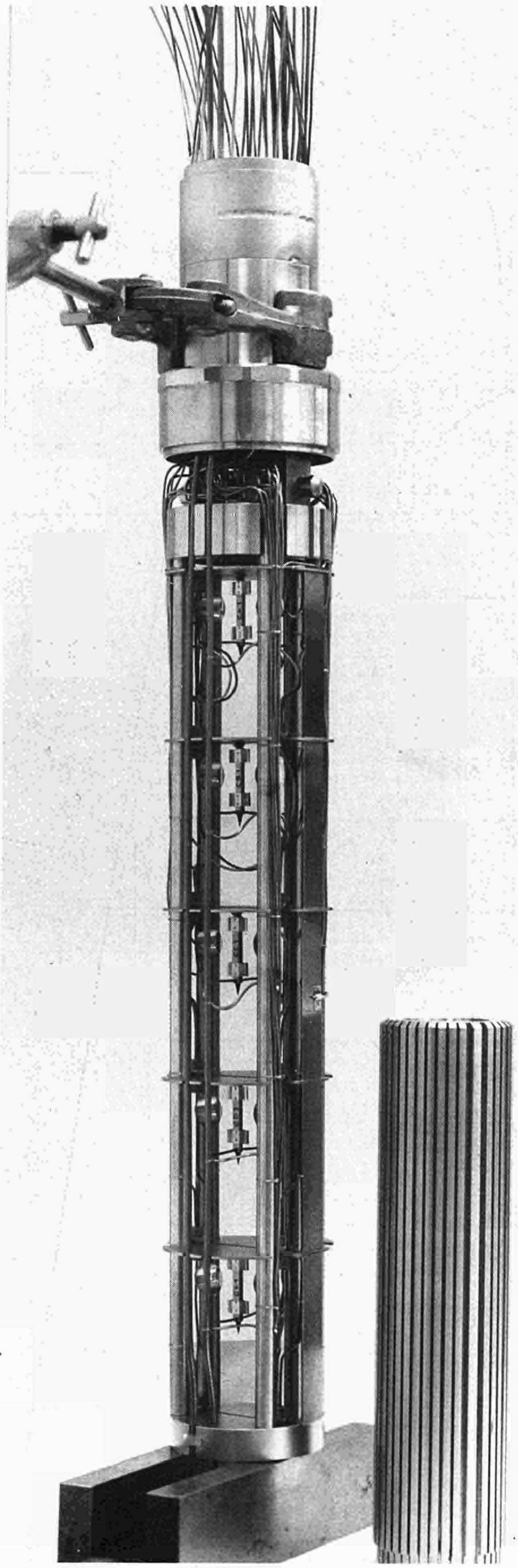


Fig. 41 "Full size" CT block experiment during assembly.

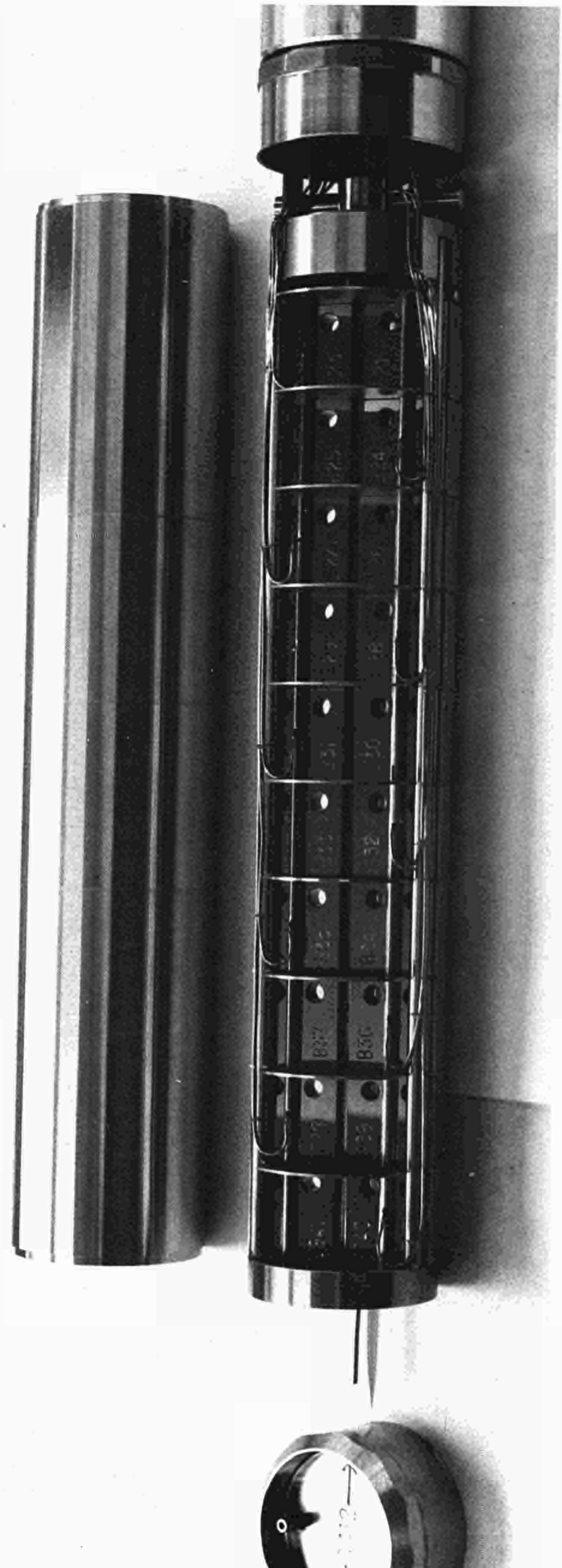


Fig. 42 "Half size" CT block experiment during assembly.

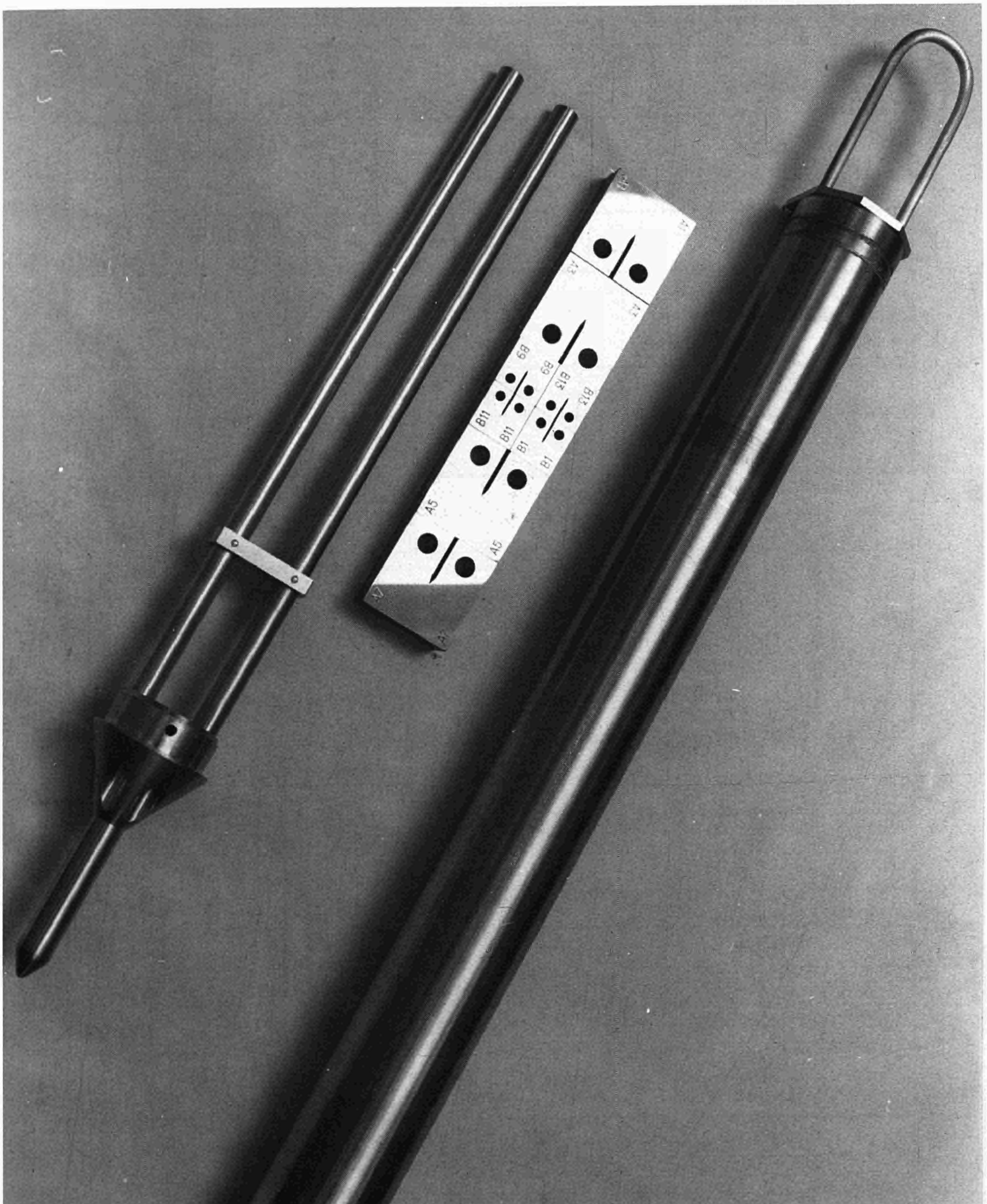


Fig. 43   Components of the CT block comparison test.

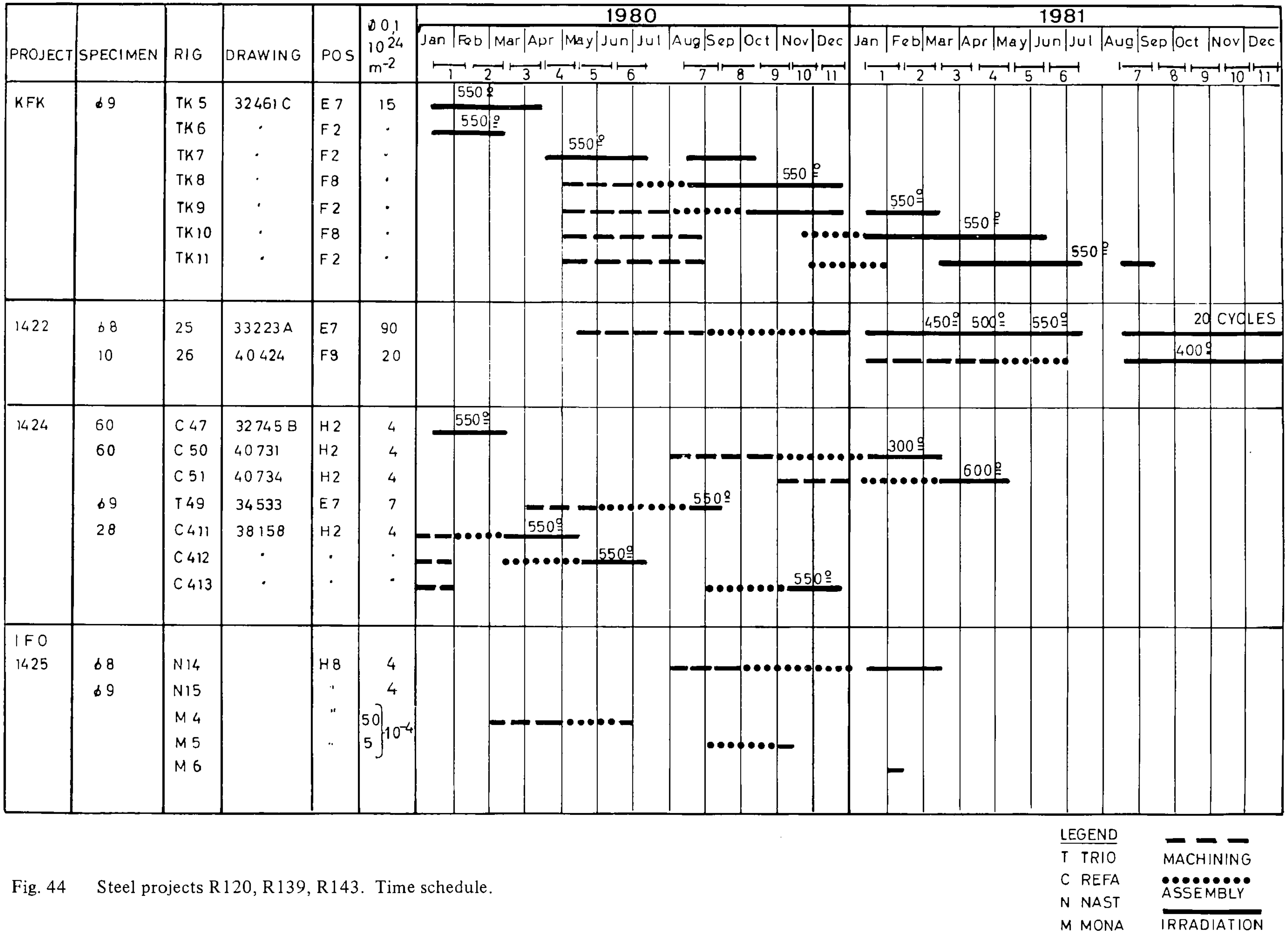


Fig. 44 Steel projects R120, R139, R143. Time schedule.

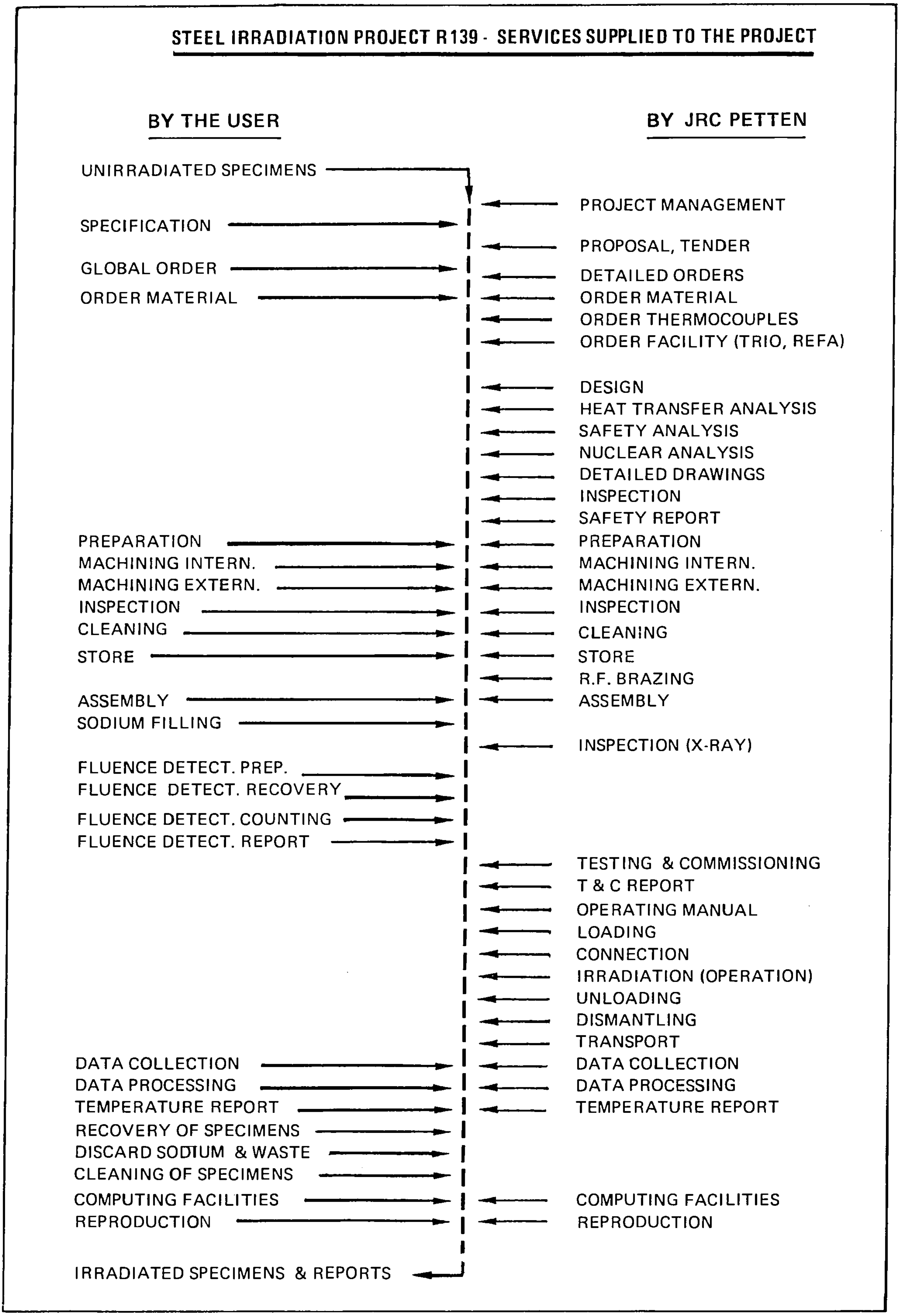

STEEL IRRADIATION PROJECT R139 - SERVICES SUPPLIED TO THE PROJECT
BY THE USER
BY JRC PETTEN
UNIRRADIATED SPECIMENS
SPECIFICATION
GLOBAL ORDER
ORDER MATERIAL
PREPARATION
MACHINING INTERN.
MACHINING EXTERN.
INSPECTION
CLEANING
STORE
ASSEMBLY
SODIUM FILLING
FLUENCE DETECT. PREP.
FLUENCE DETECT. RECOVERY
FLUENCE DETECT. COUNTING
FLUENCE DETECT. REPORT
DATA COLLECTION
DATA PROCESSING
TEMPERATURE REPORT
RECOVERY OF SPECIMENS
DISCARD SODIUM & WASTE
CLEANING OF SPECIMENS
COMPUTING FACILITIES
REPRODUCTION
IRRADIATED SPECIMENS & REPORTS
PROJECT MANAGEMENT
PROPOSAL, TENDER
DETAILED ORDERS
ORDER MATERIAL
ORDER THERMOCOUPLES
ORDER FACILITY (TRIO, REFA)
DESIGN
HEAT TRANSFER ANALYSIS
SAFETY ANALYSIS
NUCLEAR ANALYSIS
DETAILED DRAWINGS
INSPECTION
SAFETY REPORT
PREPARATION
MACHINING INTERN.
MACHINING EXTERN.
INSPECTION
CLEANING
STORE
R.F. BRAZING
ASSEMBLY
INSPECTION (X-RAY)
TESTING & COMMISSIONING
T & C REPORT
OPERATING MANUAL
LOADING
CONNECTION
IRRADIATION (OPERATION)
UNLOADING
DISMANTLING
TRANSPORT
DATA COLLECTION
DATA PROCESSING
TEMPERATURE REPORT
COMPUTING FACILITIES
REPRODUCTION

Fig. 45

E145 AUSTIN, Irradiation of Austenitic Steel Specimens for Strain Rate Studies

Small tensile samples are irradiated at 500°C ± 20°C in sodium filled capsules for the "dynamic load" project of the JRC Ispra Reactor Safety Programme.
The irradiation started in October 1978 and has since been operated without problems. All specimens remained within the specified temperature limits as shown on a typical post-cycle computer evaluated temperature plot given on Fig. 46.
From cycle 80-06 data processing of the temperature recording and temperature plotting will be carried out using the desktop computer Hewlett-Packard 9845 B.

E167 TRIESTE, Steel Creep Rig

This is a series of irradiation experiments for the JRC Ispra Fusion Reactor Materials Programme.
Examinations of the load-unload procedure of the sample tubes for intermittent dimensional measurements yielded some difficult manipulations at the rig upper parts which had to be transferred from the HFR pool side to the HFR hot cell because of possible radiation and contamination hazards of the personnel involved. The changes in handling procedure required major design modifications of irradiation rig components and especially on the semi-automatic loading jig.
Fabrication of dummy sample columns with supporting tubes is started, on which the following items will be examined:

- Functioning of tensile loading system
- Load-unload procedure of sample columns
- Possibility of dimensional measurements by means of neutron radiography.

E177 FANTASIA, Fracture Toughness of Austenitic Steels (JRC Ispra, IDEAS project)

Irradiation of this experiment started as scheduled with the first three sample holders E177/1-3 in the HFR cycle 80-02 and was continued with the sample holders E177/4-6 in the HFR cycle 80-03.
Both irradiations were carried out in HFR position H8 and operated just for one cycle to reach a target fluence in the order of $1{,}5 \times 10^{20}$ ncm^{-2}.
Irradiation temperature of the steel samples was kept at 350°C for the first irradiation step, respectively 550°C for the second one. Fig. 47 shows examples of the post-cycle computer evaluated temperature plot for both irradiation steps.

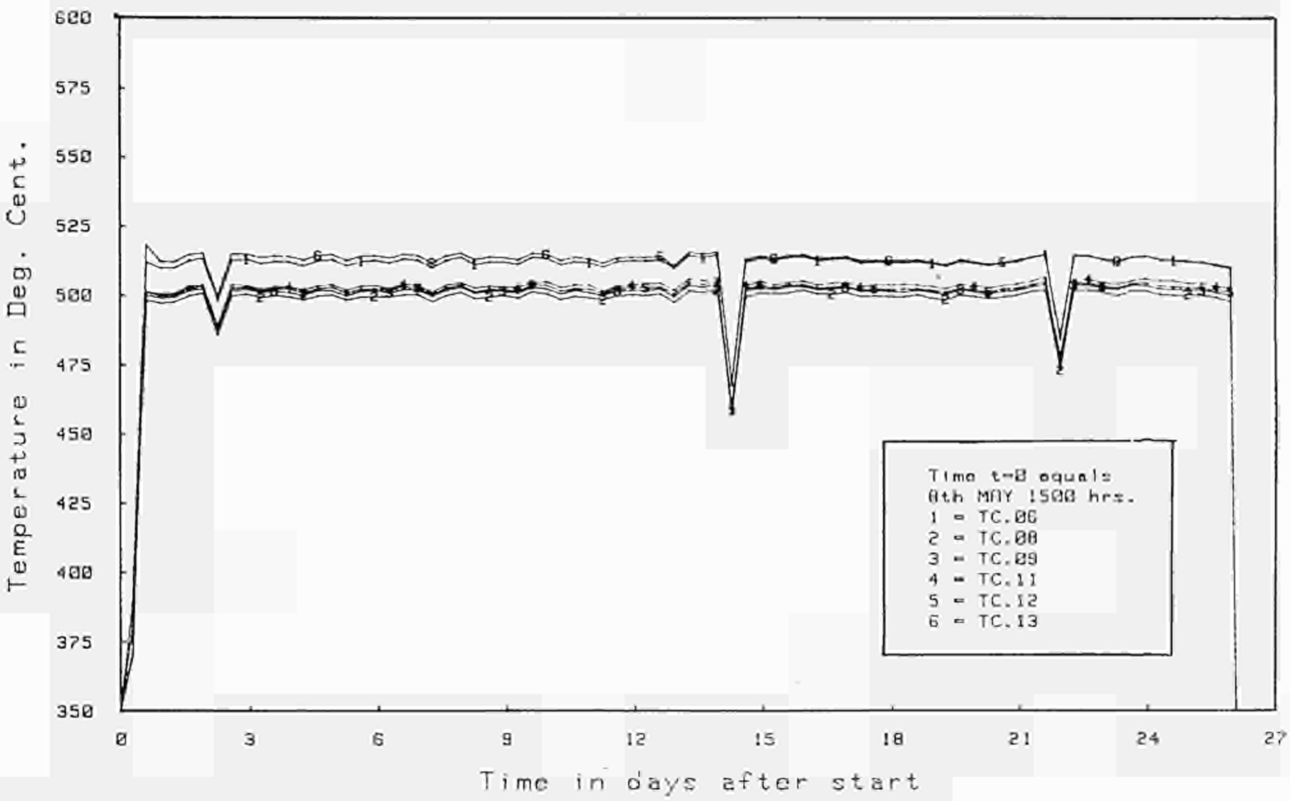
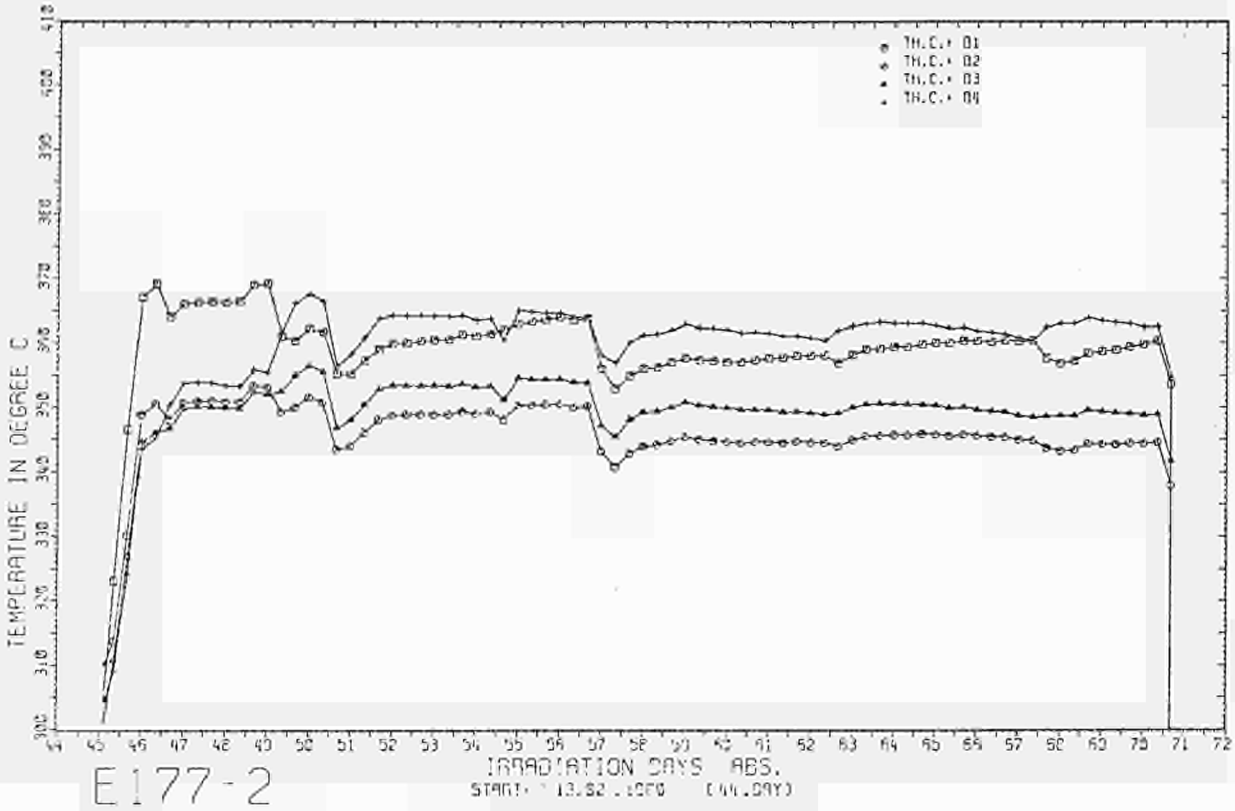


Fig. 46 Temperature history of experiment E145-02 (AUSTIN) during cycle 80.05.



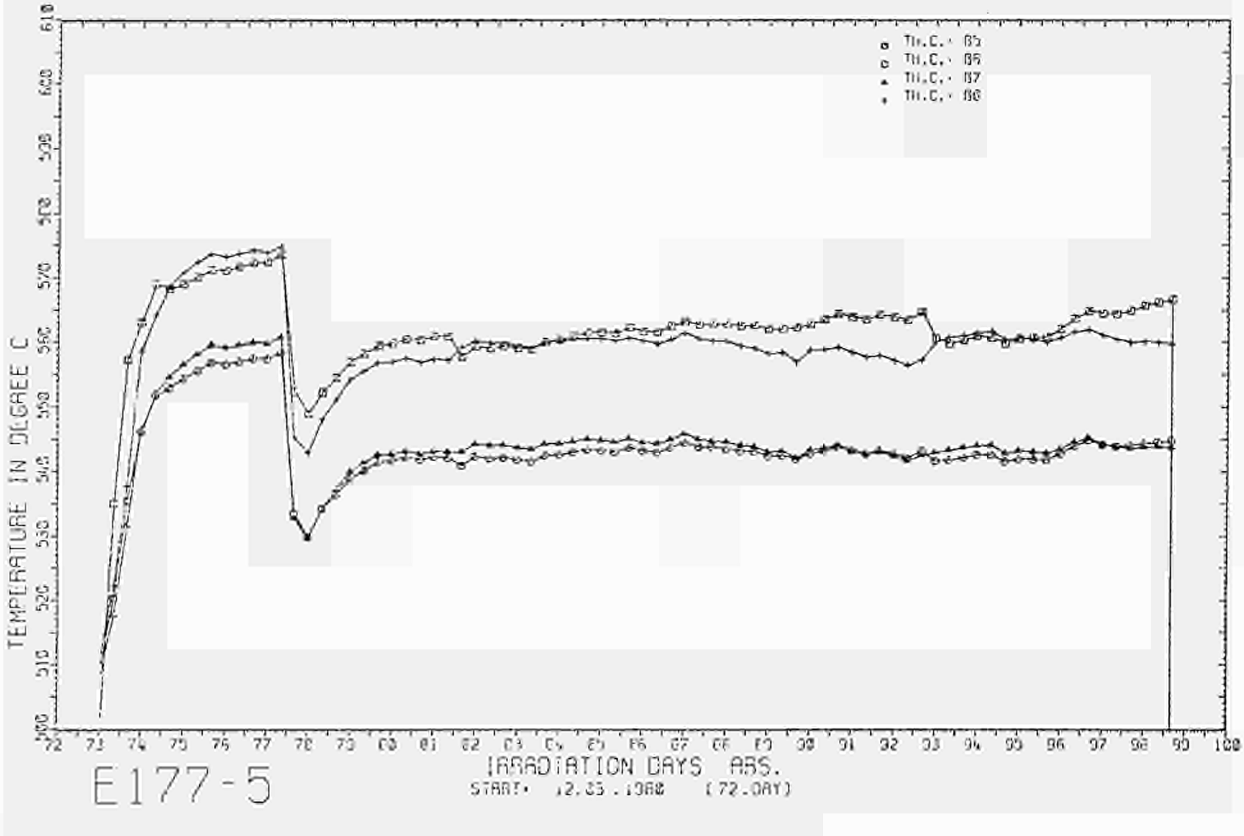


Fig. 47 Austenitic steel sample fracture toughness experiments, E177.
Examples of irradiation temperature histories during reactor cycles 80.02 and 80.03.

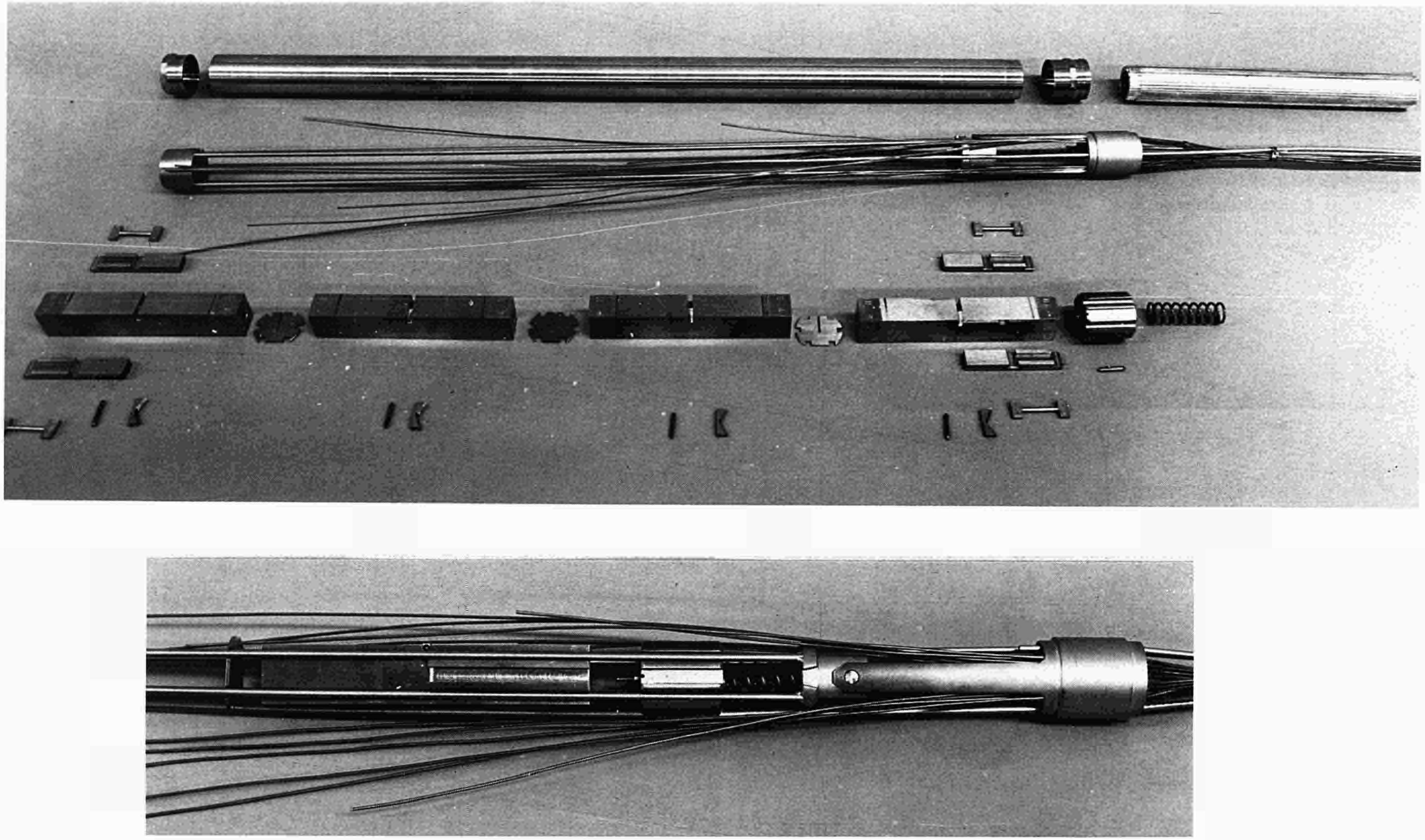


Fig. 48 Austenitic steel sample fracture toughness experiments E177. Specimen carrier assembly.

After irradiation the first six sample holders were unloaded from the TRIO irradiation facilities and transported to LSO for dismantling. The following sample holders E177/ 7 - 12 for the next two irradiation steps were assembled (Fig. 48) and are presently under reception. Irradiation is scheduled for HFR cycles 80 - 07 till 80 - 09.

D190 STORY. Short Time Irradiations of Pressure Vessel Steel Samples.

Within the FKS Programme * for LWR pressure vessel materials a number of short time, low temperature irradiations have to be carried out. Since sample temperatures have to be below 150°C in high flux core positions, direct cooling of the samples by primary circuit water is envisaged. An irradiation proposal containing feasibility studies for two different irradiation devices has been submitted to and discussed with the sponsor **. Electron beam welding tests are being prepared on suitable sealing constructions, which avoid direct contact of the non-stainless steel samples with the primary circuit water.

b) Other Structural Materials

R151 Niobium and Vanadium Irradiation between 450 and 800°C, "NIRVANA"

This project concerns fundamental radiation damage studies on candidate fusion reactor materials.
In the scope of irradiation testing of candidate fusion reactor materials in (fission) materials testing reactors numerous discussions are being held worldwide about the representativity of such experiments. Accepted strategies are :
- careful modelling, dosimetry, flux tailoring
- special in-pile simulation "tricks"

R151-31 Nb and V disks of 15 mm Ø are irradiated in an inert, extremely clean atmosphere at four temperatures, i.e. 450, 600, 700 and 800°C.
R151-31 was irradiated during 6 cycles, accumulating a fast neutron fluence of $\sim 6 \times 10^{21}$ cm^{-2} (> 0.1 MeV) compared to the 2 cycles or $\sim 2 \times 10^{21}$ cm^{-2} of R151-11. It started in cycle 79-07 (August, 9th), and performed well until its scheduled end in February, 1980 [7]. Since that time the three sample holders are waiting for dismantling.

R151-12 is the complementary experiment to R151-11. It contains, in a Refa 170, tensile and fatigue specimens, protected against NaK contact by individual Nb sheaths. Temperature control is done by means of a double gas gap, individually controlled for each temperature region.
R151-12 started its irradiation in the last cycle of the year, on November, 19th. Being placed in a considerably warmer position of the reactor than originally planned (E7 instead of D2) which meant a maximum nuclear heating of 8.2 instead of 5.7 W/g, the irradiation temperatures were, during the first few days of the cycle, slightly too high.
This could be overcome by operating the reactor at only 42 MW instead of 45 MW, and by tolerating an irradiation temperature of the top drum of 500 °C rather than 450°C. During the last two days of the cycle some phenomena were observed that could be an indication of a small leak in the NaK containment. Later, this leak was confirmed by neutron radiography, which showed that during the three weeks maintenance stop in December/January, about 50°/o of the 700 cm^3 NaK contained in the sample carrier had leaked out into the Refa thimble.
The irradiation was interrupted after 1 cycle (schedule: 2 cycles) [6].
A safe dismantling procedure is presently being elaborated.
The continuation of the NIRVANA programme has been cancelled, in view of the problems with R151-12 and after a major failure during the assembly of capsule R151-32.

R158 "HOBBIE", Zircaloy Creep Experiments, ECN project 1.085

During the past half year the irradiation reports on the sixth and seventh experiment have been compiled (see ref. [8] and [9]), one irradiation has been finished while the irradiation of a new experiment has been started and was continuing at the end of the reporting period.

R158-7 (HOBBIE-7). The irradiation of this capsule was continued in reactor cycle 80.01 after the experiment had already been run for one cycle in 1979.
Initially the same conditions as for the previous cycle, i.e. 343°C (650°F) and 18.6 MPa (2700 psi) external pressure, have been applied. After 160 hours the pressure in the capsule and specimen were reversed and adjusted to 3.45 MPa (500 psi) internal overpressure.

*) Research Programme "Component Safety"
**) KFA-Jülich,

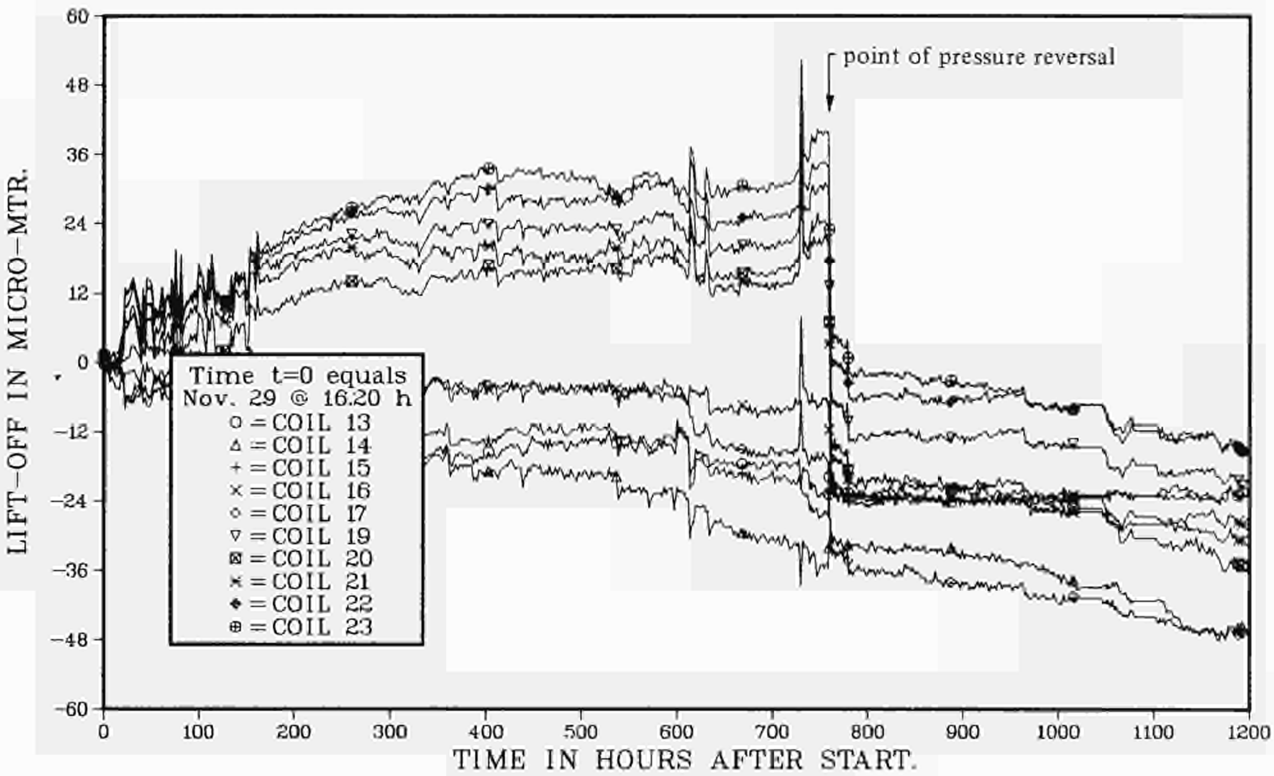


Fig. 49 Experiment R158-07. Cladding deformation vs. time.

In spite of the numerous reactor disturbances the irradiation yielded no technical problems. None of the eddy-current coils in the deformation monitoring device failed. A typical plot of the lift-off measurements as a function of the irradiation time is shown in Fig. 49.

R158-8 (HOBBIE-8)

The irradiation started on April 12 for three cycles in position H8. During the first cycle the same conditions as for the previous capsule, i.e. 343^{o}C (650^{o}F) and 18.6 MPA (2700 psi) external overpressure, were applied. After 250 hours a leakage developed, helium entered and subsequently escaped from the cooling water flow. As the resulting increase in gas-activity in the reactor building was almost negligible it was not necessary to stop the irradiation. Probably the leakage was located at the solder joint of the upper bulkhead.

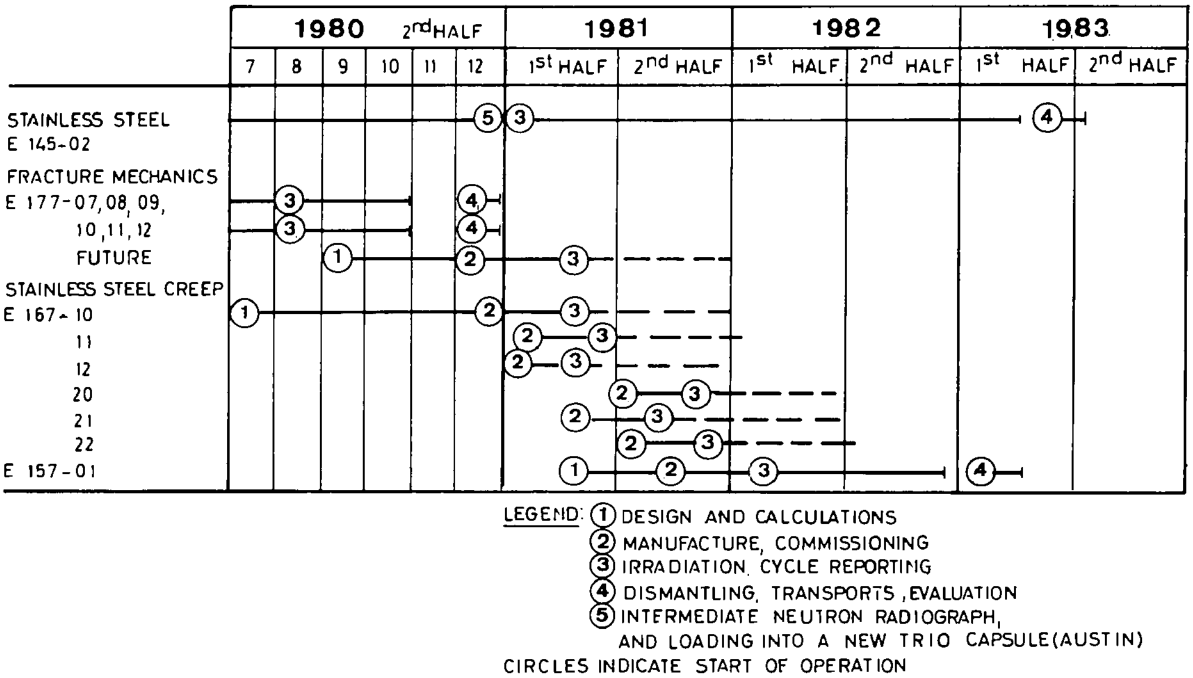
Although the leak rate gradually increased, sufficient helium could be supplied to maintain the required pressure conditions and the irradiation could be continued up to the end of the cycle.

During the reactor stop the pressure difference across the specimen was reversed. The irradiation was continued with cycle 80.05 at an internal overpressure in the specimen of 5.2 MPa (750 psi). Because the pressure vessel was now at low pressure the leak rate was negligible and the irradiation could be continued without problems.

References to paragraph 2.2.3.3

1. Irradiation of steel specimens in sodium, using a TRIO facility. In-pile performance of R139-24. D. Vader, R.A. memo 80-02 (January 1980)
2. Irradiation of steel specimens in sodium, using a REFA facility. In-pile performance of R139-45. D. Vader, R.A. memo 80-06 (February 1980)
3. Irradiation of steel specimens in sodium, using a REFA facility. In-pile performance of R139-46. D. Vader, R.A. memo 80-12 (January 1980)
4. Project R139. Reassessment of the nuclear heating influenced correlation of specimen temperature to thermocouple temperature in SINAS type steel irradiations. P. Zeisser, Technical Note P/10/80/19 (May 1980)

Fig. 50 STRUCTURAL MATERIAL IRRADIATIONS (EXCEPT R120, R139, R143)
SURVEY OF FUTURE ACTIVITIES



5. Bestralingsverslag MONA-4 (R143).
D. Vader,
R.A. memo 80-19 (June 1980)
6. NIRVANA R151-12. Irradiation Report.
H. Scheurer,
Technical Note P/10/80/10 (March 2980)
7. NIRVANA R151-31. Irradiation Report.
H. Scheurer,
Technical Note P/10/80/11 (March 1980)
8. Irradiation report of the HOBBIE-6 capsule.
Th. van der Kaa,
ECN 80-038 (February 1980)
9. Irradiation report of the HOBBIE-7 capsule.
Th. van der Kaa,
ECN 80-100 (June 1980)

2.2.3.4 LWR Fuel

D125, D176, D178, Power ramp tests of pre-irradiated LWR fuel pins

The irradiation programme for power ramp testing of pre-irradiated LWR fuel pins was continued (see Table 13). However, due to operational difficulties of the new PSF table and trolleys, to deficiencies in out-of-pile installation, and due to lack of operation personnel the number of tests performed was less than scheduled. Within the reference period 6 in-situ, 1 modified in-situ and 6 other tests were performed.

Table 13 Number of power ramp test on LWR fuel pins.

Year	Ramp test types: Start-up SU	In-situ IS	Modified in-situ ISM	Prototype resp. special tests	Total/Year, ( ) - with pre-irradiated fuel pins
1976	4	8	–	4	16 (12)
1977	2	20	2	4	28 (24)
1978	–	13	19	4	36 (32)
1979	–	15	5	12	32 (26)
1980*	–	6	1	6	13 (12)
Total	6	62	27	30	125 (106)

*) First half of 1980

On two fuel pins irradiation continued after ramp testing for re-ramping after a burn-up increase of about 3 MWd/kg (U). In a demonstration test with a BWR fuel pin the capability of the BWFC-installation for performance of fast ramps at 600 $Wcm^{-1}min^{-1}$ power ramp speed for simulation of control rod movement was successfully shown.

The BWFC-RA-type capsule for continuous monitoring of fuel pin length became operational within the ramping programme. On 15 pre-irradiated fuel pins pre-ramp examinations at the hot cells at Petten was completed. On 16 ramp tested fuel pins post-irradiation examination was terminated.
In June, 16 fuel pins were delivered to KWU hot cells for further investigation. 31 pre-irradiated fuel pins were received for continuation of the ramping programme.

Due to increasing contamination of the primary systems of the BWFC installation an unscheduled maintenance stop was required in May. During this stop also a system for pressurisation and pressure testing of newly loaded capsules was installed. By this means an important reduction in operation and capsule handling has been achieved.

The computer programme developed in the previous period for calculation of form factors and power distributions vs. position of the experiment relative to the HFR core, control rod position, and axial position of the fuel column has been extended with a programme which allows comparison of these calculations with the gamma-scan measurement.
On most of the experiments performed this programme was applied. It was found, that the power profiles calculated on basis of the thermal flux data are confirmed by the gamma scan results (Fig. 51). A major part of this work is performed on the HP 9845 computer system (Fig. 52).

On basis of the results derived by TWOTRAN calculations for the power density distribution in LWR fuel pins in the PSF (asymmetric distributions due to the flux gradient) two-dimensional temperature calculations were performed. The calculations confirmed previous experience from metallographic investigations that in spite of pronounced asymmetry the temperature fields are nearly symmetric.
The local values of radial heat flow at the fuel pin surface deviate only by about ± 5 °/o (Fig. 53).

Assembly of 5 capsule carriers of the new standard version was completed. The future ramp test programme for a period up to 1983 has been discussed by all participating organizations. About 25 to 30 ramp tests are anticipated per year.

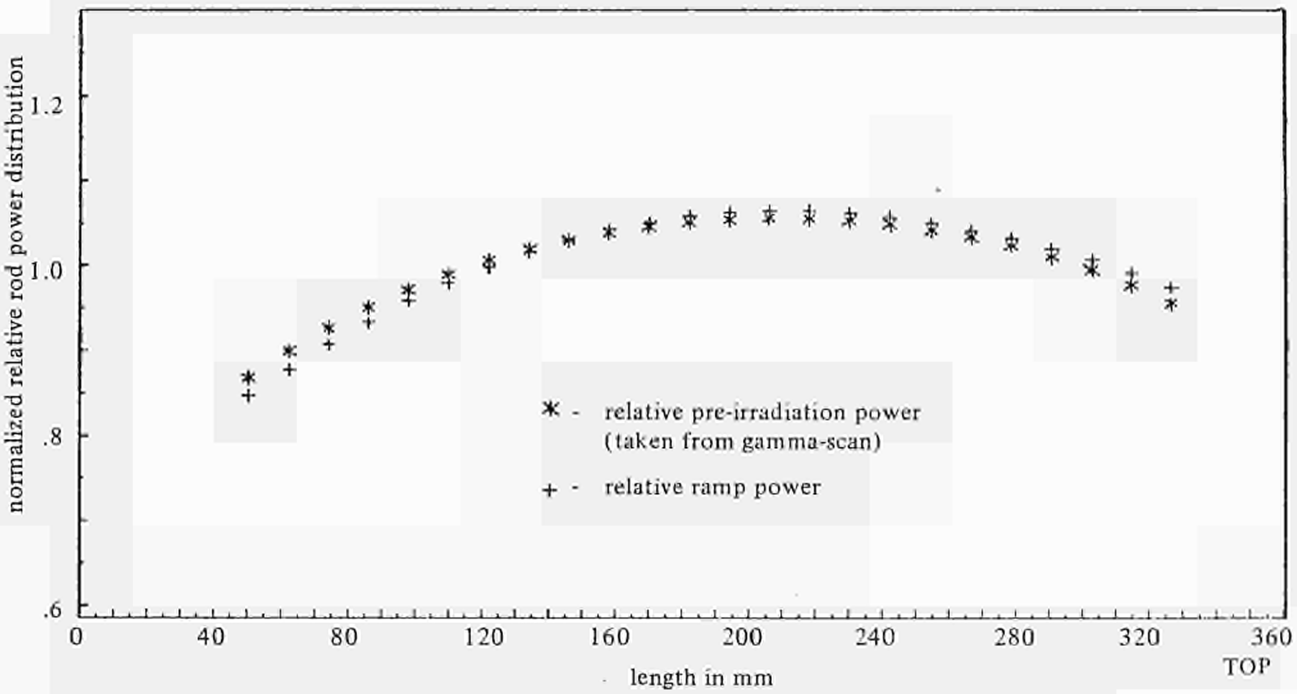


Fig. 51 LWR fuel rod ramp tests. Comparison of calculated and measured axial power distribution.

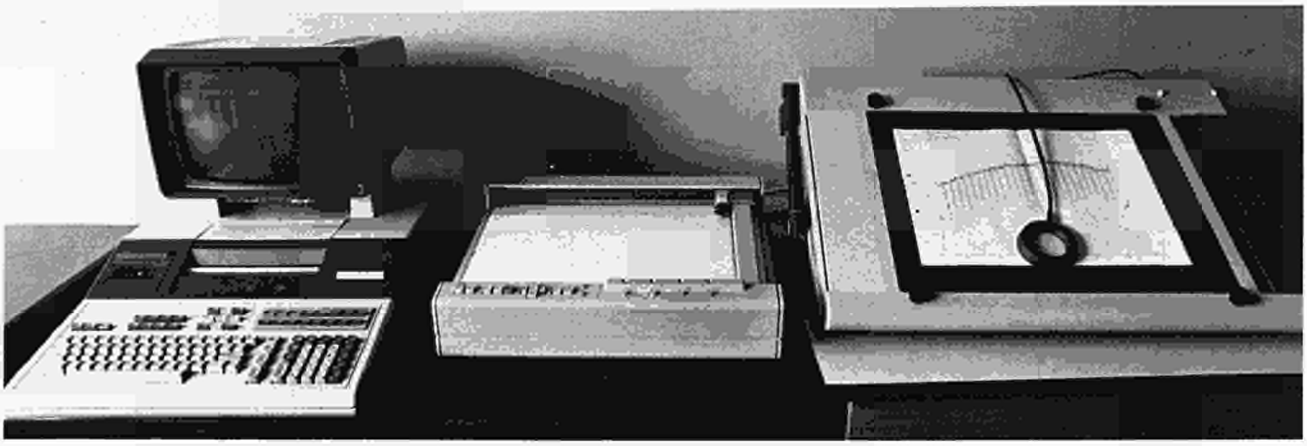


Fig. 52 HP 9845 System (dual floppy disk drive not shown on this photograph).

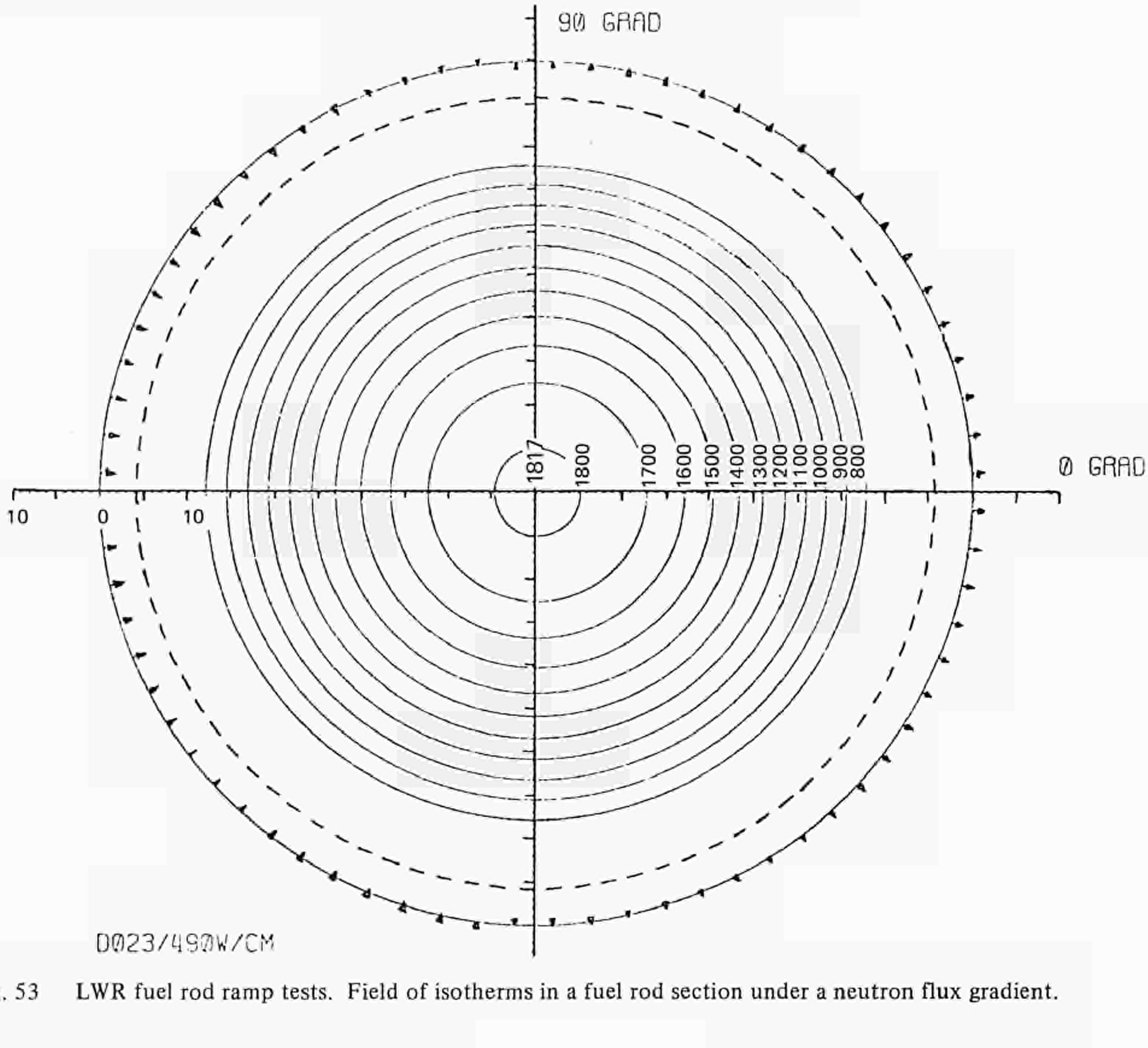


Fig. 53 LWR fuel rod ramp tests. Field of isotherms in a fuel rod section under a neutron flux gradient.

D128 In-pile measurements on LWR fuel

- Post-irradiation examens on the first capsule (D128-02) have been terminated.
- The second experiment could not be continued in irradiation due to a leaking seal at the pressure vessel. Repair procedures are under investigation.

Design of the fuel pins for two new D128 experiments started at KWU. Design for the irradiation device had already been completed in the preceeding reporting period.

R174 "POTRA", Prototype In-core Irradiation Facility for Power Transients
(ECN project 8.290.23)

After a period during which little work was done on the "POTRA" facility renewed interest, a.o. from the Dutch Utilities (KEMA) in the utilization of the device has reactivated design and development work on the project. The POTRA facility has been designed with two aims in mind, i.e.

- providing transient irradiation capacity for long fuel pins,
- relieving the pool side facility (which is technically the logical place for transient irradiations, but which is already up to full capacity) by utilizing an in-core (reflector) irradiation position.

Technically the facility will have much in common with the BWFC experiments, i.e. the same heat removal principle (boiling or stagnant water layer) will be applied, the facility will be reloadable and all out-of-pile cooling, sampling and instrumentation services will be supplied by the BWFC installation.
Main differences between the two facilities consist of the application of an absorber gas controlled power regulation principle for the POTRA facility and its longer fuel pin accommodation space. The in-pile section consists of a thimble, an insert piece and the irradiation capsule.
In Fig. 54 a schematic impression of the in-pile section is given.

The process condition in the capsule will be the same as for the BWR conditions, 70 bar and ~ 280^{o}C. The power control system using BF_3 is identical to the installation developed for the TOP experiment (see paragr. 2.2.3.5).
The main purpose of the present work is to demonstrate the feasibility of the facility and to verify its thermohydraulic, power rating and power control characteristics. A two-cycle irradiation period is considered sufficient for this qualification.
Eventually the facility will probably be used for a joint ECN-GKN irradiation programme. This programme comprises the determination of the susceptibility of pre-irradiated pellet and vipac-type fuel pins to power changes.

Reference to paragraph 2.2.3.4

1. Correlation of fission gas release and redistribution of volatiles in test fuel rods due to power ramping.
F. Sontheimer, W. Vogl, H. Knaab,
Enlarged Halden Programme Group Meeting, Lillehammer, Norway (June 1980).

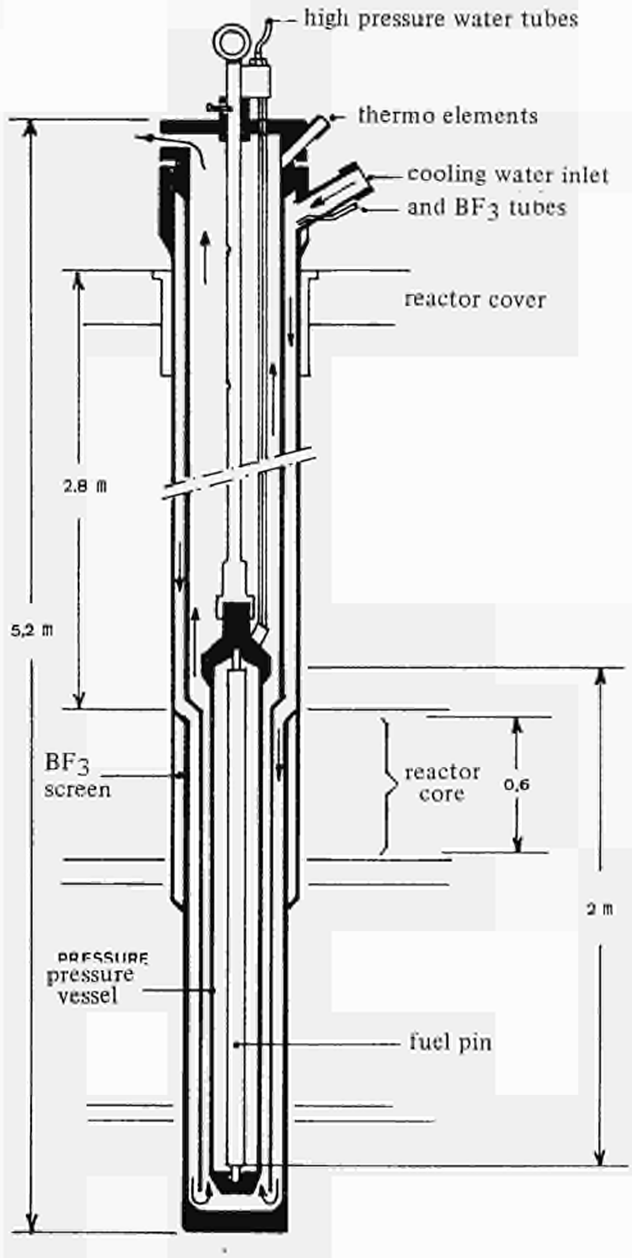


Fig. 54 R174. Power transient capsule. Schematic layout.

2.2.3.5 Fast Reactor Fuel

Fast reactor fuel experiments in HFR Petten currently fall under three categories:

a) safety tests, featuring severe abnormal and/or transient conditions which will normally lead to pin failure.
These are mainly the ECN sponsored experiments R054, R063 and in future R120 in support of SNR 300.
b) transient tests with "mild" transients and/or off-normal conditions which will normally not lead to pin failure.

These are mainly KfK-sponsored experiments (D173, D183, etc.) for SNR 300.

c) fundamental study oriented fuel behaviour experiments for JRC Karlsruhe (E084, E170, E172 and E181).

a) Safety Tests

R054 "SHOT". Fast Reactor Fuel Test at Stationary High Over Temperature (ECN project 1.413)

The irradiation programme as such has been terminated. Remaining work consists of the completion of the irradiation reports. During this period the irradiation reports of experiments R54-F37, F38 and F44 have been completed [1, 2, 3].

R63 "LOC", Fast Reactor Loss-of-Cooling experiment. (ECN project 1.413)

This irradiation programme addresses the behaviour of short UO_2 fuel pins of SNR-300 specification under conditions where external cooling is suddenly stopped, while full power generation still continues for a short period.
Starting conditions for all irradiations are:
linear fuel power : 550 W/cm
Na-temperature : 500 - 550 ^{o}C
while fuel burn-up (0 - 65 MWd/kg), internal fuel pressure (0 - 75 bar), shroud type (open or closed) and loss-of-cooling time (10 to 20 sec) are varied for the different experiments.

R63-20 (LOC 20)
As reported earlier, the irradiation of this capsule had to be stopped because of a leak between the outer safety containment of the capsule and the reactor pool. Based upon an assessment of the chance of a simultaneous leakage of the sodium containment and the sodium potassium containment, the reactor safety committee rendered a positive advice with regard to the continued application of the capsule for the intended Loss-of-Coolant transient.
In the mean time, several problems with the instrumentation for transient data acquisition continued to cause long delays in the execution of the transient. At the end it proved impossible to repair the instrumentation. By borrowing from the TOP project and from the noise analysis group a new instrumentation set-up could be composed. On May 7 the transient irradiation has been performed. The "Loss-of-Coolant" time was 15.5 seconds.
A neutrograph of the capsule has been made, but as the leakage of the cooling channel made it impossible to remove all water from the capsule, the result was rather poor.

R63-21 (LOC 21)
On May 8, the transient irradiation has been performed. The Loss-of-Coolant time was 17.5 seconds. As also this capsule had a leakage in the cooling channel, again the subsequent neutrograph was of inferior quality.

R124 "TOP", Fast Reactor Fuel Pin Over Power Experiments (ECN Project 1.417).

This irradiation programme is designed to study the behaviour of short, single UO_2 fuel pins of LMFBR specifications under relatively slow and low overpower transients (rise times variable from 0.5 - 15 seconds, overpowers from 1 to 4 times nominal power). Both fresh and pre-irradiated fuel pins will be used.
Experimental instrumentation will be applied to identify fuel and clad axial deformation, fission gas pressure build-up, fuel and sodium temperatures, flow and pressure conditions, etc. before and during the overpower transient.

R124-1 Design and engineering of prototype irradiation facility.

- The design work of the irradiation facility is practically completed (Fig. 55).
The prototype of the sodium-water heat exchanger has been tested successfully in the Component Test Facility (CTF) at the ECN Technology Division. Presently a hydraulic model of the test section is under construction. It contains an electrically heated pin. This mock-up will be used for measurement of the thermo-hydraulic behaviour of the TOP system, complete with linear induction pump and heat exchanger, under both nominal and power transient conditions in the CTF.
The draft design and safety report is ready except for certain stress- and thermohydraulic calculations which are being carried out presently. The construction of the gas supply systems and instrumentation panels is nearing completion. They will be installed in the reactor hall in August 1980.
- Data handling system.
Software is nearly completed. The computer room in the reactor hall is ready. Installation of the data handling system is planned for August 1980.
- BF_3 dynamic shielding system.
The exact replica of the TOP-BF_3 system has been successfully tested with N_2. A leaking electromagnetic valve prohibited testing with BF_3. New valves have been ordered in April, but are not yet delivered.
The BF_3 destillation apparatus has been tested with natural BF_3. The system works

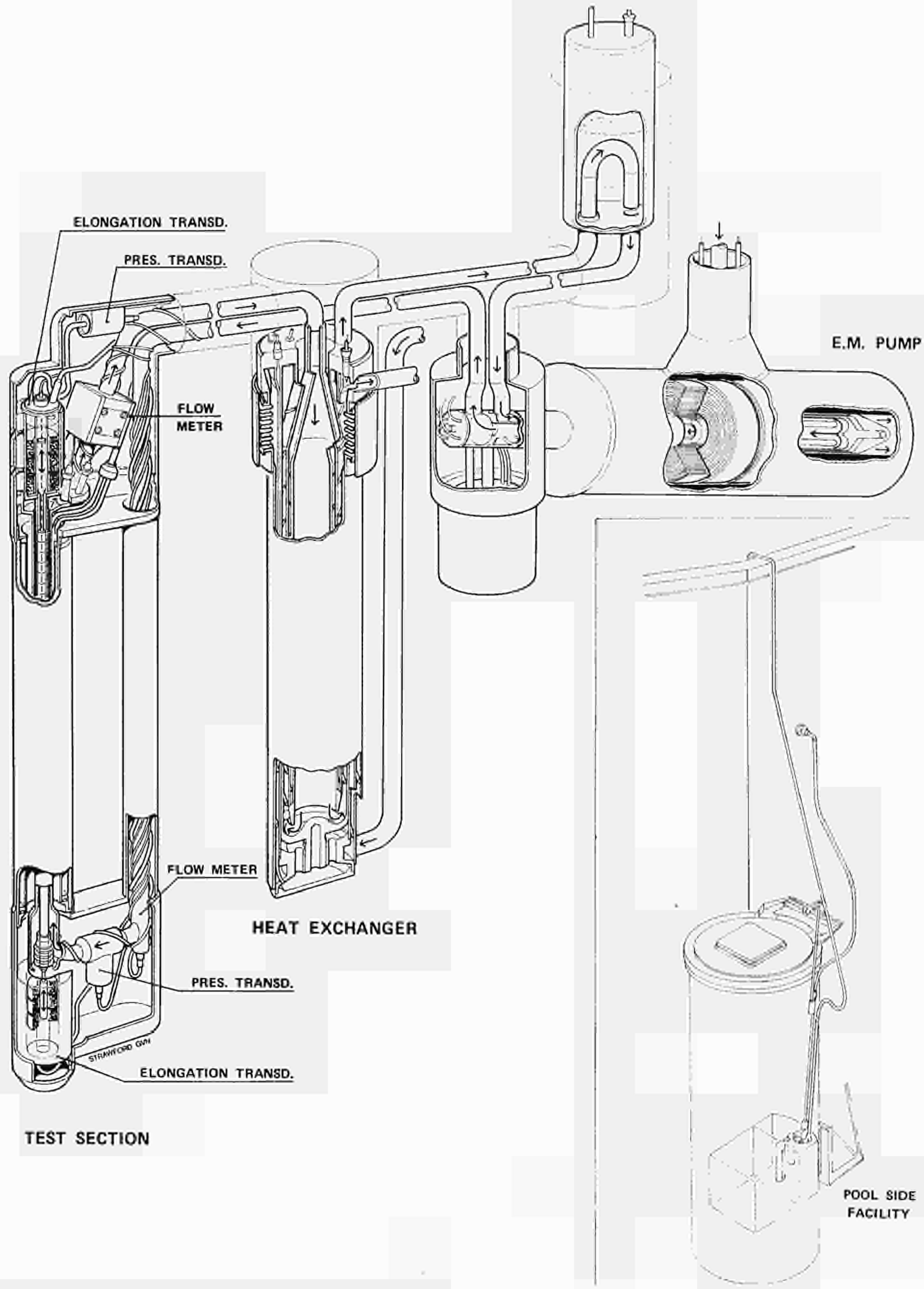


Fig. 55 R124, HFR TOP irradiation facility.
Schematic layout of the PSF loop capsule.

properly, only the excess BF_3 flowing back to the CaF_2 was not absorbed fast enough, due to slow diffusion into the CaF_2 powder. A small extension of the inlet line to the bottom of the CaF_2 container which will cause the gas to blow upwards and provide thorough mixing with the powder, will correct this problem.

Instrumentation.

Irradiation of elongation and pressure transducers in experiment R163, INTEC resulted in the choice of the Seybrook elongation transducers for TOP, as they proved far more reliable than the formerly chosen Schaevitz transducers. The Kaman pressure transducers also showed reliable behaviour.

Experiments in support of R124

R163 Instrumentation Test Capsule (INTEC)

The irradiation of the INTEC capsule was continued during the HFR-cycles 80-01, 80-03 and 80-04. During this irradiation the ECN pressure transducer, the Schaevitz LVDT and one heater element failed.

The failure of the ECN pressure transducer was caused by a leak in the transducer vessel. This leak was also responsible for some activity release during the pressure transducer calibration routine. Therefore no further calibrations were performed.

The malfunctioning of the Schaevitz XS-ZTR displacement transducer started during cycle 80-03 and was probably caused by short-circuiting effects in the primary transducer coil.

No explanation of the failure of heater element no. 1 can be given sofar.

The self-powered neutron detector electronic circuit was repaired before the start of cycle 80-01. The most important improvement appeared to be the addition of an electronic filter at the input gate of the current amplifiers. This eliminated the 300 kHz interference signal. In Fig. 57 the output signal of neutron detector no. 2 is compared with the output signal of a N17 reactor channel during a rapid reactor startup.

The test results obtained in HFR cycle 79-11 were evaluated and reported [4]. In Fig. 56 the sensitivity of the Kaman KL 1911 pressure transducer vs. irradiation time in cycle 79-11 is given. This sensitivity shows a slight decrease to max. 2.5°/o at the end of the cycle (corresponding with a thermal neutron fluence of approximately 10^{20} neutrons/cm^2).

Preliminary conclusions from the INTEC-experiment are:

- The Kaman KP 1911 pressure transducer performed excellently and showed little nuclear radiation sensitivity,
- The Sangamo F35 differential transformer also gave reliable results over the whole period of the experiment and showed

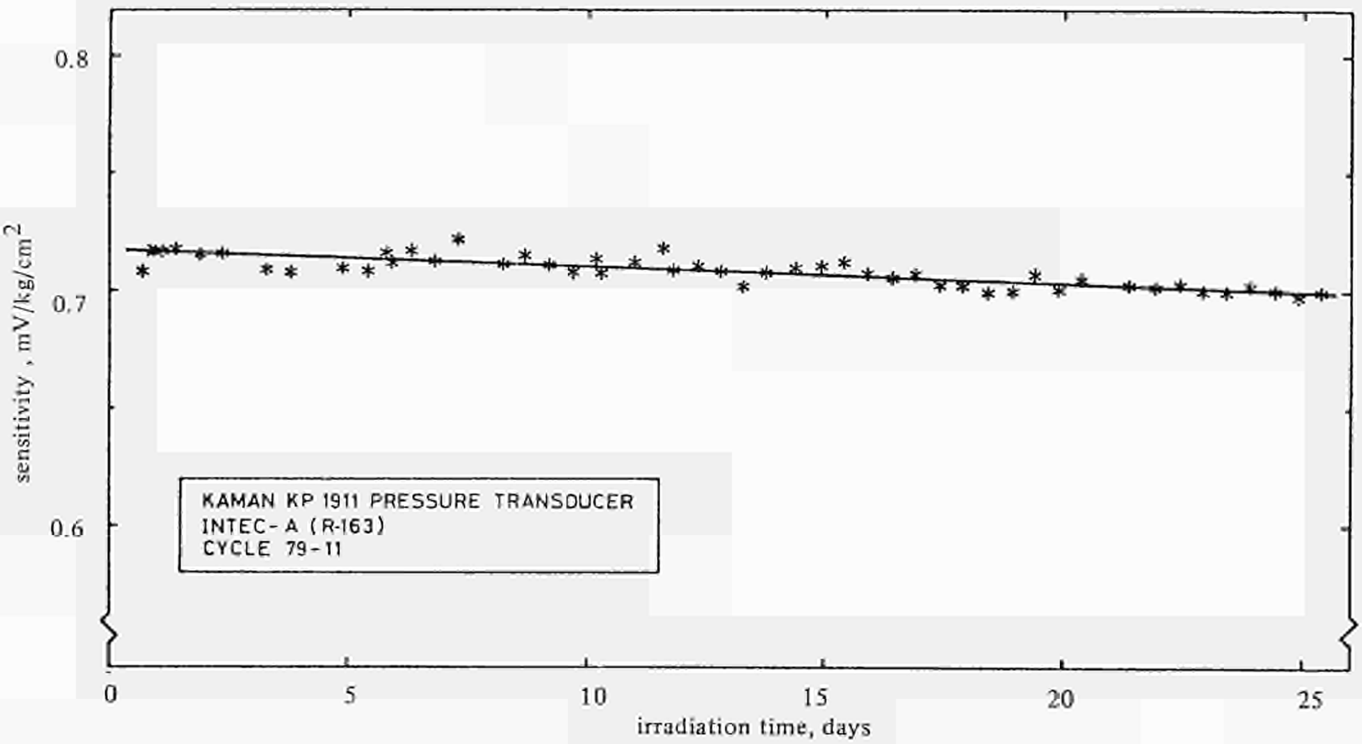


Fig. 56 Sensitivity of the Kaman KP 1911 pressure transducer vs. irradiation time.

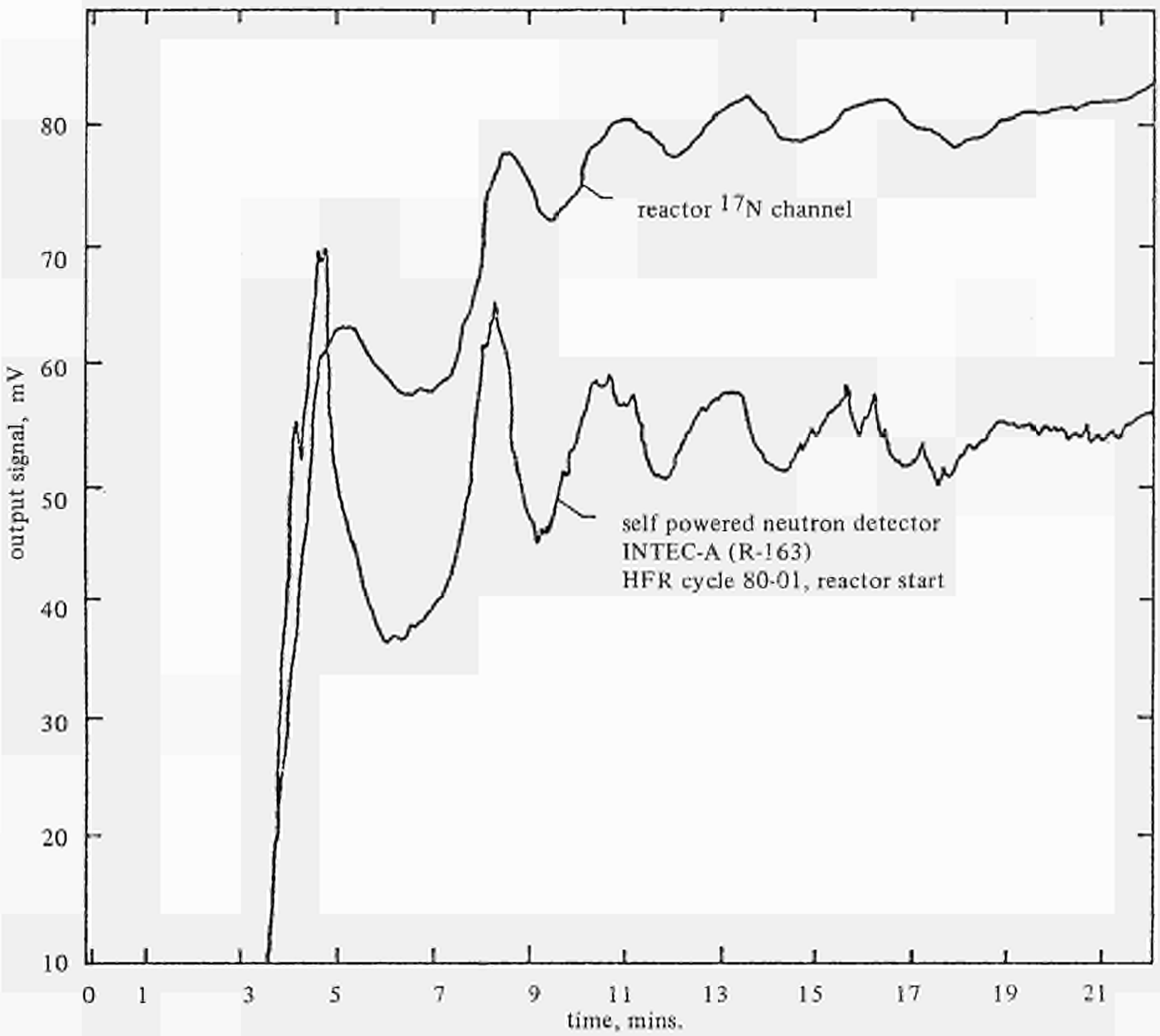


Fig. 57 Instrumentation test rig INTEC (R163). Signals of a Co self-powered neutron detector and of a reactor ^{17}N channel during a fast reactor start-up.

almost no radiation induced drift,

- the failure of the Schaevitz 100 XS-ZTR displacement transducer, combined with the discouraging results of the out-of-pile tests, makes this type of transducer appear less suitable for in-core application.

As a direct result it was decided to replace the Schaevitz transducer in the HFR TOP experiment by a modified version of the Sybrook LD10 (Sybrook transducers are identical to the Sangamo displacement transducers).

b) Operational ("mild") Transient Experiments

This class of irradiation tests aims at investigations into fast reactor fuel pins under slight transient cooling perturbations and/or reactivity increases as can be expected to occur rather frequently during normal operation of future large fast breeder reactors. The overall experimental plan includes power ramps on pre-irradiated fuel pins, requiring a remote encapsulation facility (see paragr. 2.3.3.6), as well as some work on carbide fuel.

D173 DUELL.

Short fully-enriched mixed oxide fuel pins are power ramped in twin pool side facility capsules. The first test was carried out in June 1979. Six fuel pins have been irradiated during the period January to June 1980. These were start-up ramp tests with a preconditioning at 80^{o}/o of full power during 24, 2 and 10 hours. All six irradiations complied with the required maximum linear power of 565 W/cm and fuel can temperatures from 400^{o}C (bottom) to 600^{o}C (top). Fig. 58 gives the power history of DUELL-14.

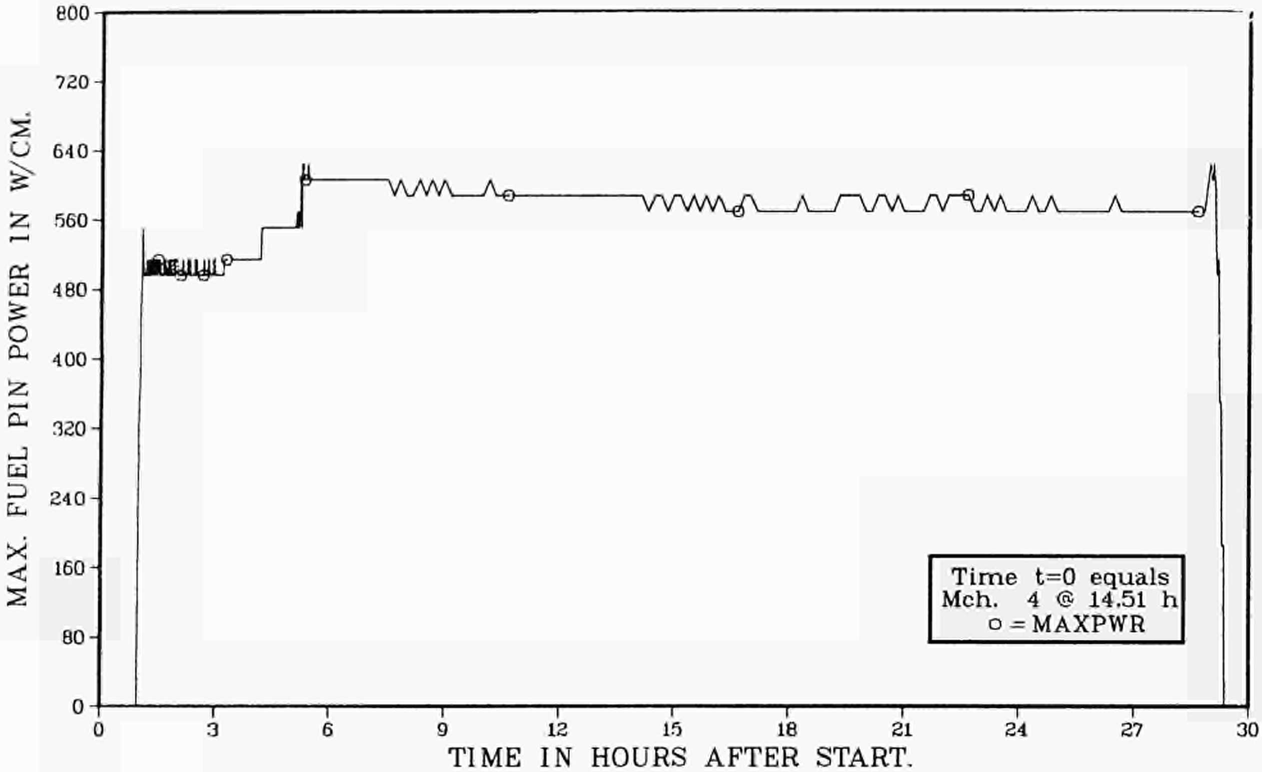


Fig. 58 Transient experiment D173-14. Power history, cycle 80.03.

The present working schedule includes one more irradiation of two pins in 1980 and a series of 4 x 2 pins in early 1981, which are prepared at present. Fig. 60 shows components of DUELL 18 during assembly.

The analytical means of mathematical modelling were able to comply with a strong circumferential power gradient, observed on the first DUELL pins. For future experiments an attempt has been made to improve the neutron flux distribution by means of selected outer absorbers (See paragr. 2.1.3.6).
Fig. 59 shows the capsule carrier with the absorbers.
A dosimetry experiment is being prepared at present and scheduled for irradiation in August 1980 in order to verify the calculated results. A time schedule of some DUELL experiments (10 to 27) can be seen in Fig. 63.

D183 KAKADU

Another twin PSF capsule has been designed to accommodate long fuel pins for transient testing (overall length up to 1.6 m).
Components for 4 test vehicles (8 fuel pins) had been manufactured but the irradiations have been delayed because of the above-mentioned flux gradient problem. The programme schedule presently forecasts the first irradiation in October, 1980.

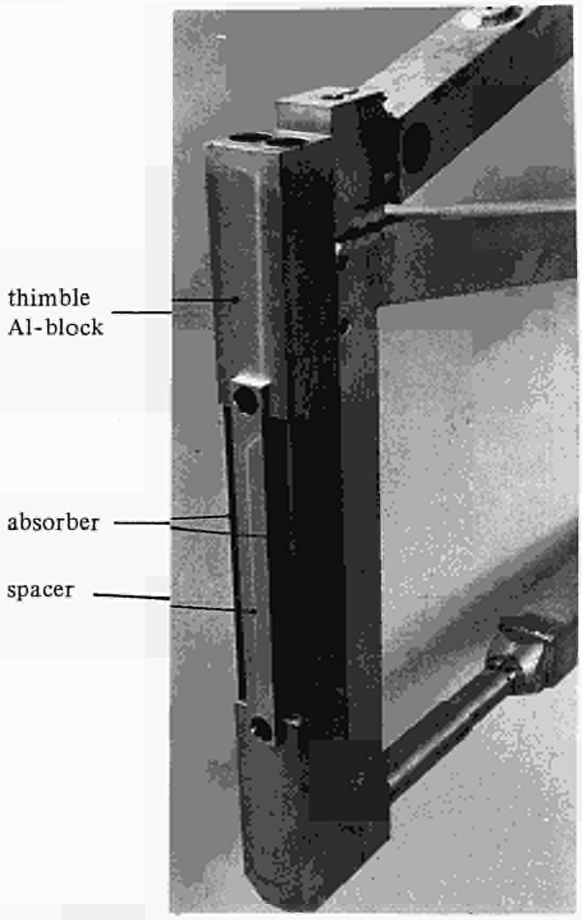


Fig. 59 DUELL series D173. Modified capsule carrier.

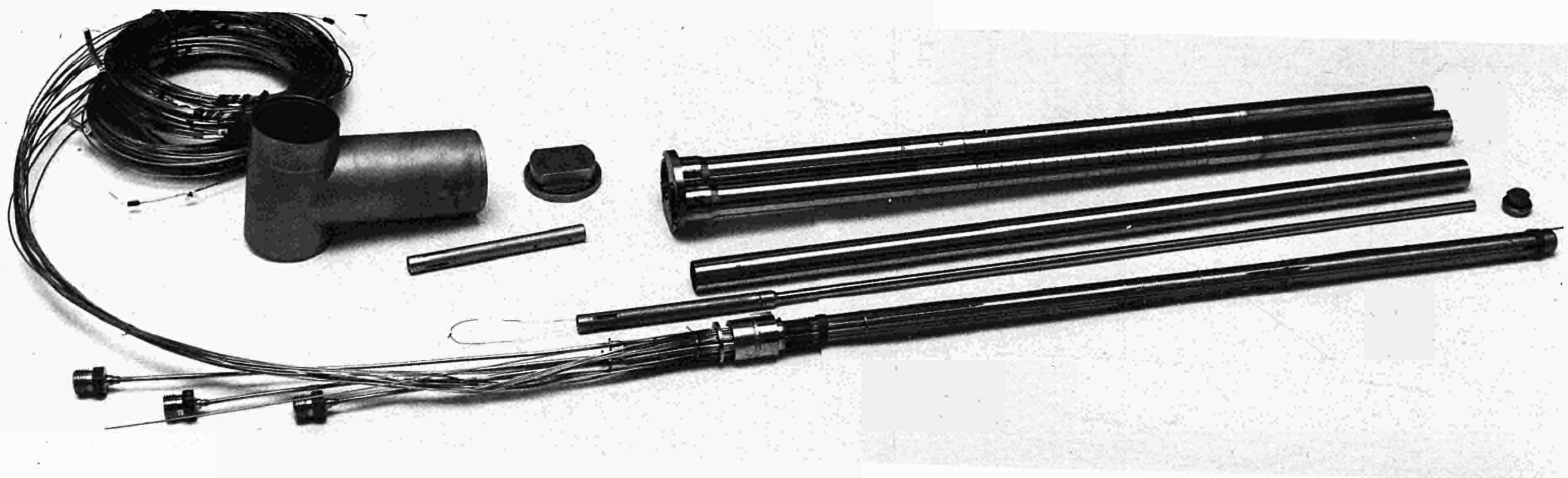


Fig. 60 DUELL series D173. Components of DUELL 18 under assembly.

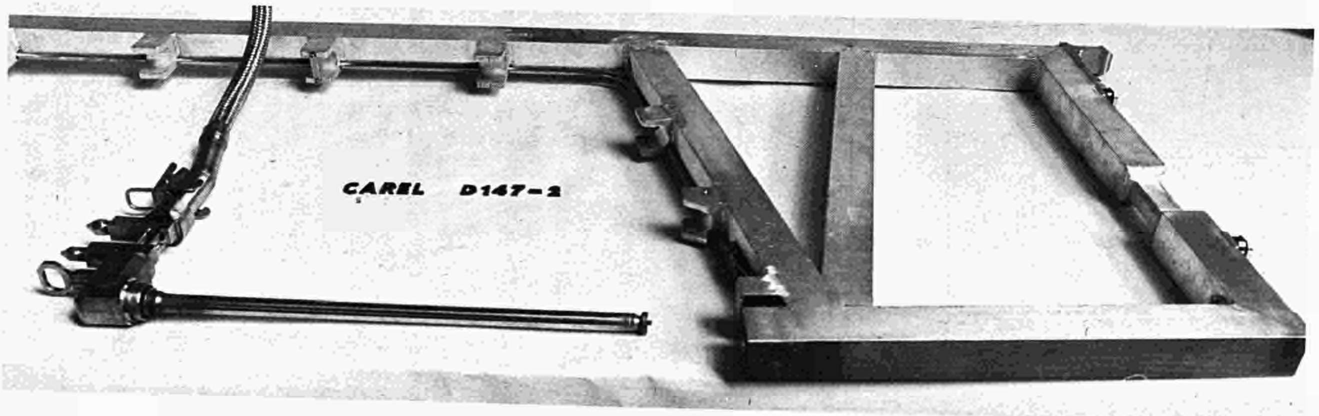


Fig. 61 D147 "CAREL". Capsule and carrier after assembly.

D147 Carbide fuel testing in a PSF capsule (CAREL)

The first irradiation of this cadmium-screened facility had to be interrupted shortly after its start in July 1980. The initial idea of repairing the damaged capsule has been abandoned.
The second experiment of this series has been assembled from March until May 1980 and is ready for in-pile operation in cycle 80-07. The cadmium screen has been modified, using strips instead of a perforated tube, to increase the linear power with respect to the preceeding experiment from 600 to 800 W/cm. Fig. 61 shows the completed capsule and carrier.

D148 Carbide fuel testing in a TRIO capsule (CATRI)

The irradiation of the first three fuel pins has been completed successfully after 13 cycles (330 full power days) [5].
A burn-up of max. 87.000 MWd/t has been achieved without any failure of the pins or the irradiation capsules. Fig. 62 shows a neutron radiography picture of one of the pins after irradiation. Dismantling and PIE are being prepared.
The follow-up experiment, again with three fuel pins is at present being assembled. The beginning of the irradiation is scheduled for September 1980.

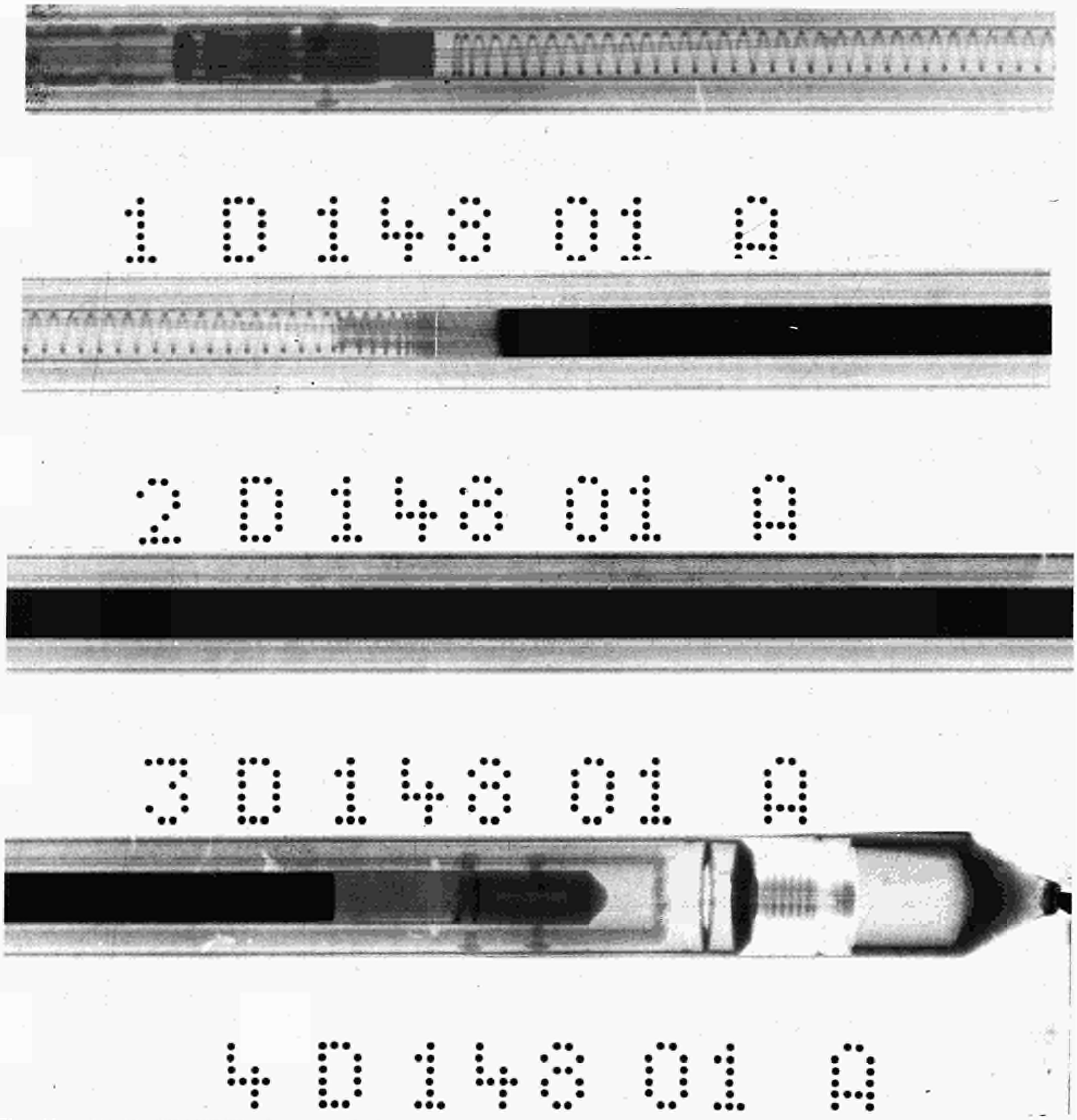
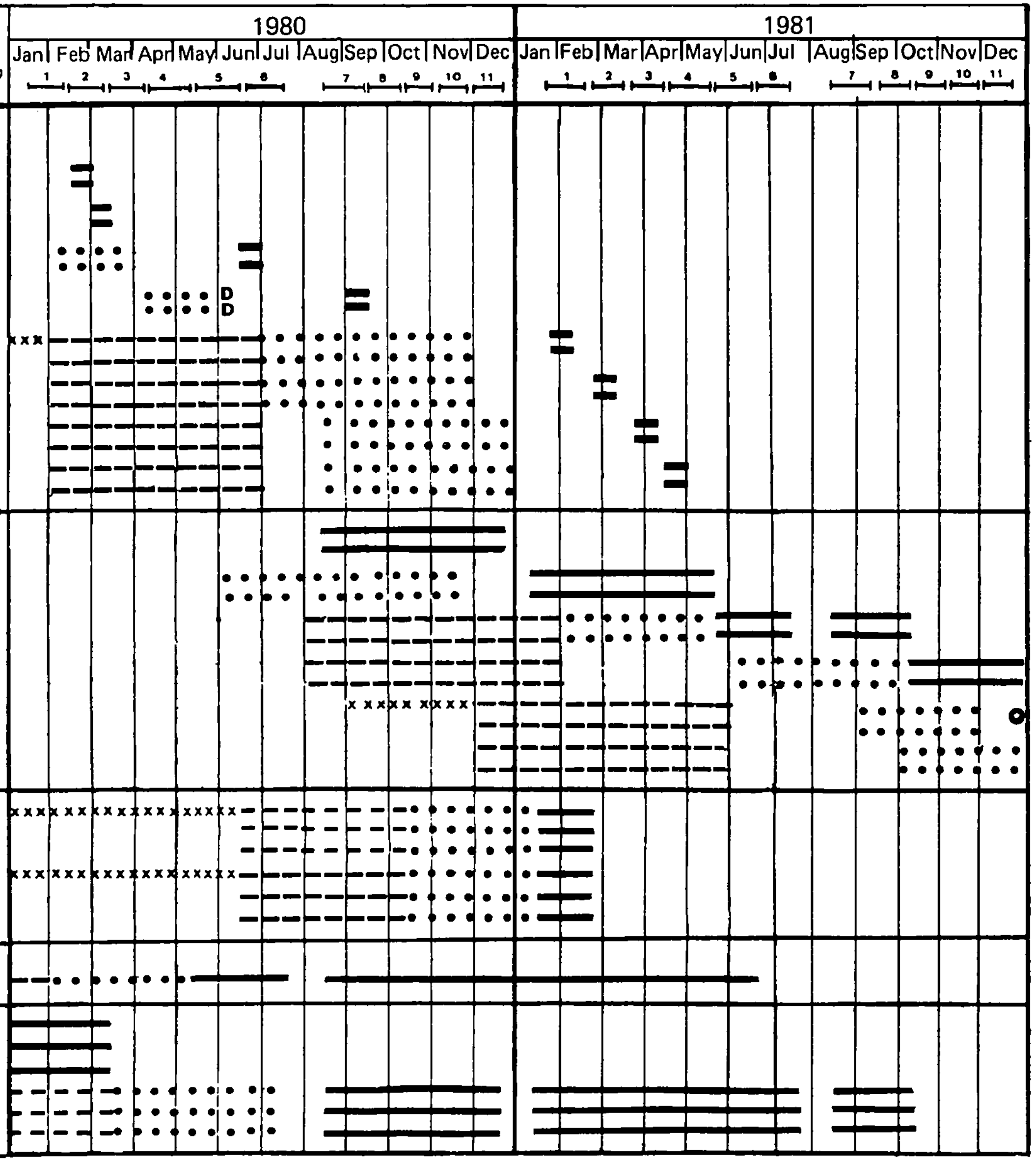


Fig. 62 D148, "CATRI". Post-irradiation neutron radiograph of one of the specimen carriers.

PROJECT	RIG Nº	POS HFR	DRWG GCO	PIN Ø	DAYS IRRAD
D.173 DUELL	10	5	35 478 C	6	12
	11	5	–	6	12
	12	5	–	6	4
	13	5	–	6	4
	14	5	–	6	3
	15	5	–	6	3
	16	5	–	6	2
	17	5	–	6	2
	18	5	–	6	3
	19	5		6	3
	20	5	39 838	7.6	4
	21	5	–	7.6	4
	22	5	–	7.6	4
	23	5	–	7.6	4
	24	5	–	7.6	4
	25	5	–	7.6	4
	26	5	–	7.6	4
	27	5	–	7.6	4
D.183 KAKADU	21	7	38 801	6	100
	22	7	–	6	100
	23	7	–	6	100
	24	7	–	6	100
	25	7	38 213	7.6	100
	26	7	"	7.6	100
	27	7	–	7.6	100
	28	7	–	7.6	100
	50	7			10
	51	7			10
	52	7			10
	53	7			10
D.184 MONOX	11	G7		6	100
	12	–		6	100
	13	–		6	100
	14	–		7.6	100
	15	–		7.6	100
	16	–		7.6	100
D147 CAREL	1	6	28742	8.5	350
	2	4	37 /96	8.5	
D 148 CATRI	11	G3	31 250	8.5	350
	12		–	8.5	
	13		–	8.5	
	14	G3	31250 F	8.5	350
	15		–	8.5	
	16		–	8.5	



KfK
LMFBR PROJECTS. Time schedule of phase 2 — Prevision 05.05.1980

O OPERATION EUROS CELL — | Ordered — Design × × × — Machining — — — Assembly • • •

Fig. 63 Detailed schedule of "mild" transient and carbide fuel irradiations.

D184 MONOX and TRIOX. Oxide fuel overpower experiments

The initial requirement to detect the power to melting in oxide fuel pins will be met by a fast flux short time (FFST) experiment, where a simple pin and capsule (MONOX) is irradiated in the C7 position in a reentrant thimble with pool water cooling and cadmium screen. The capsules are loaded and unloaded during reactor operation. Figs. 64 and 65 show a horizontal and a vertical cross-section of the experiment.

Additionally an overpower experiment will be performed to assess the overpower behaviour (close to melting) of oxide fuel pins. Three pins and capsules (TRIOX) are irradiated simultaneously in the C7 position in a TRIO type device, similar to that of the D148 project, under cadmium in a fast flux long time (FFLT) thimble, cooled by the primary coolant water and not reloadable during reactor operation.

Both, MONOX and TRIOX experiments and FFST and FFLT thimbles are at present being designed and fabrication and assembly are scheduled for the second half of 1980.

The fuel pins and the irradiation programme are prepared by KfK Karlsruhe.

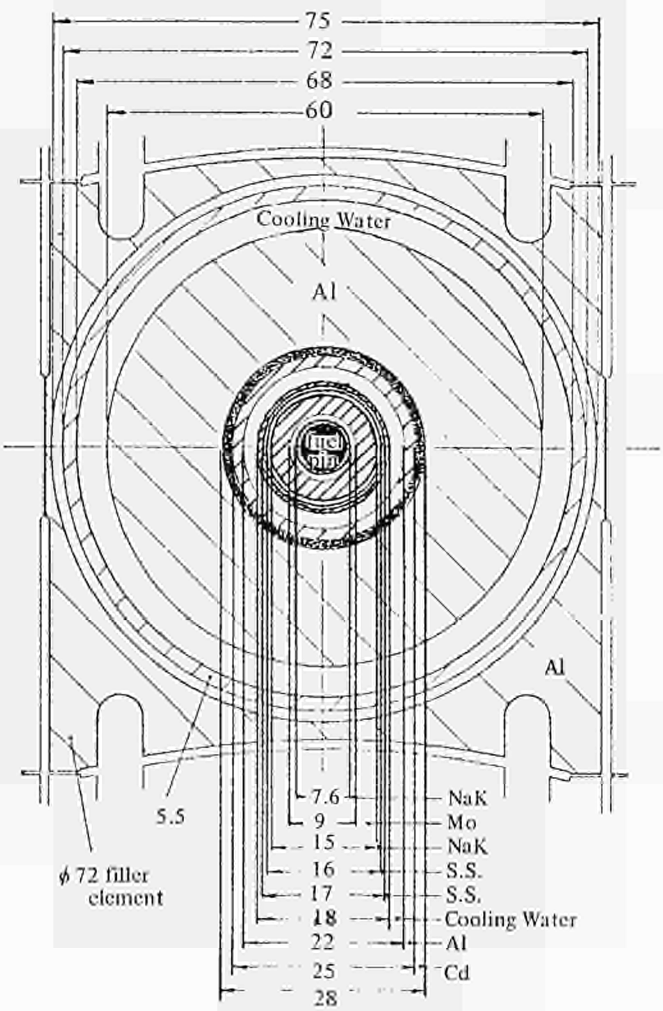


Fig. 64 D184, MONOX (FFST). Horizontal cross section.

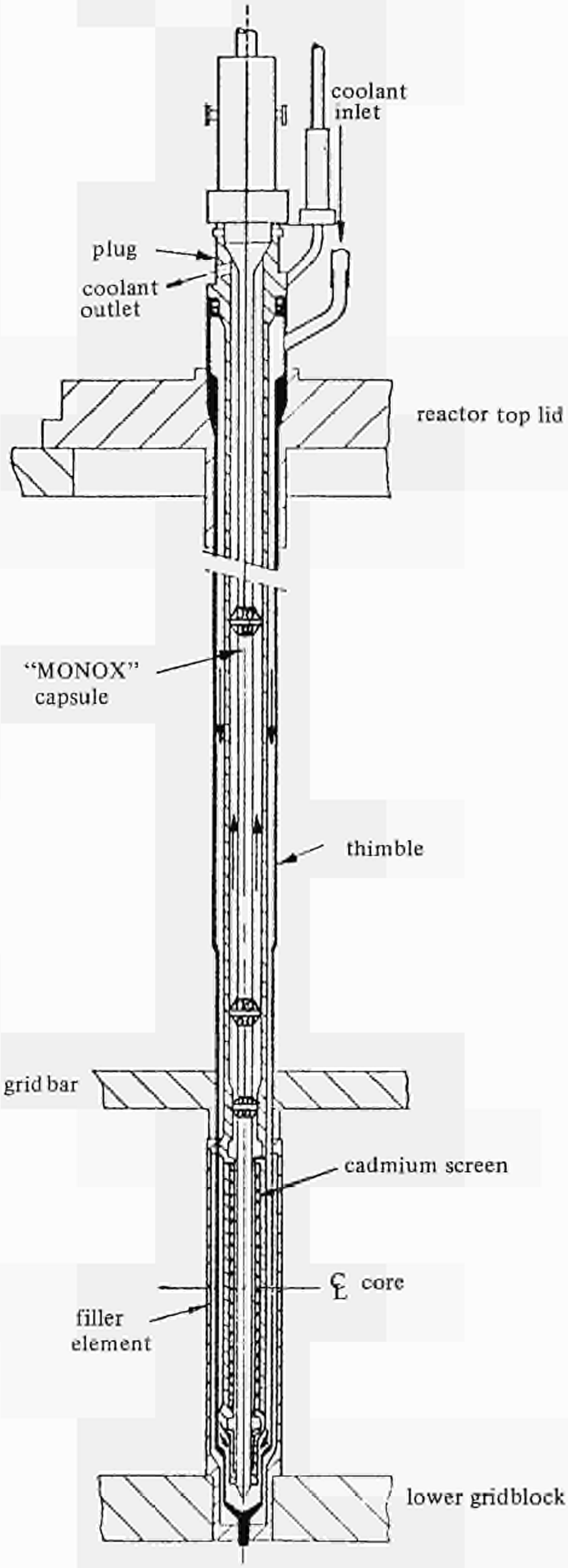


Fig. 65 D184, MONOX (FFST). Vertical cross section.

c) Other Types of Irradiations

E170 POCY, POwer CYcling experiment with fuel pin profile gauge.

In this experiment (JRC Karlsruhe designation - POCY 02, 03), mixed carbide fuel pins are power cycled between 600 and 1000 Wcm^{-1} at midwall clad temperatures between 400 and 500°C.

At intervals the fuel pin profile has to be measured. The irradiation device chosen is a PSF chimney capsule in which the fuel pin can be lifted and passed through a diameter gauge incorporated into the device above core top level.

This experiment has now operated within the designed parameters for over two hundred days. Power levels of between 150 and 1000 Wcm^{-1} have been obtained, and the fuel pin profile has been periodically recorded as required by the sponsor, over a number of discrete angular positions. The fuel pin profile traces clearly show the swelling phenomena (Fig. 67).

The power and temperature history of a typical cycle is shown on Figs. 66 a and b.

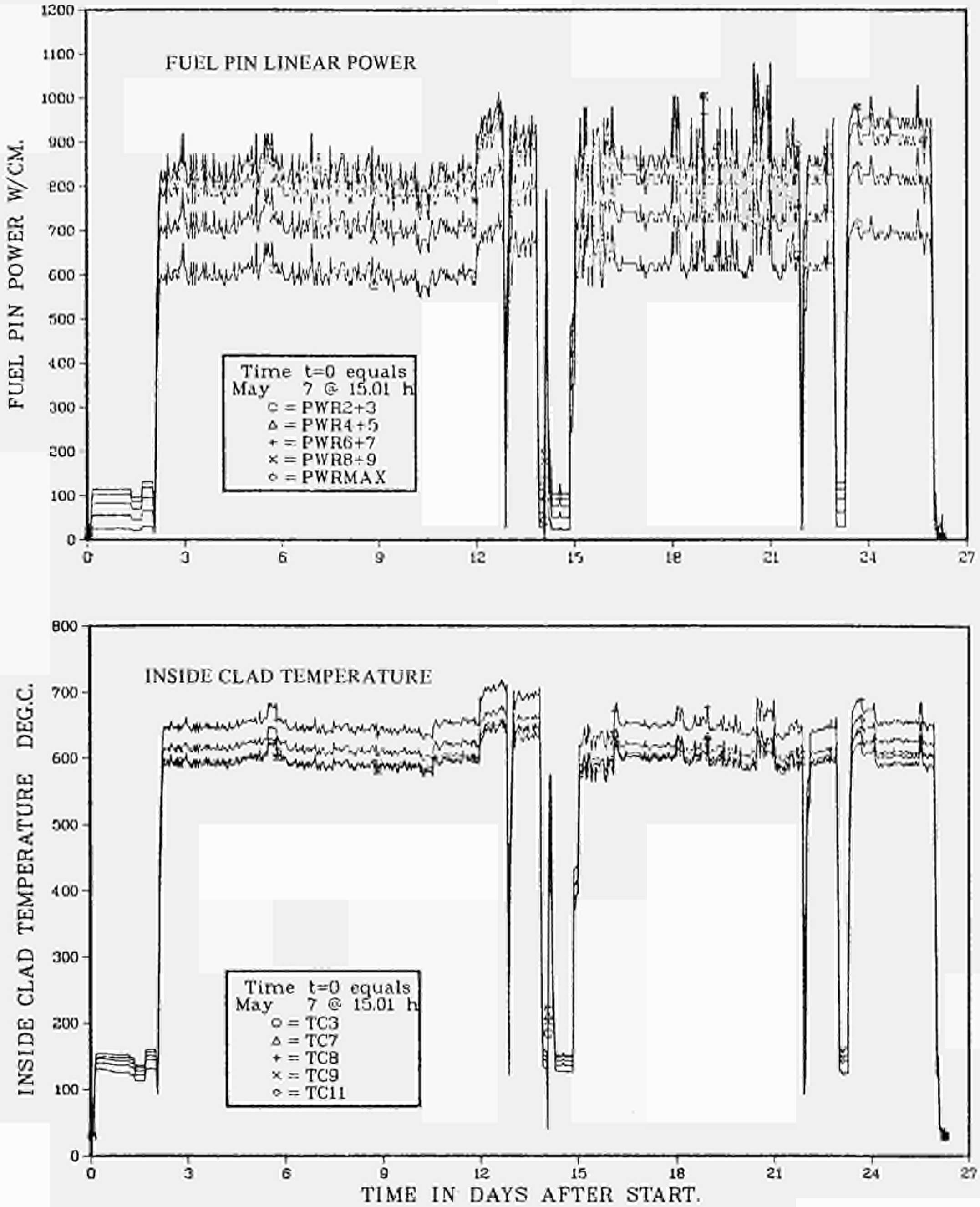


Fig. 66a E170-1, POCY. Performance diagrams during cycle 80.05. Main parameters.

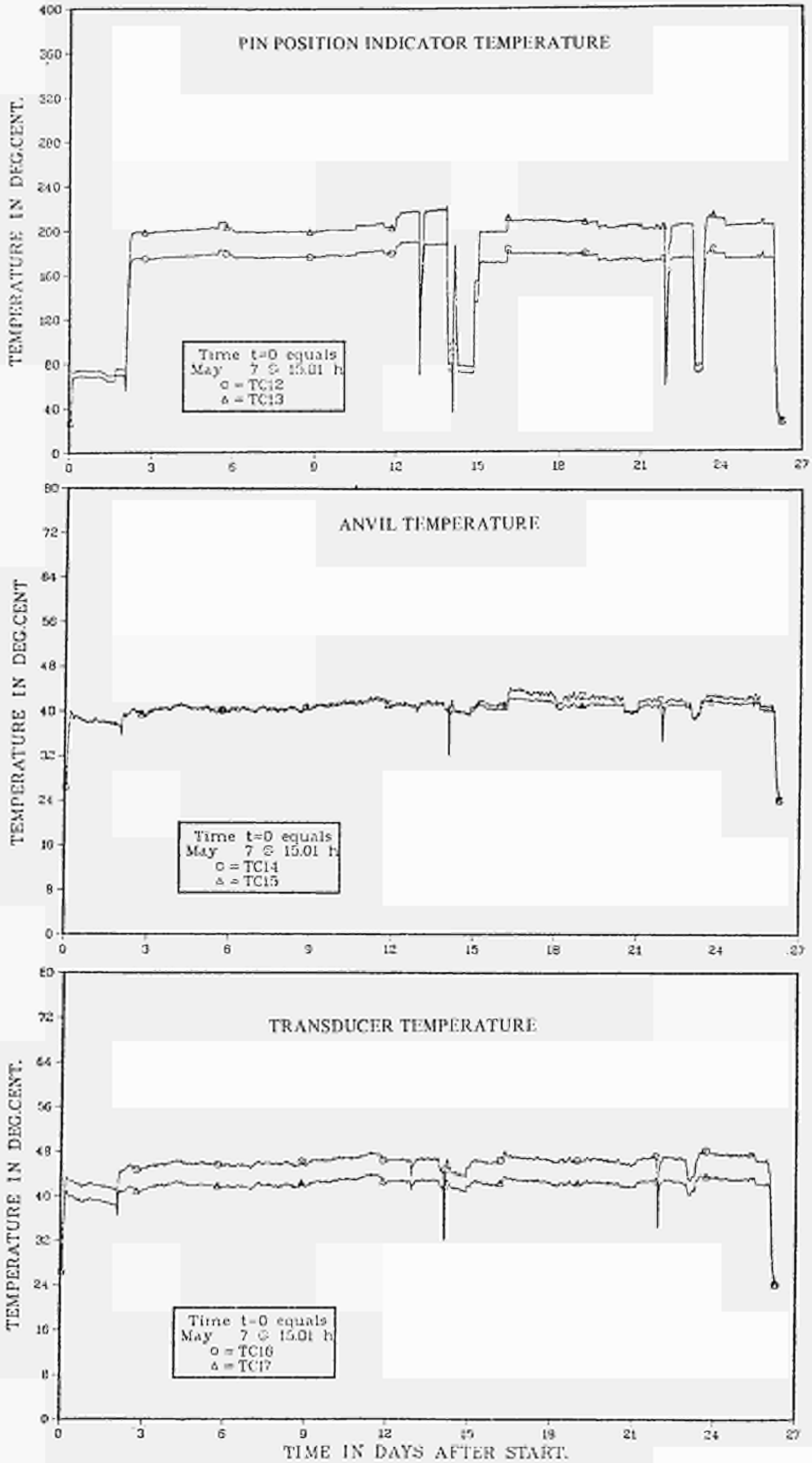


Fig. 66b E170-1, POCY. Performance diagrams during cycle 80.05.
Auxiliary parameters.

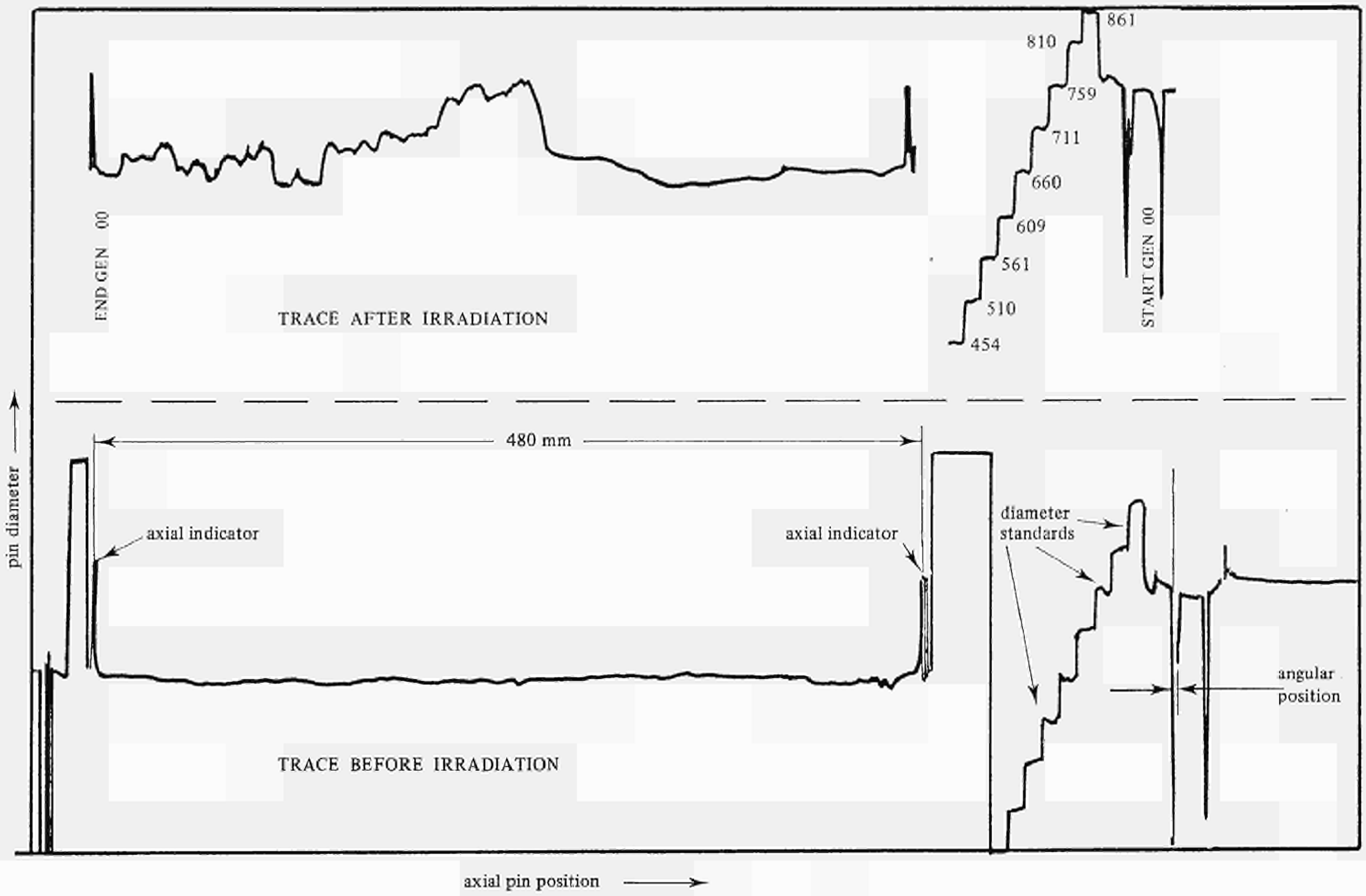


Fig. 67 E170, POCY. Fuel pin diameter traces before and after irradiation.

E172 Oxide fuel corrosion studies (CORROX)

The irradiation of this experiment has been completed in December 1979. During the dismantling in March 1980, no evidence came up to explain the erroneous temperature behaviour during in-pile operation.
The fuel pins have been dispatched to JRC Karlsruhe. The microscopic examination of 6 sections of the outer and inner capsule tube is still pending, in order to assess any change in gas gaps width. Neutrographic and visual inspection of the fuel pin did not reveal any defects of the fuel can.

D191 Fuel creep rig design study

A proposal has been made for a device to measure the fission enhanced creep of a small stack (20 mm) of fuel pellets, in a PSF position. The maximum power will be in the order of 3000 Wcm^{-3}, with fuel temperatures of around 1000°C. The sample will operate with up to 70 MPa of compressive load.
The proposed design stems from an earlier experiment (E093) carried out in HFR in 1972/73 (Fig. 68).

E084-07 TRANSON

As a continuation of earlier TRESON experiments this new irradiation for JRC Karlsruhe will incorporate an improved ultrasonic fuel centre line thermometer, and transient power operation.

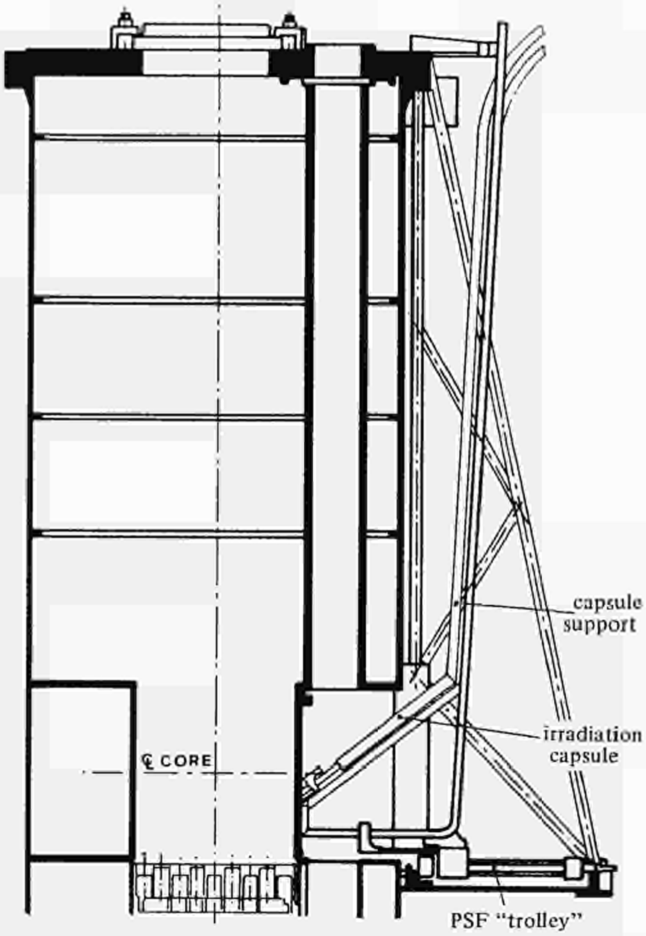
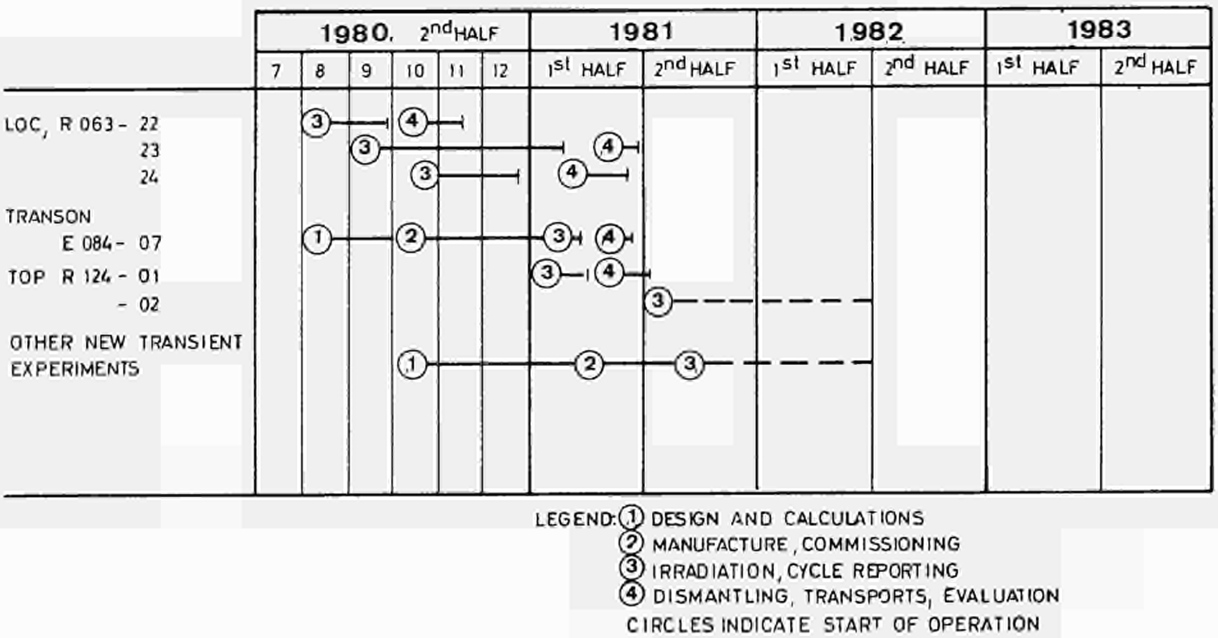


Fig. 68 Fuel creep rig design study. Capsule arrangement in the Pool Side Facility.

Fig. 69 FAST REACTOR FUEL IRRADIATIONS (EXCEPT D 147, D 148, D 173, D 183, D 184)
SURVEY OF FUTURE ACTIVITIES



References to paragraph 2.2.3.5

1. Bestralingsverslag SHOT capsule R054-F37.
   A.J. Nolten,
   R.A. memo 80-8 (February 1980)
2. Bestralingsverslag SHOT capsule R054-F38.
   A.J. Nolten,
   R.A. memo 80-10 (March 1980)
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4. Bestralings- en evaluatieverslag van de INTEC-A capsule gedurende HFR cyclus 79-11 (Exp. R163).
   W.J. Stam, H. Smedinga,
   ECN 80-89 (May 1980)
5. The in-pile performance of D148-01. Mixed carbide irradiation of a TRIO type rig, CATRI.
   R. Holt,
   Technical Note P/10/80/16 (April 1980).

2.2.3.6 Miscellaneous

a) HFR vessel material surveillance irradiations (RX092)
Specimens have been irradiated in position E3 during two reactor cycles.

b) Gamma calorimeter (RX161)
Reporting is contained in paragraph 2.1.3.3 of this report.

c) Radioisotope production and activation analysis
- Utilization of standard isotope and rabbit facilities. A survey is given in Table 14 below.
- Particular remarks :

ER136-03 ^{99}Mo production from fissile targets (FIT)

The new in-core FIT, developed for the irradiation of 3 targets with diameter 22 mm and 4 gr ^{235}U, 93°/o enriched, is in irradiation. During the reporting period four irradiations have been carried out.
A new study, FIT-05, is at present under design, using the same system as FIT-03 but three legs (modified TRIO-capsule) instead of a REFA capsule. This FIT-05 allows the irradiation of 3 x 3 targets simultaneously, using the primary cooling system of the reactor, and the loading and unloading of the targets during reactor operation.
This TRIO-FIT has been designed as a replacement of the PSF-FIT rig (FIT-04).
At present two FIT irradiations are carried out weekly and transported with two containers to the reprocessing plant.

ER144 High Flux Facility for Isotopes (HIFI)

During the reporting period the HIFI-01 device has been replaced by HIFI-02. This replacement was necessary because of mechanical problems on device no. 1.
The irradiation of the iridium capsules in HIFI-02 continues normally.

Table 14 Utilization of standard isotope and rabbit facilities.

Code of the facility	Facility	Number of irradiations				
		first half 1978	second half 1978	first half 1979	second half 1979	first half 1980
PR 1	Pneumatic rabbit system	772	501	723	502	629
HR	Hydraulic rabbit system	6	11	36	34	26
ER6	Pool side isotope facility (PIF)	60	23	60	– *)	–
ER8	High flux poolside isotope facility (HFPIF)	15	9	30	58	49
ER70	Poolside rotating facility (PROF)	44	63	76	53	94
ER7-2	Reloadable isotope plug (RIP)	14	12	24	– **)	–
ER90	Reloadable isotope facility (RIF)	262	191	287	232	209
GIF	Gamma irradiation facility	*16*	*1*	*12*	*1*	*10;*
ER136	Fissile irradiation facility (FIT)	12	–	27	12	10
ER144A	High flux facility for isotopes (HIFI)	–	10	89	83	31
Total (except GIF)		1135	820	1352	974	1048

*) facility removed from PSF table
**) facility removed from the reflector.

ER179 ^{99}Mo production of fissile targets (MOLY)

A new PSF facility, for the irradiation of fissile targets in form of plates has been developed. Each plate, with the dimensions 220 x 40 x 1.3 mm, contains 4 gr ^{235}U, 93°/o enriched. Eight plates as maximum will be irradiated simultaneously during about 6 days and transported after 24 hours cooling to the reprocessing plant.

ER179 - ER8 Special Design

Special design work was carried out to use PSF9, which is partially occupied by the neutron radiography camera, for HFPIF (see paragr. 2.3.3.7 of this report).
A special carrier was developed (Fig. 88) which allows to irradiate the fissile targets of the MOLY project and at the back of the MOLY holder two pipes for the HFPIF capsules. The cooling water for the MOLY project is supplied by the PSF cooling system.

d) Special irradiations

E149 BONI, Boron silicate pellet irradiation.

The purpose of this series of experiments is to simulate long term radiation damage effects in vitrified fission waste by short-term fission product recoil damage (JRC Ispra programme) The third experiment, planned for 10 reactor cycles (260 days), started irradiation in October 1979, and has since then been operated without problems. The irradiation has taken place in the position PSF 3 until the cycle 79-11.
From cycle 80-03 the experiment occupies position PSF 1. Due to the proximity of the neutrography camera, the construction of a new support was necessary (see paragr. 2.3.3.7 and Fig. 88).

2.3 General Activities

2.3.1 Objectives

Considerable effort has to be placed into keeping equipment and competence on the required level. The general activities within the HFR project include:

- operation and maintenance of ancillary services and laboratories (e.g. workshops, hot laboratories, general purpose control equipment, computing facilities),
- design studies and development of new irradiation devices,
- irradiation technology and other research,
- programme management.

i.e. support work not directly linked to a specific irradiation experiment.
About 8°/o of the annual HFR budget and 30°/o of the scientific-technical JRC staff capacity are allocated to general activities.

2.3.2 Methods

A total of 12 to 15 general activities are defined for each year, according to their nature (see above), and manpower/money are allocated. For the period under review these have been:

a) *operation and maintenance*

- testing and commissioning
- experiment operation
- dismantling cell
- data acquisition and computing facilities
- neutron radiography
- post-irradiation examens
- assembly, workshops

b) *design studies and development*

- standard irradiation devices
- in-pile instrumentation
- transient condition facilities
- feasibility studies
- creep facilities
- computer codes
- LWR irradiation facilities

c) *irradiation technology and other research*

- reactor upgrading
- development and design of the new reactor tank
- reduced enrichment research and test reactor fuel studies

d) *programme management*

- documentation and editing
- CPM planning of irradiation experiments
- reactor utilization management
- working groups, conferences
- ACPM meetings

2.3.3 Results

2.3.3.1 Experiment Control Installations
(Proj. 310)

The cooling water flow measurement has been improved by a new equipment consisting of a differential sensor based on the principle of the dynamic pressure measurement, and of an electronic differential pressure transmitter. The output signal is in mA.
The advantage is a combination of accuracy (about $1^{o}/o$) with high resistance against impurities. Until now four water supply systems are equipped with the system.

A small additional crane has been mounted near the reactor pool edge for easier loading/unloading of irradiation capsules. A storage problem has arisen in pool nr. 3 (next to the dismantling cell) due to an inadequate flow of irradiated capsules to the ECN Hot Cells (LSO). Fig. 70 gives an impression of the crowded pool. Corrective measures could be initiated towards the end of the reporting period.

2.3.3.2 Testing and Commissioning (Proj. 300)

A particularly high number of specimen carriers (39) and complete irradiation devices (7) have been tested, and commissioned for transfer to the reactor (Fig. 71).

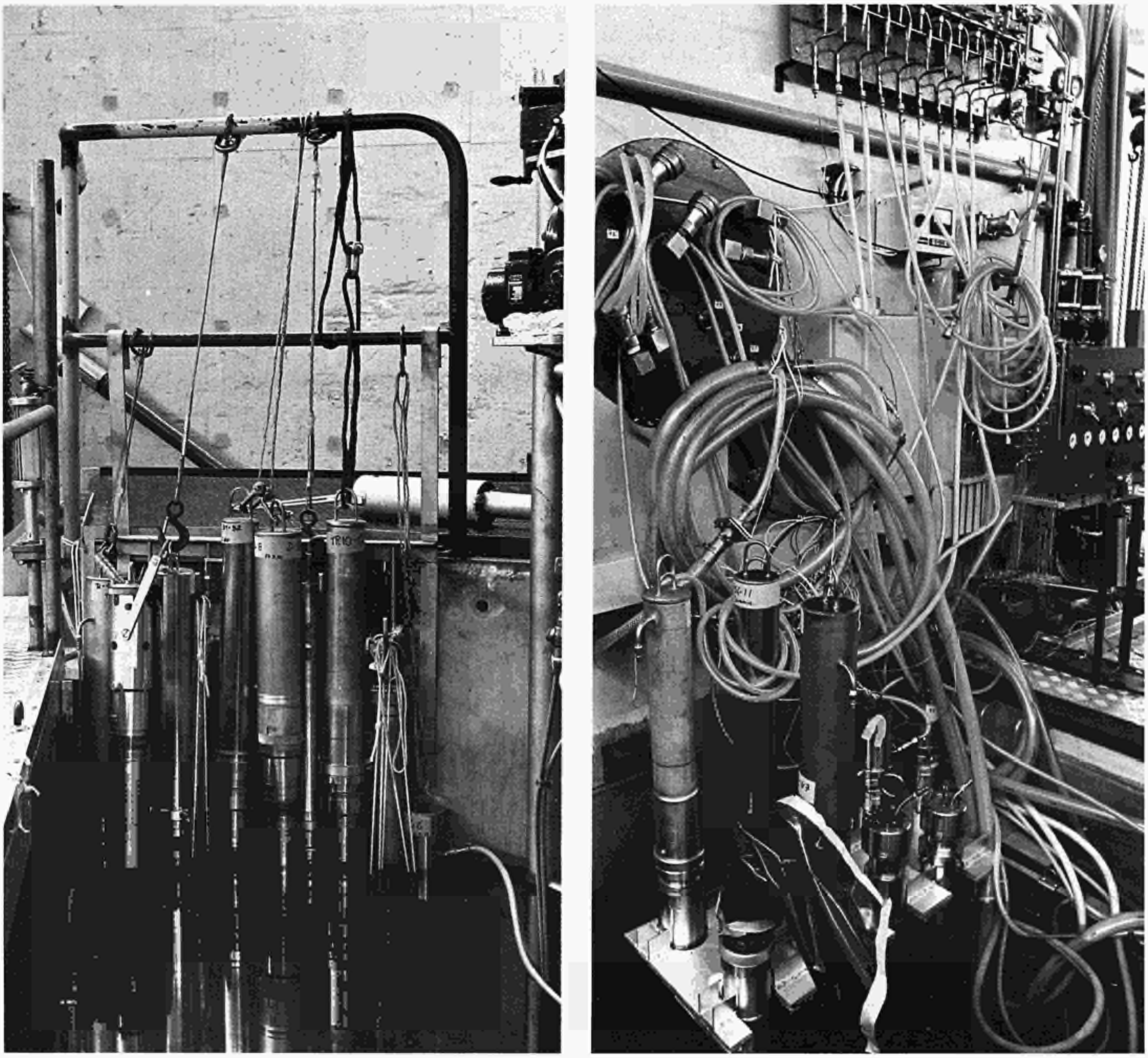


Fig. 70 Irradiation devices in temporary storage in HFR pool nr. 3.

Fig. 71 Testing and commissioning of HFR irradiation devices.

2.3.3.3 **Dismantling Cell** (Proj. 330)

The horizontal posting facility has been replaced by a completely re-designed installation (see Fig. 73).
During a major in-cell revision a new type double cover sluice was installed on the inner side of the new horizontal posting facility (see Fig. 74).
The new facility is now operational.
In order to safely handle transport containers in front of this posting facility the HFR hall crane has been modified.

In January 1980, the special shielded transport container for on-site nuclear transports (WII) was damaged by an accident, with the consequence of blocking nearly all DM cell work. An order for a new cask (W III) was placed and repair work of WII was started immediately. The repair could be accomplished within six weeks.
The number of on-site transports became so high that in future twin transports of WII and WIII will take place on a special carrier which has to be developed.

The cell-team provided, next to a 2 weeks' revision of the cell, the following services during the reporting period:

- dismantling of 25 specimen carriers,
- preparation and surveillance of 17 on-site, 13 waste, and 4 external transports,
- inspection of a damaged filler element (see Fig. 72),
- disassembly of 5 obsolete capsules,
- preparation for waste disposal (crushing) of 35 HFR fuel element end fittings.

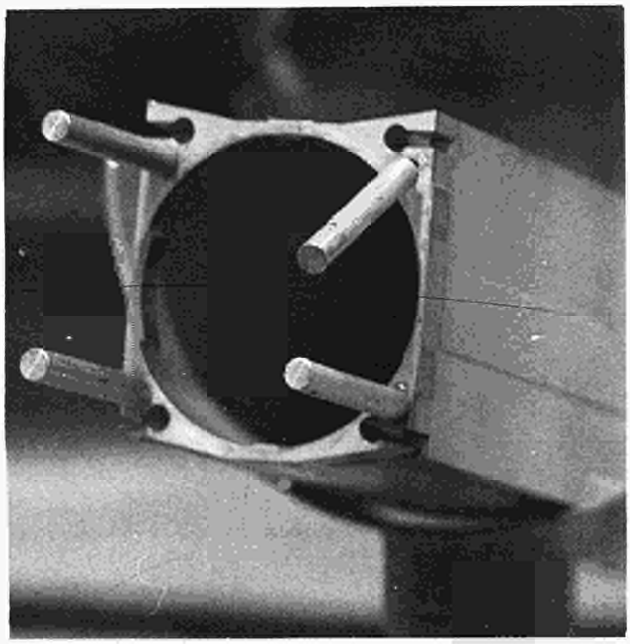
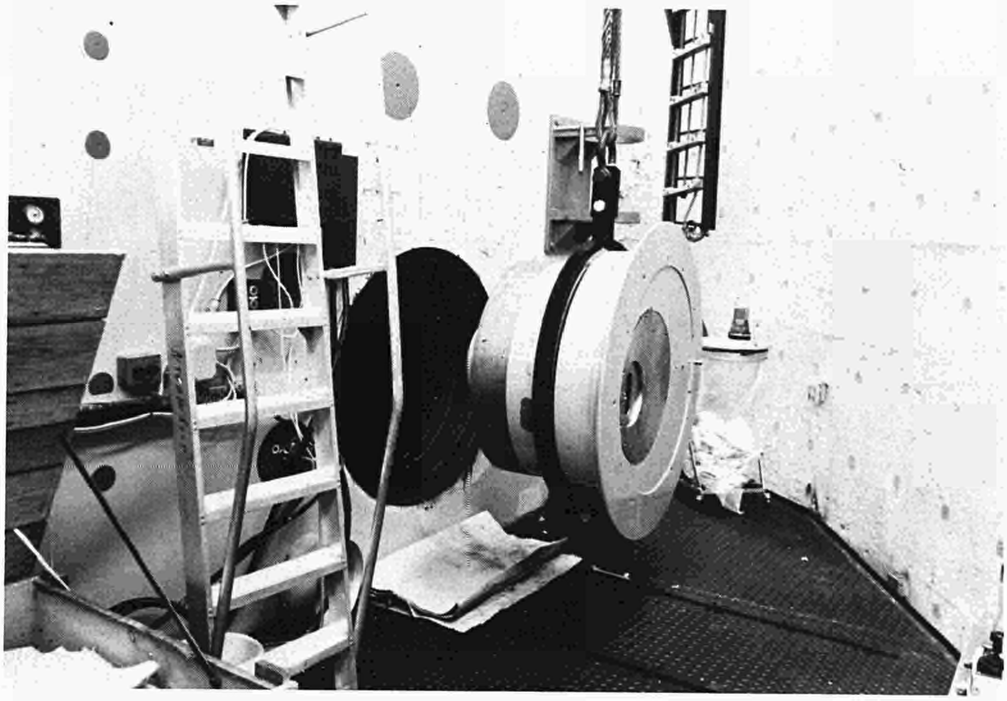


Fig. 72 HFR Dismantling Cell.
Inspection of a damaged filler element.



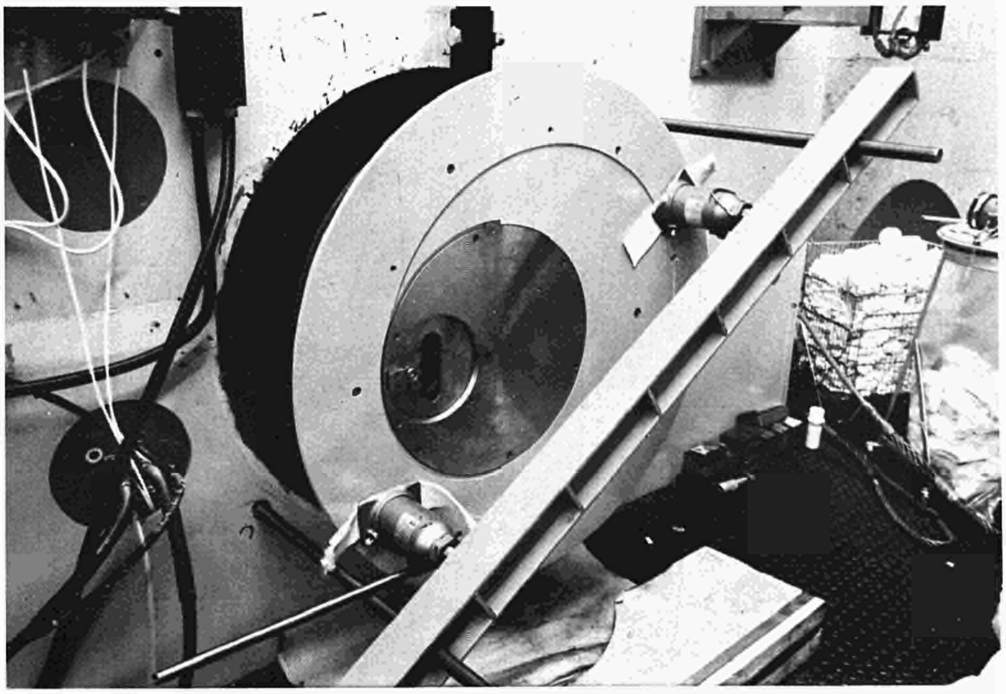


Fig. 73 HFR Dismantling Cell.
Installation of the new horizontal posting facility, April 1980.

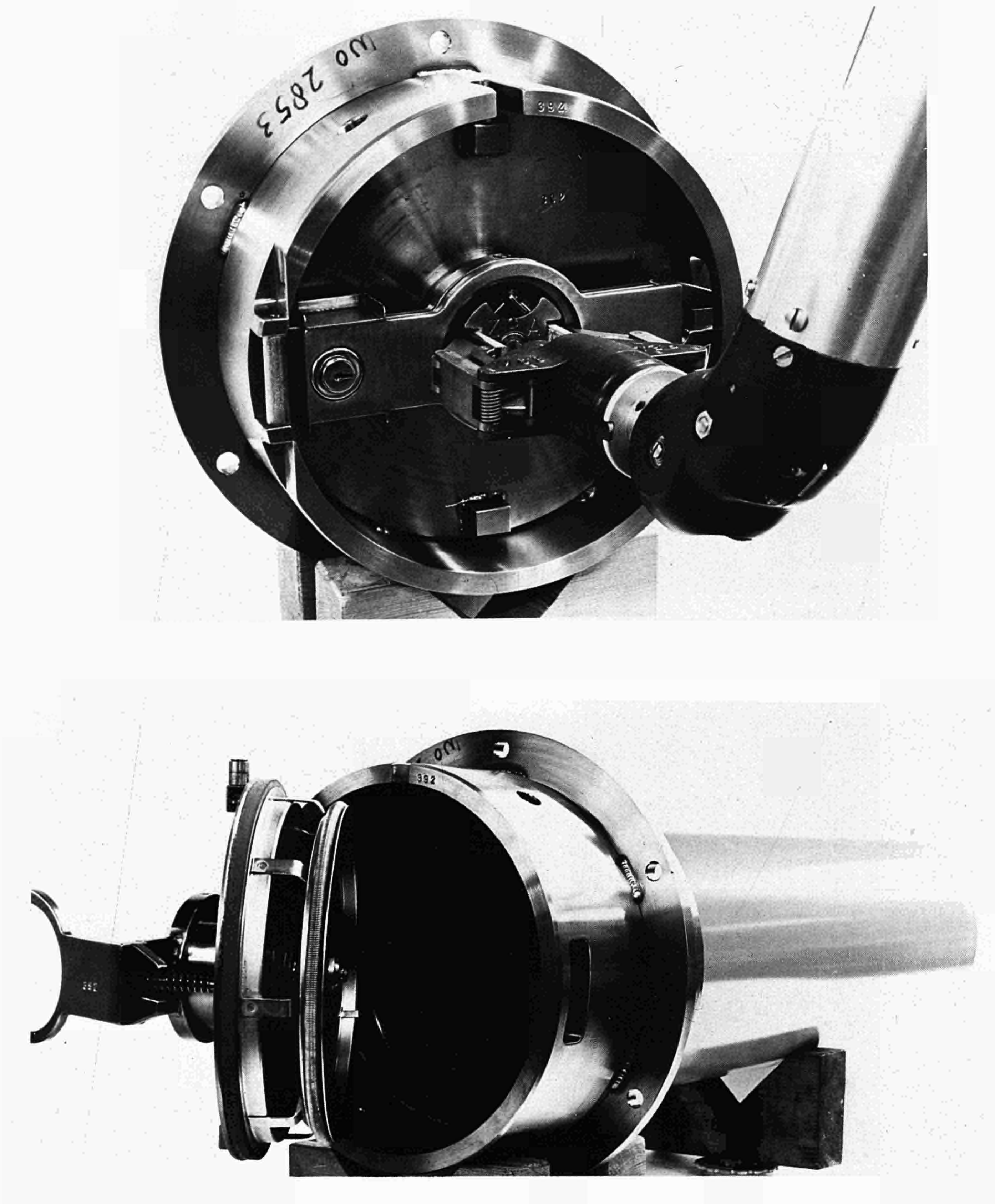


Fig. 74 HFR Dismantling Cell. Double-sealing transfer lock.

2.3.3.4 Neutron Radiography Installations (Proj. 340. ECN Project 8.293)

a) HFR pool camera

During the first half year of 1980, 116 neutron radiographs (of which 12 for ECN projects) have been made of a variety of objects, mostly irradiated fuel pins and materials samples with the associated irradiation capsules.
The new compact control panel (Fig. 77) for the in-pool camera which is mounted on the wall of the HFR dismantling cell is functioning quite satisfactorily.
The international comparison of the accuracy of various profile measuring apparatus has yielded results from the laboratories at Geesthacht, Mol, Harwell, Risø , Saclay and Ispra. Most of the reported measurements on the neutrogram (Fig. 75) of a special calibration pin (see HFR Programme Progress Report, July-December 1979) show good correspondence with actual dimensions. The results will be analized in a special report.
A special movable negative carrier has been designed and fabricated to facilitate the use of the profile projector for the measurement of large (~ 40 cm length) neutrograms.
First test neutron radiographs have been taken using thin cadmium filters. This technique enables the structure of fast reactor fuel to be visualised which is a total absorber of thermal neutrons.

b) Beam tube camera (ER 169)

During the January shutdown period the new "dry" neutron radiography installation has been installed in and around beam tube H8. Figures 78 through 82 give an impression of the new installation and the mounting activities.
During the first tests of the new apparatus it appeared that manual operation of the central rotating plug, which a.o. contains the film camera, is rather difficult and impractical. A special drive mechanism which will reduce the forces needed for operation of the rotating plug has been designed and is presently under construction.
As part of the characterization of the new installation neutron flux measurements at the position of the film plane have been carried out. As appears from Fig. 76 a relatively flat neutron flux distribution, with the average flux density lying slightly above 10^{11} m^{-2} s^{-1} has been achieved across the surface of the film.

Fig. 75 Reproduction of a part of the dummy calibration pin.

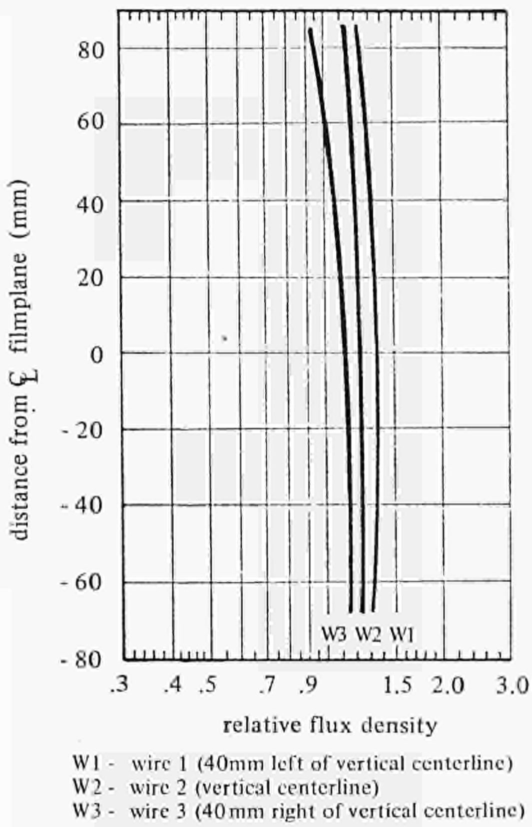


Fig. 76 Dry neutron radiography camera. Relative vertical flux density distributions obtained from measurements with gold wire pieces in front of beam tube HB-8.

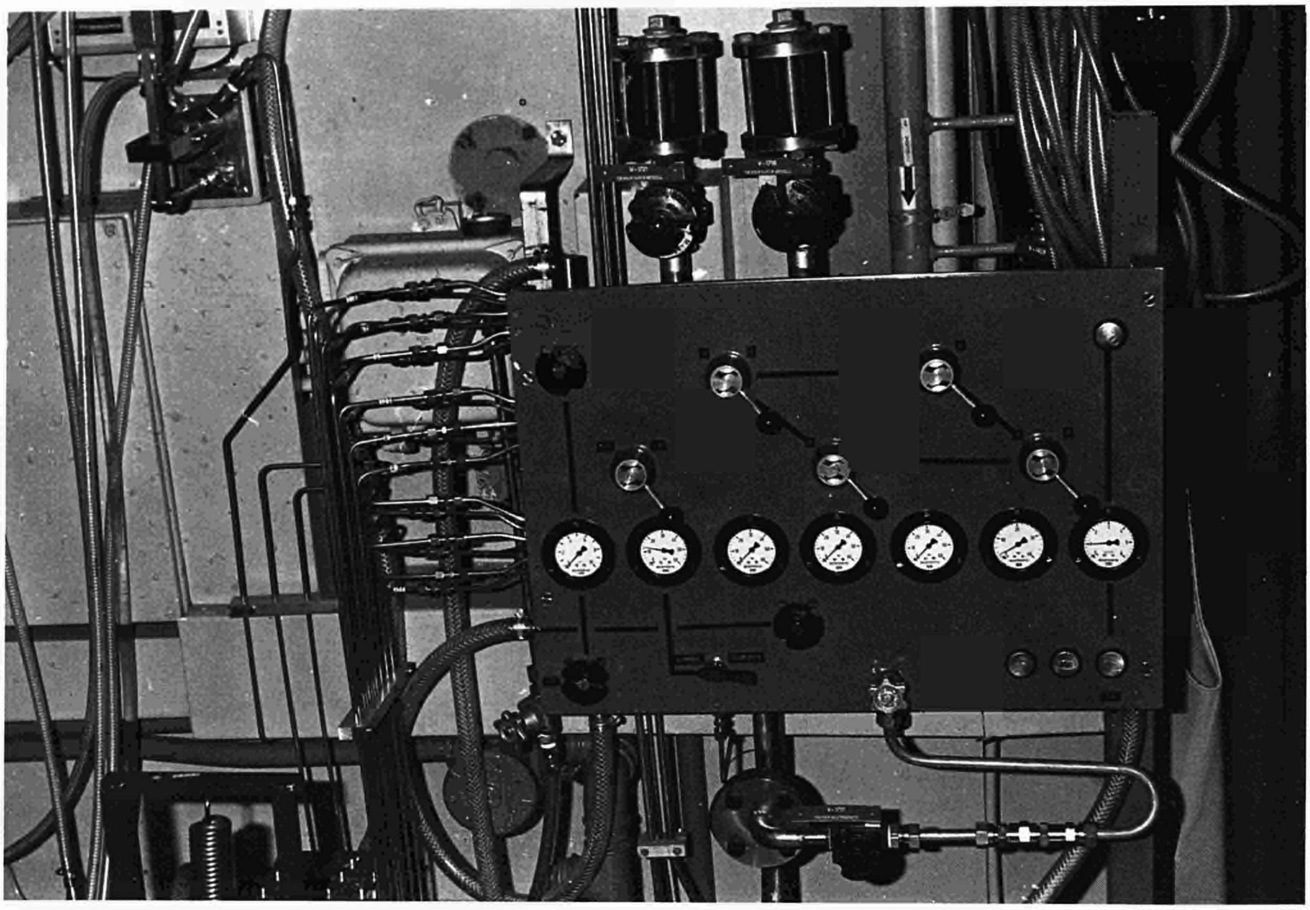


Fig. 77 The new control panel for the HFR pool camera placed against the DM cell wall.

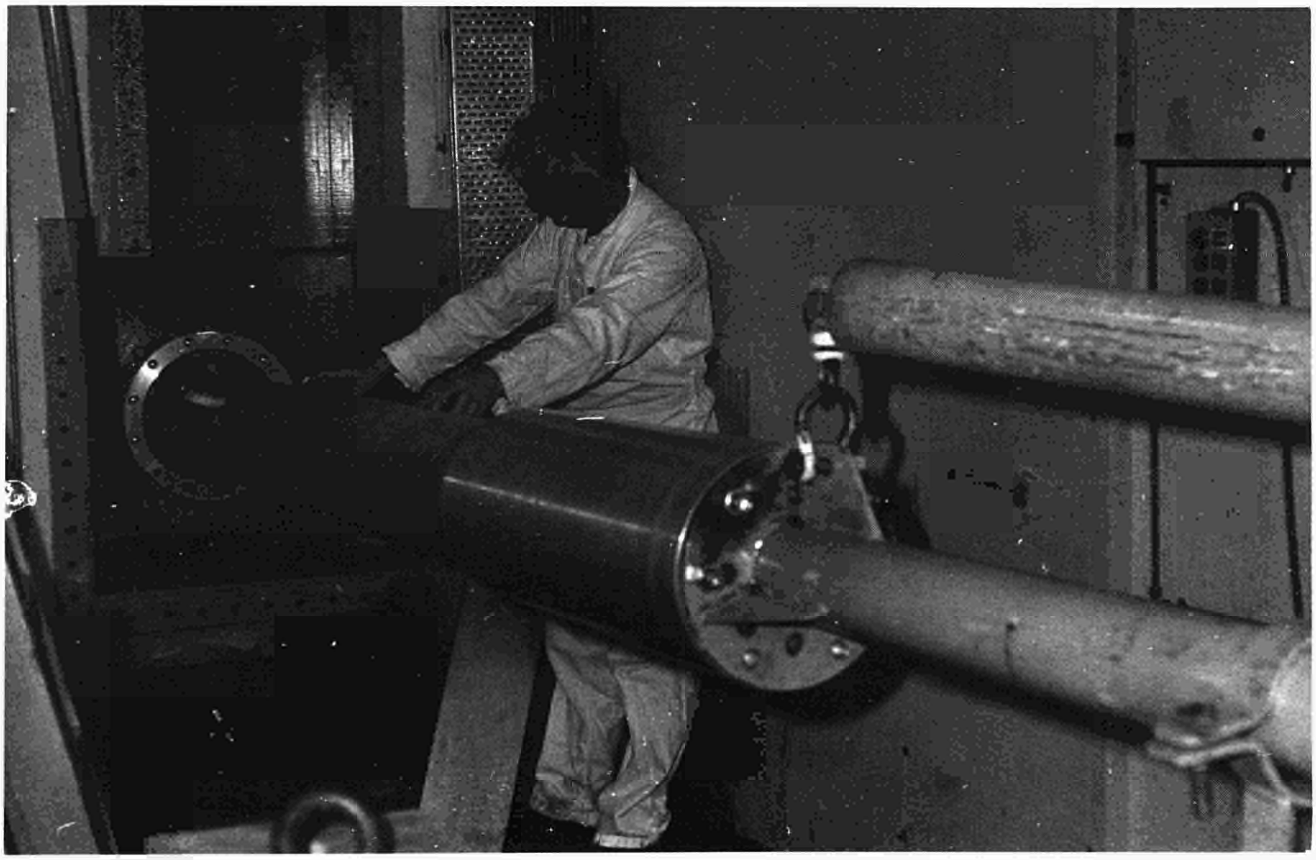


Fig. 78 Dry neutron radiography camera. Placing of the in-pile collimator in beam tube HB-8.

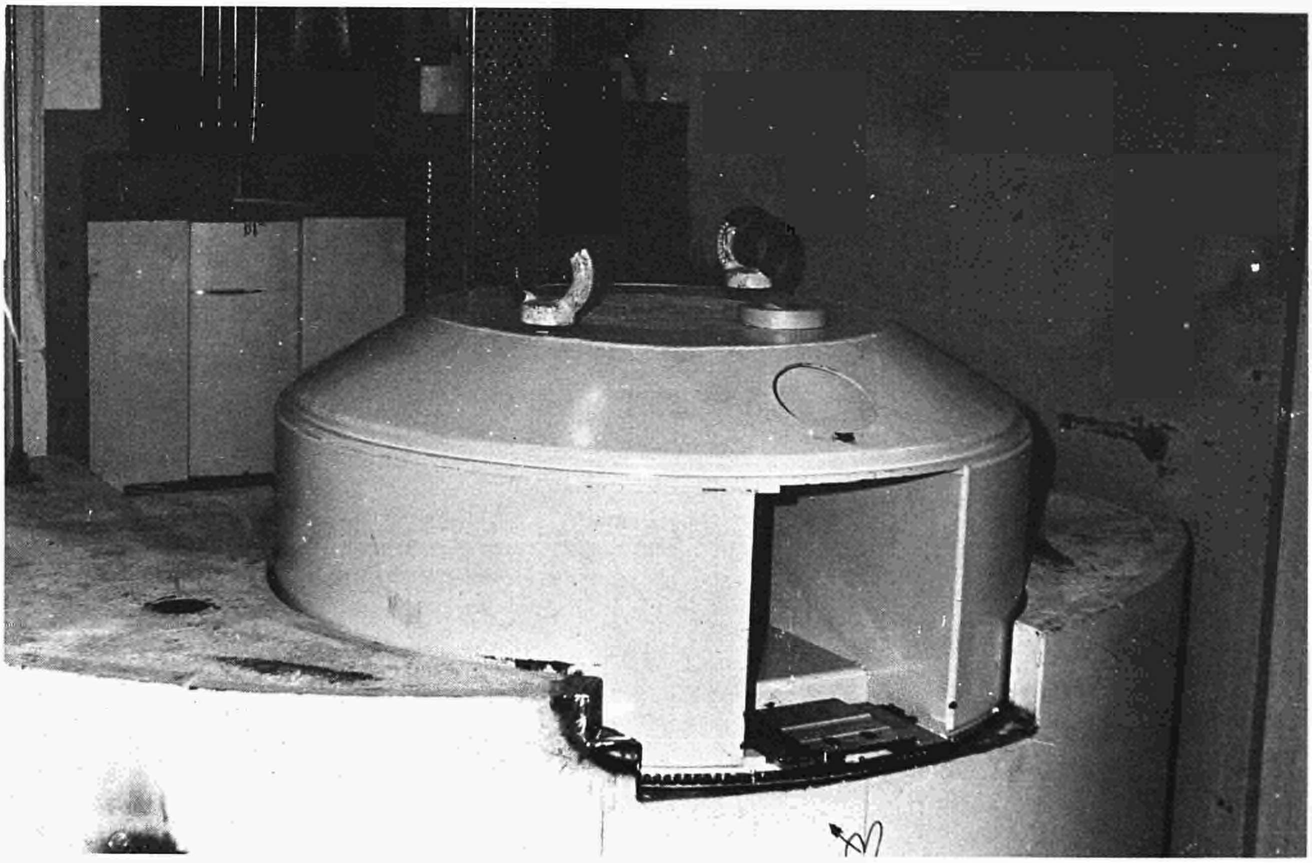


Fig. 79 Dry neutron radiography camera. Position of the central rotation table inside the biological shielding blocks.

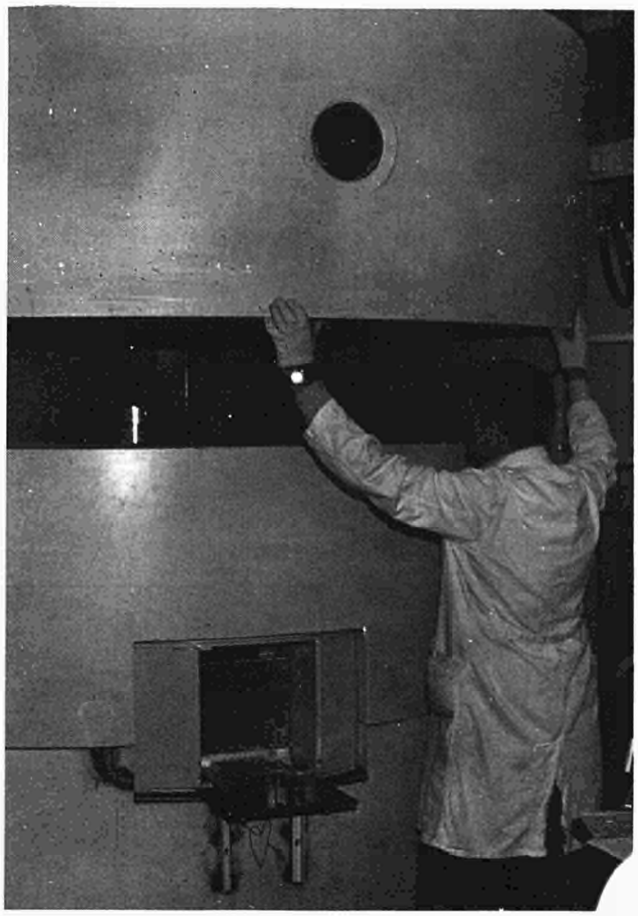


Fig. 80 Dry neutron radiography camera. Placing of the last biological shielding block .

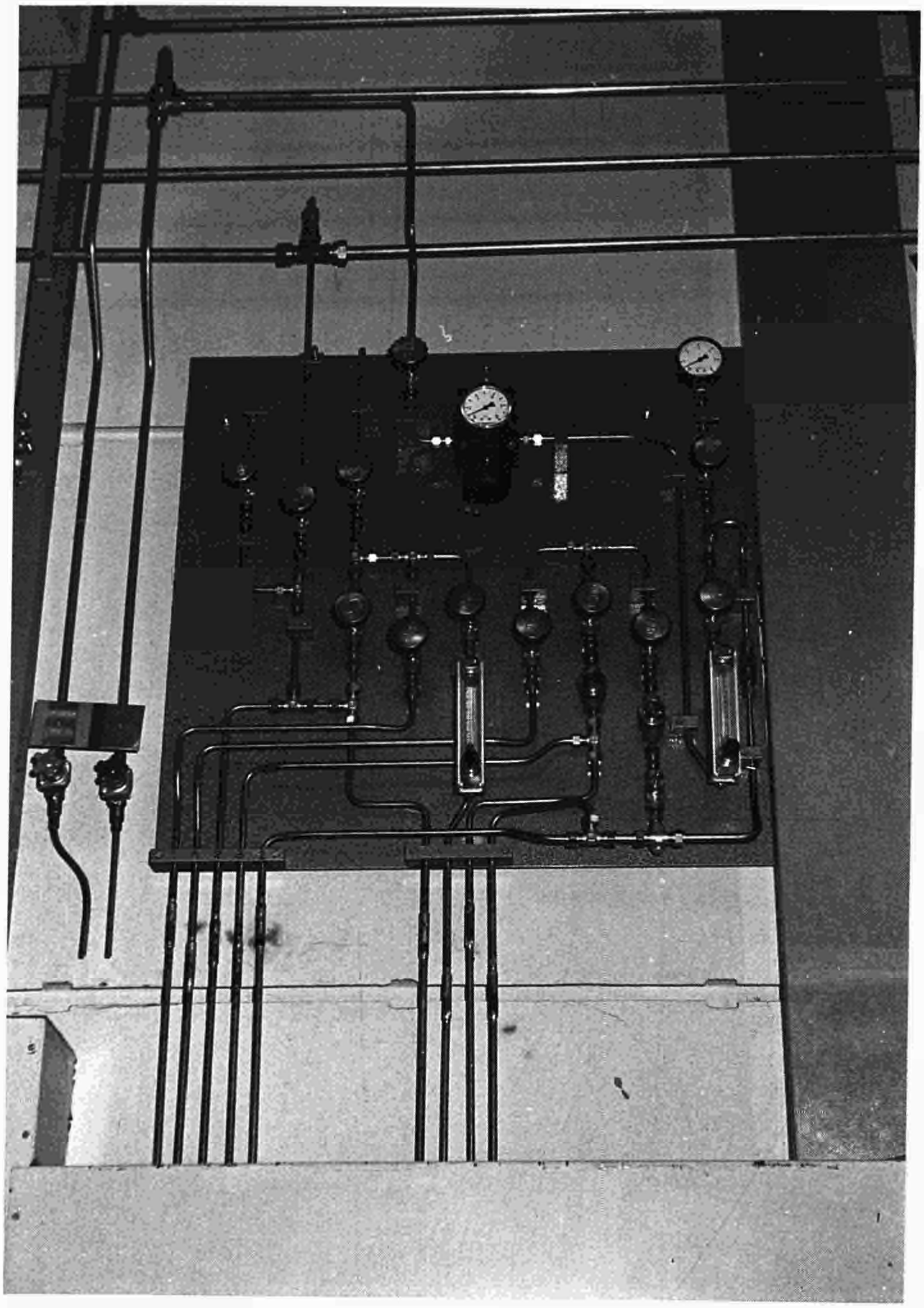


Fig. 81 Dry neutron radiography camera.
Service panel for cooling water and He-gas supply above the HB-8 facility.

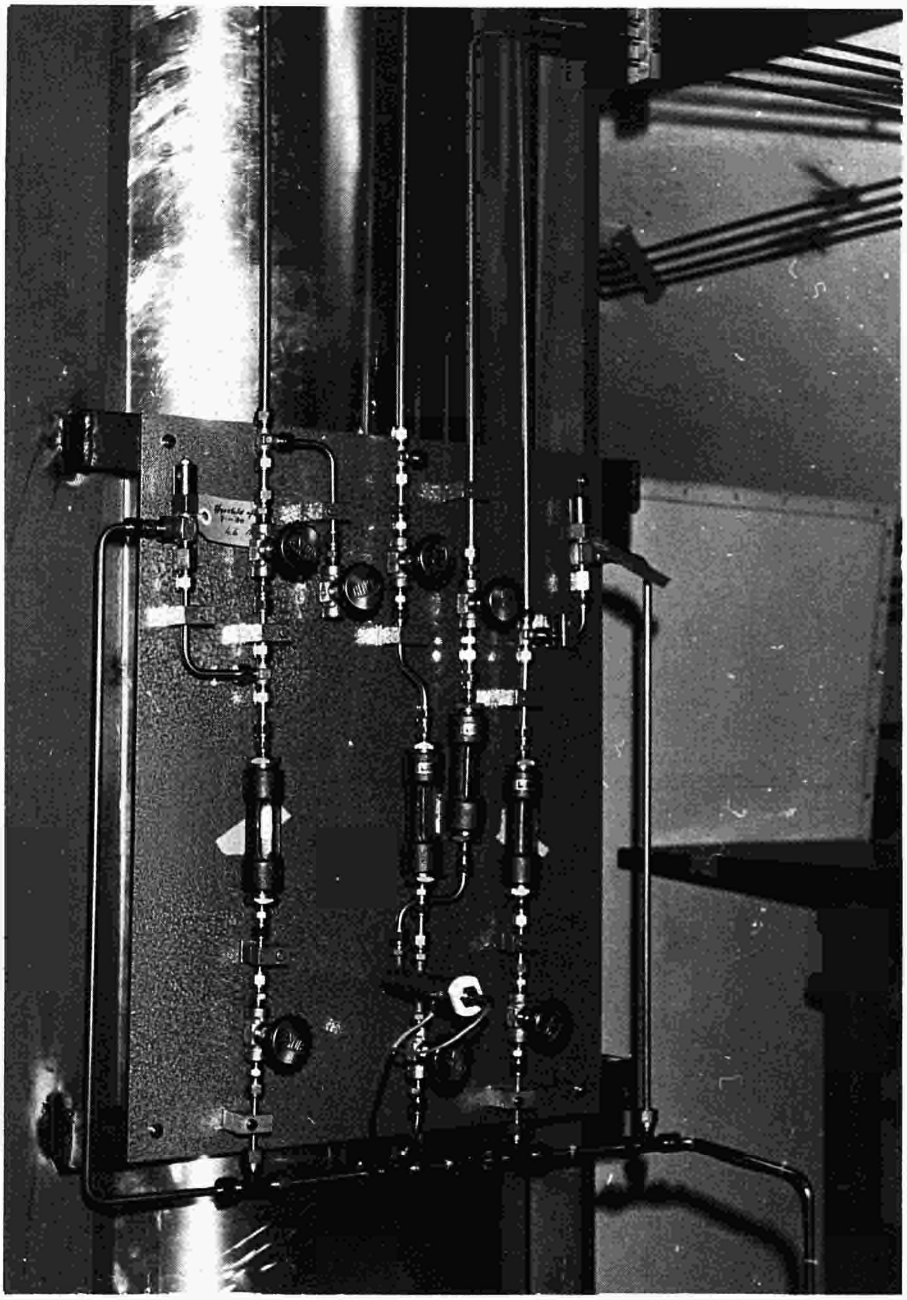


Fig. 82 Dry neutron radiography camera.
Control panel against the lead column in the cellar of the HFR.

2.3.3.5 Post-Irradiation Work (Proj. 350)

Within the frame of intermittent out-of-pile creep measurements of irradiated graphite samples a variety of dimensional measurements and/or reloadings of active samples into TRIO rigs have been carried out on the D156-04, -12, -20 and on the D85-36, -37 experiments (see paragr. 2.2.3.1 of this report).

A new shielded container which can accommodate and transport 3 reloaded sample holders is being constructed and will be operational in the second half of 1980.
This container enables the radiation-protected and contamination-controlled charging of a TRIO 129 rig with active sample holders.

2.3.3.6 Remote Encapsulation System EUROS (Proj. 360)

The EUROS project will provide the means of remote encapsulation of irradiated LMFBR fuel pins into specially prepared double wall capsules loaded into the core or chimney and Pool Side Facility of the HFR.

During the reporting period, all the major components of the in-cell equipment were ready for testing. Welding head and in-cell furnace have been fixed on their supports, and mounted on the trolley. First cold tests outside of the cell have been started. A first series of tests would be carried out, with dummies to control the welding and the heating apparatus. The following tests are the filling method of Na (NaK) and closing of the filling tubes.

The connections on the hot cell door, and the transformations of the EUROS cell door, have been finished. The revised time table shows that the first cold tests inside of the cell would take place at the end of 1980.

2.3.3.7 Standard Irradiation Devices. Assembly Laboratories (Proj. 400)

a) In-tank reloadable standard capsules

In anticipation of the normal requirements for standard irradiation devices (TRIO's and REFA's), 10 manufacturing organizations have been invited to tender for about 50 assemblies for the next five years.

A new standard irradiation device, TRIO 131, is being introduced (Fig. 83). The existing TRIO 129 has an internal diameter of 29^{H8} mm, and the new TRIO 131 has an internal diameter of $31{,}50^{H8}$ mm (see paragr. 2.2.3.2 of this report).

A further development concerns the introduction of a TRIO head which incorporates three separate displacement devices, so that each of the installed sample carriers can be moved independently. The first two of these rig heads will be manufactured this year (Fig. 84).

b) New PSF devices

Transient condition experiments in the HFR pool used to suffer from
- mechanical and nuclear interferences between neighbouring PSF positions,
- elastic deformations in the trolley drive mechanisms causing jerky movements of the irradiation capsules.

A series of measures have been initiated to improve the quality of transient experimentation, viz.:
- installation of a new PSF table support (see HFR Programme Progress Report July-December 1979), and several geometrical adjustments of the new table with respect to the reactor vessel,
- installation of separation plates defining experimental compartments (Fig. 87),
- additional rollers on the lower face of PSF capsule supports for smooth guiding inside the separation plates (Fig. 86),
- development of three new drive mechanisms (trolleys), construction and out-pile testing of one of them, using a so-called "push-pull" transmission (Fig. 85).
  This type will be used for the first time during the second half of 1980.

For PSF positions 9 and 10 which are regularly occupied by the front end of the pool neutron radiography camera, special capsule supports have been developed and manufactured (Fig. 88).

c) Assembly laboratories

24 specimen carriers; 7 complete irradiation devices, and 5 PSF supports have been assembled during the reporting period.

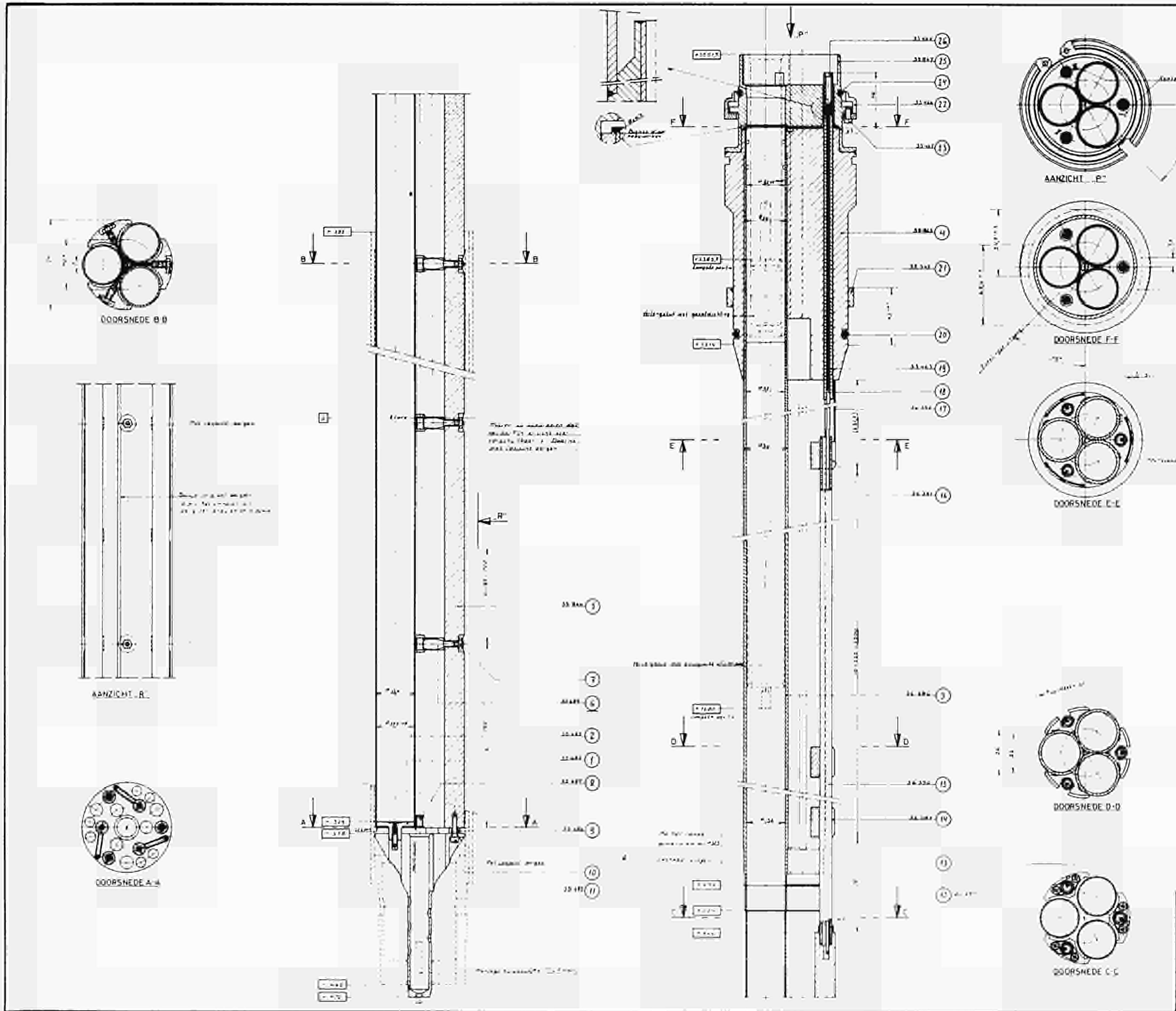


Fig. 83

TRIO facility with increased useful volume.
General assembly.

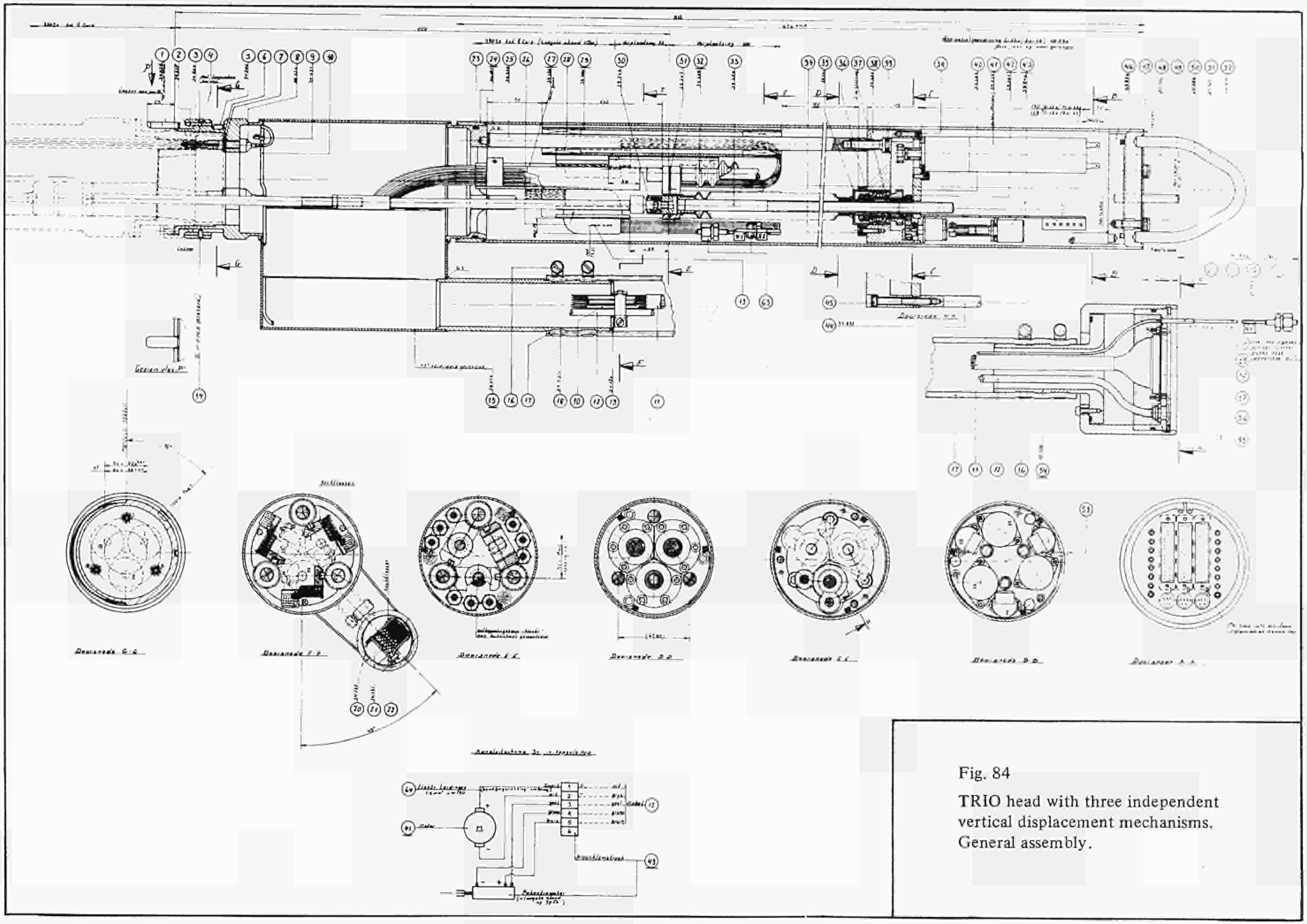


Fig. 84

TRIO head with three independent vertical displacement mechanisms. General assembly.

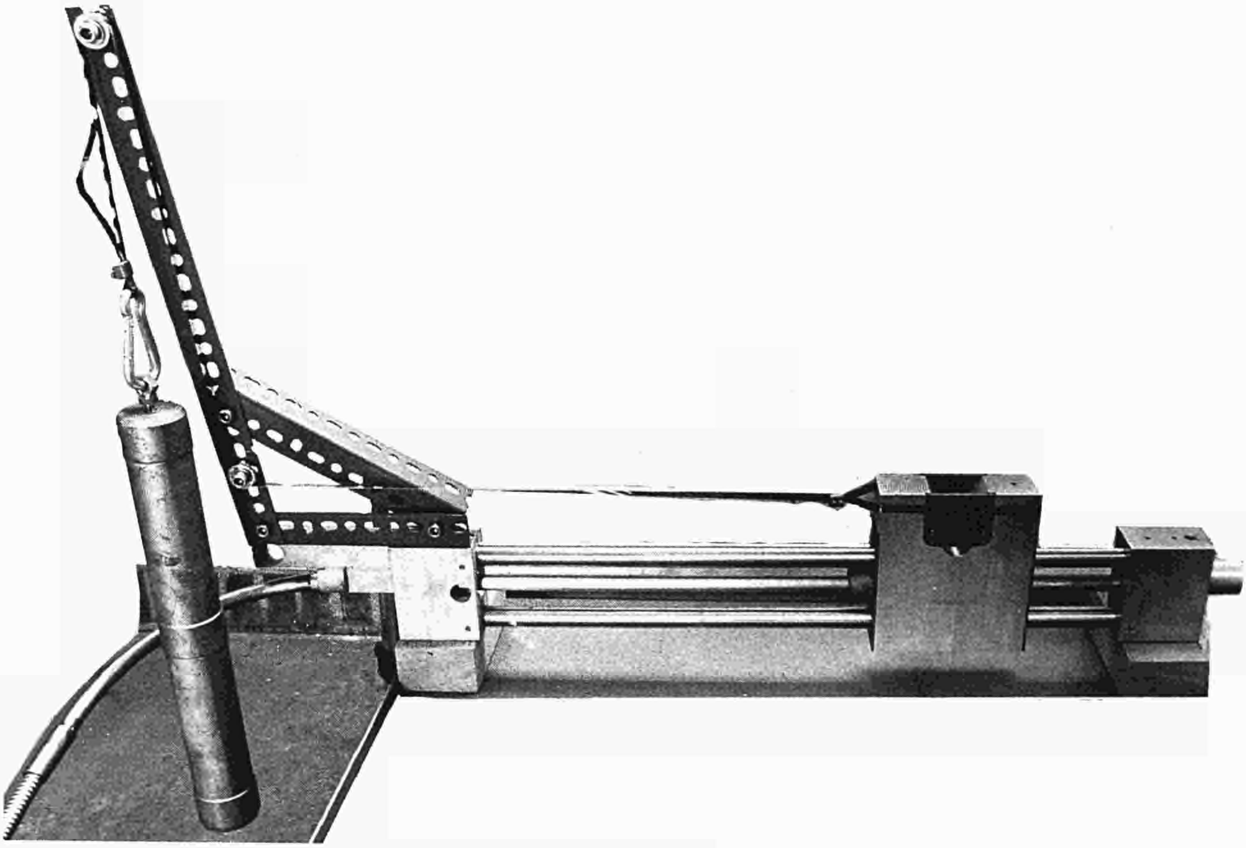


Fig. 85 PSF "push-pull" trolley testing using a dead weight load.

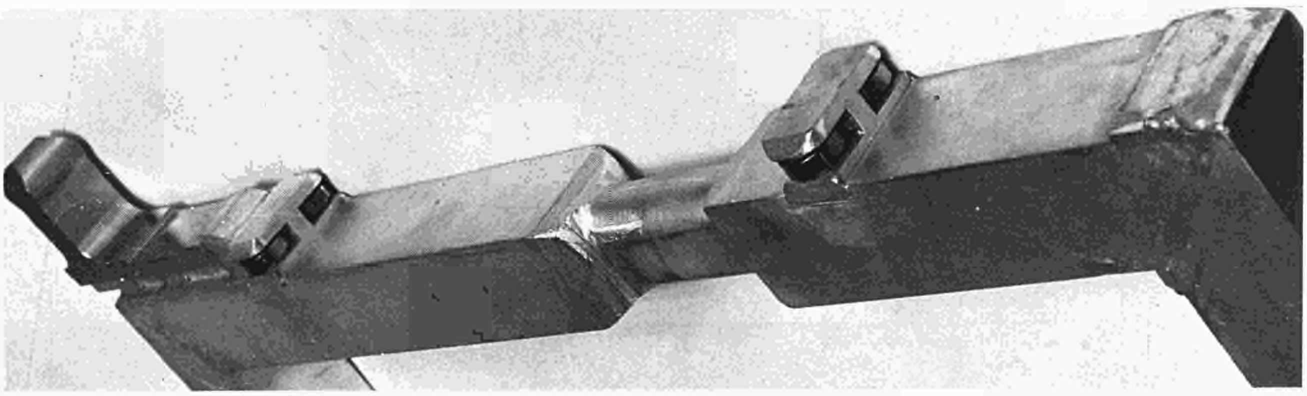


Fig. 86 PSF support for operation with the new separation plates. View from underneath.

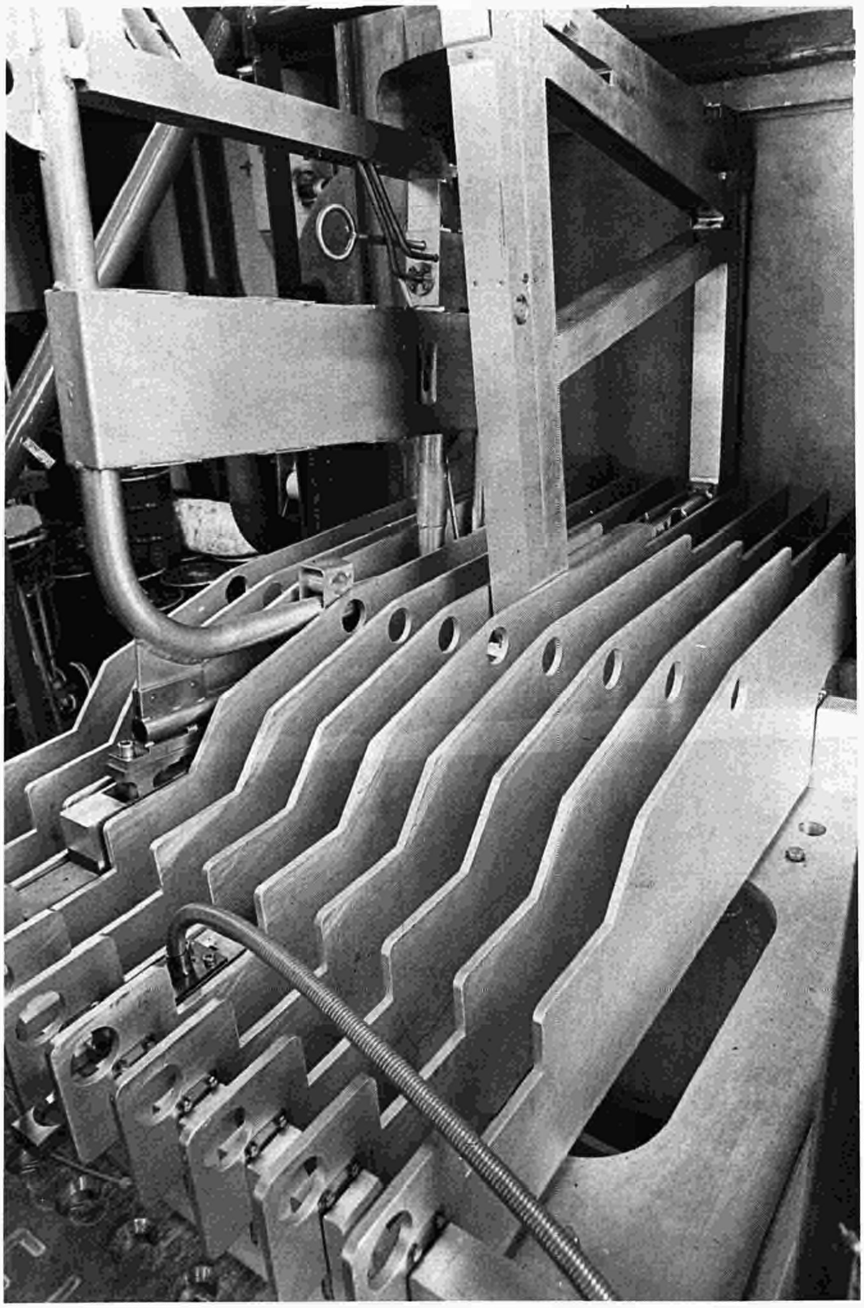


Fig. 87 PSF support table with installed separation plates.
"Push-pull" cable in the foreground.

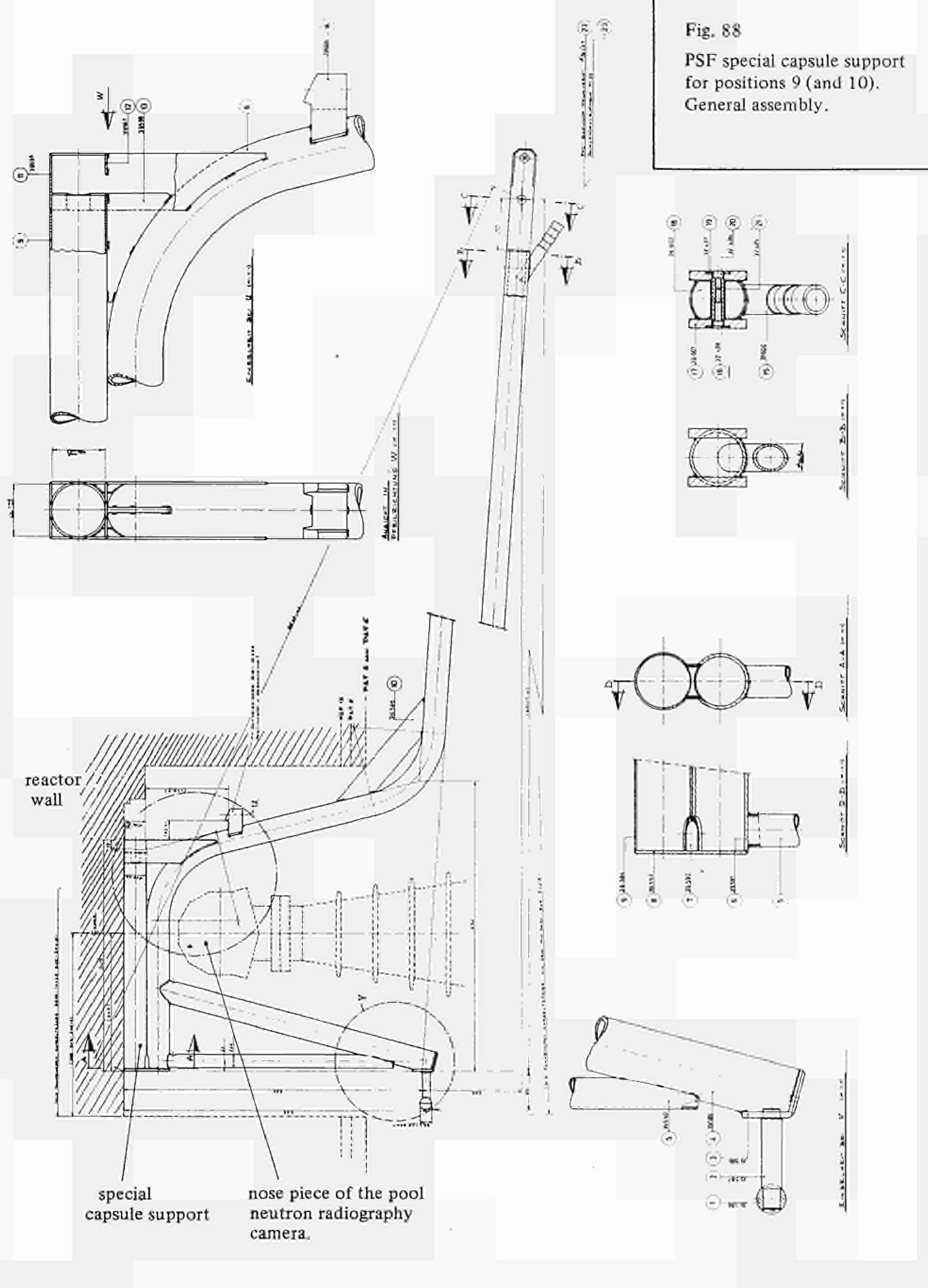


Fig. 88

PSF special capsule support for positions 9 (and 10). General assembly.

2.3.3.8 Development of LWR Fuel Pin Testing Facilities (Proj. 405)

Two capsule carriers with gamma thermometer and self-powered neutron detectors have been installed in HFR and tested successfully.
For digital data recording and evaluation of eddy current and diameter measurement on irradiated fuel pins in the HFR pool a "MINC-11" system has been installed (Fig. 90).
Evaluation of the BWFC-RA capsule prototype testing with a fuel pin length monitoring system was started.

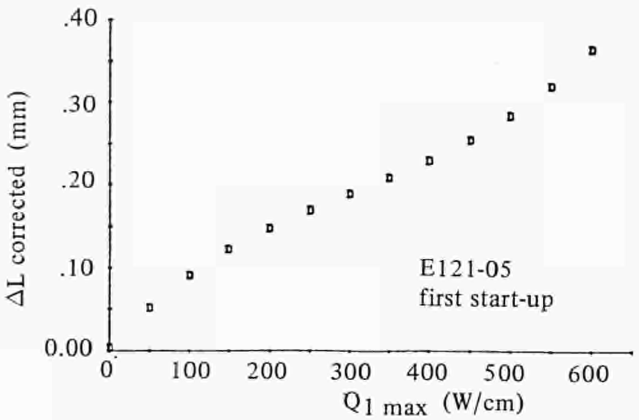


Fig. 89 Prototype testing of the BWFC-RA capsule. Fuel rod length vs. $Q_{1\ max}$.

Fig. 89 gives an impression of fuel pin length change versus fuel pin power.
For developments and improvements of the PSF and its drive systems, see paragr. 2.3.3.7 above.

2.3.3.9 Development of a Control System for Swept HTR Fuel Experiments (Proj. 406)

By January 1980, all sweep loop components had been transported to the HFR. Cabling of instrumentation cabinets and tubing of all gas panels have been terminated. Figs. 91 and 92 give a general impression of the command panels and the accompanying cables and tubing on site. Commissioning of the installation has been started and will be terminated in August [1].
A five weeks training course for experiment operators is planned for September, 1980.

The gamma spectrometry equipment, a multi-channel analyser with a 4 k analog-to-digital converter (ADC) and a PDP 11/23 host computer with dual floppy disk and hard disk mass storage possibilities, have been delivered. This equipment is currently being commissioned.

The design of the extended sampling station has been finished and fabrication will start in July 1980. The design and safety report has been compiled. A detailed manual will be distributed in August [2].

Fig. 90 LWR fuel pin eddy current and diameter measurements. New data acquisition system.

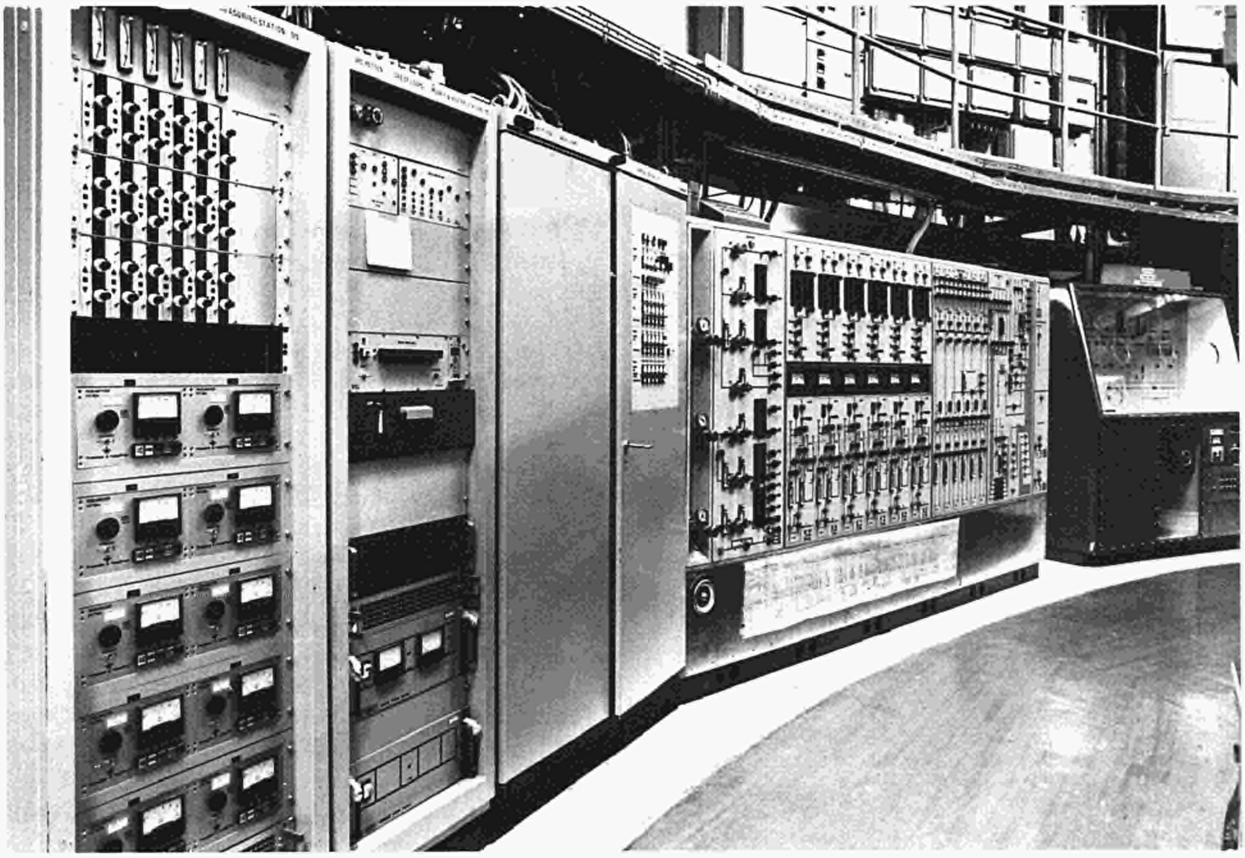


Fig. 91 HTR sweep loops. Instrumentation and command panel installed in the reactor building.

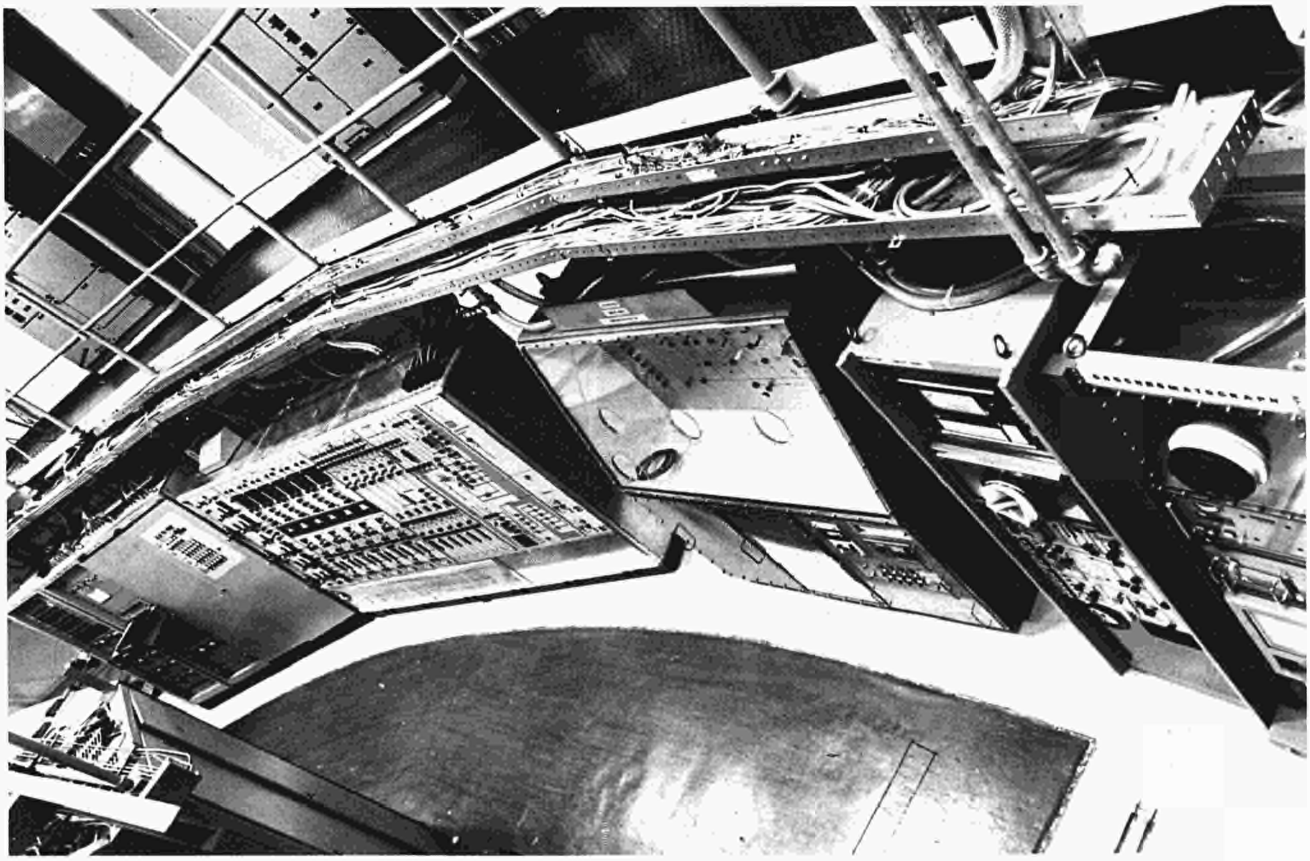


Fig. 92 HTR sweep loops. Instrumentation and command panel installed in the reactor building.

Hot operation with experiment E138-2 will start in November, 1980. The overall time needed for this project has then been seven years, marked by the following milestones:

- start of the project	end 1973
- design phase:	
mechanical	1974 - 75
electrical, electronics	1976 - 77
- manufacture, assembly, component testing:	
mechanical	1976 - 77
electrical, electronics	1977 - 78
- cold test of the complete installation:	
1st part (Technology hall)	1979
2nd part (HFR building)	Apr.-Oct.1980
- active operation	from Nov.,1980

References to paragraph 2.3.3.9

1. Abnahmetest der Sweep Loops.
R. Conrad, G. Robert,
Technical Memorandum, IT/80/1678, March 1980

2. Manual for Sweep Loops.
R. Conrad, G. Robert,
Technical Memorandum, IT/80/1679, Aug. 1980

2.3.3.10 Computing Facilities

a) General

The ever increasing needs for design calculations and for data acquisition, processing, and retrieval have caused a considerable enlargement of facilities on one hand and the training of more and more staff for computer operation, on the other hand. During the first half of 1980, the hardware used within the HFR programme comprised:

- the JRC PDP11/15 installation with multi-user access, mainly working for BWFC data processing, special heat transfer problems and other tasks requiring specific software support,
- seven PDP11 mini computer-based data acquisition and on-line treatment systems inside the reactor building, to be enlarged to 11 by the end of the year,
- two HP 9845 office computer installations (see below),
- the CDC CYBER 175 batch terminal (see below),

b) HP 9845B desk top computers

One installation is used specifically for the BWFC Programme (see paragr. 2.2.3.4 and Fig. 52) for data processing, a new filing (data bank) system, and special LWR fuel pin transient behaviour code work.

The second installation has more general tasks. A number of own programs have been developed:

- Program "ADECAY" calculates activation of an isotope during irradiation or activity of a radio-isotope during a cooling time period. Plots the results on logarithmic grid.
- Program "WUNDE 1" calculates radial temperature distribution in a cylindrical geometry receiving fission heating and/or nuclear heating.
  Plots temperature profiles as well as temperature vs. linear power (on various radii or varying the gas mixture) and temperature vs. radius limiting the gas gap.
- Program "BERNAT" calculates burn-out conditions in a water-cooled capsule using the Bernath's formula. Plots permissible heat flux density and safety factor vs. flow.
- Program "HFR-OC" plots the percentages of the HFR reactor occupation vs. cycle (maximum 11 cycles) in the form of a line chart.
- Program "TEXPLO" plots a text or block (for example an organigram) on the external plotter. A preliminary plot appears on the video of the computer, enabling corrections or modifications to be made before the external plotting.

During a training course held in Petten during two weeks in January/February 1980, 10 members of the JRC staff have been introduced into programming and operation of these machines.

c) CYBER batch terminal

The provided facilities include (Fig. 93):

- line printer — 300 lines / minute
- card reader — 300 cards / minute
- paper tape reader — 500 characters / second
- flexible disc drive — 253 k bytes
- control V.D.U, which selects the input and output devices required and can also be used interactively with the main frame.
- four satellite V.D.U.'s, which are connected to the batch terminal on four separate circuits through a multi-plexer.

A total of 33 V.D.U. connections are provided for HFR Division personnel in their offices, so that the provided services can be used remotely.

Communication with the main frame is at 9600 baud, and response time rarely exceeds a few seconds even during a period of high main frame usage.

The main utilization, on a self-service basis, is for data processing of steady-state HFR irradiation experiments (200 to 400 signal channels) and for thermal behaviour design work using a two dimensional, transient code.

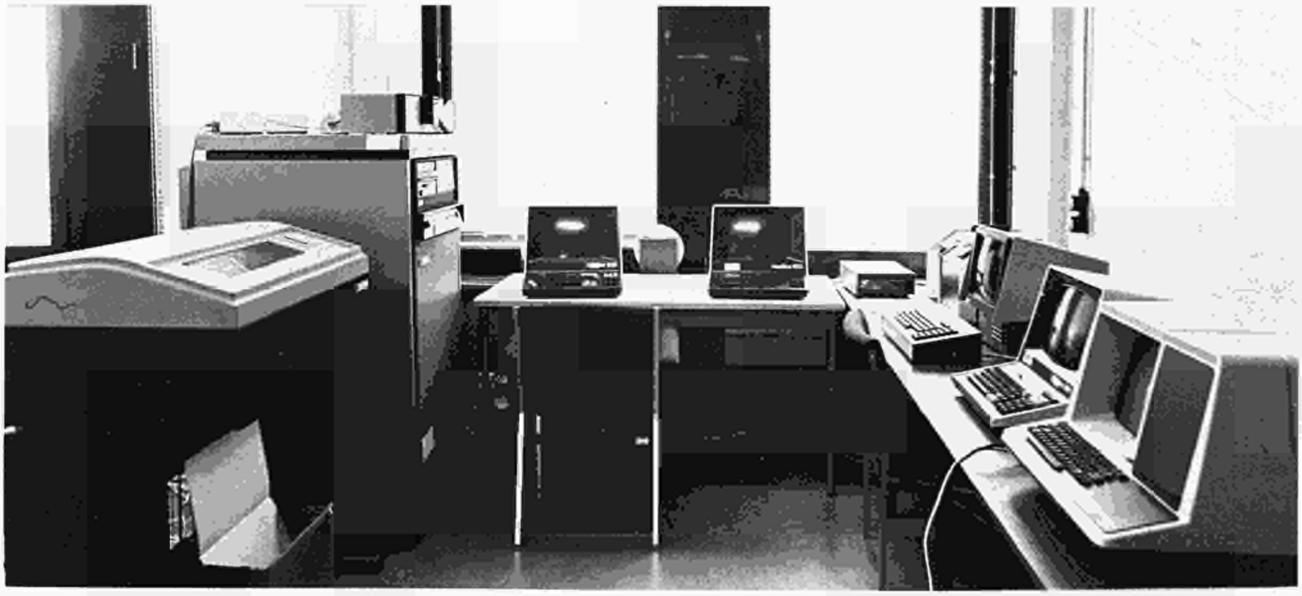


Fig. 93 CYBER batch terminal.

2.3.3.11 Project Management

- Planning

The HFR planning meeting was held three times and three editions of the loading chart (nos. 39 through 41) were issued.
The working group of timing schedule and practical aspects for the post-irradiation phase of experiments met three times.
First test runs with a new irradiation project planning code "PLANIT" have been carried out.

- Documentation

The 13th Newsletter of the "Euratom Working Group on Reactor Dosimetry" was assembled and issued.
The Proceedings of the "Third ASTM-Euratom Symposium on Reactor Dosimetry", Ispra, 1 - 5 Oct. , 1979 (EUR 6813 en/FS), have been assembled.

- EWGRD

The 44th meeting of the "Euratom Working Group on Reactor Dosimetry" was held on April 29th, 1980, at the Petten Forum.
Its subgroups "Irradiation damage" and "Nuclear data" met on April 28th, 1980, also in Petten.

- IDWG

A meeting of the Select Committee of the "Irradiation Devices Working Group" was held on May 22nd, 1980, in Petten. The 26th plenary meeting of the Group, now scheduled for 8/10 October 1980 at the GKSS Research Centre Geesthacht (Federal Republic of Germany), is under preparation.
In the general frame of the IDWG, a second experts' meeting on "Neutron Radiography" was held in Petten on 27/28 May, 1980.

- ACPM

The Advisory Committee on Programme Management met in Petten on June 13, 1980. It reviewed status and progress of the HFR Programme on the basis of 17 documents prepared by JRC Petten.

2.3.3.12 Irradiation Technology (ECN)

- In-core instrumentation testing

Most of the work during the reporting period was associated with the irradiation and evaluation of the INTEC experiment. The results obtained sofar have been summarized under exp. R163 (see paragr. 2.2.3.5 of this report).

- Thermocouples

During the reporting period some 500 mineral-insulated metal sheathed thermocouples have been welded and tested for a variety of in-core and other projects.

- Heater elements

In view of the unreliable behaviour of the heater elements in the INTEC experiment the quality aspects of in-core heaters have been reassessed and discussed with manufacturers. New specifications for future heater deliveries have been drawn up.

- Heat pipe

A literature search has been carried out in order to investigate the applicability of heat pipes in irradiation devices. A project plan specifying the development activities which are required to test and qualify a heat pipe for in-core application is being drafted.

Conclusions

3. CONCLUSIONS

3.1 HFR Operation and Maintenance

The operation of HFR Petten has been pursued during the first half year of 1980, according to the 1980/83 JRC Programme decisions.
The reactor operated within a few percent of the cycle calendar established by the end of 1979. Perturbances resulted from a large number of scheduled (10) and unscheduled (25) power reductions and scrams.
The overall plant availability was reduced by only about one percent as a consequence of these perturbations. However, the effect on irradiation tests was undesirable.

During the winter maintenance period, a certain number of components have been inspected, serviced and replaced. The good state of health of the plant has been confirmed.

Development work continued in support of the design of new irradiation facilities and of refined data computations. A new PSF table has been installed for improved transient irradiation experiments.

3.2 Reactor Utilization

Hampered by a lack of staff, the average occupation could not reach the planned level and remained below the average of the past three years :

Reporting period	Planned utilization (in °/o of the theoretical full capacity)	Achieved utilization
Jan./June '77	74	66
July/Dec. '77	70	56
Jan./June '78	76	64
July/Dec. '78	70	67
Jan./June '79	84	80
July/Dec. '79	77	74
Jan./June '80	85	64

The most significant achievements were:
- the successfull performance of nine HTR graphite irradiation experiments
- the irradiation of a large number of tensile and fracture mechanics steel specimens within narrow temperature limits.
- the successful continuation of a new series of fast breeder reactor fuel transient irradiation tests,
- 12 ramp tests on pre-irradiated LWR fuel pins,
- permanent utilization of nine horizontal beam tubes.
- intensified occupation of radioisotope production facilities, particularly for activation analysis.

3.3 General Activities

A reduced number of projects could be pursued besides the current maintenance of equipment and laboratories, i.e.
- detail design and specifications for a future reactor tank,
- manufacture of several new standard irradiation devices,
- work on new LWR fuel irradiation devices,
- design and development of new PSF 'trolleys' and improvements of the new PSF table,
- pre-start-up testing of a large out-of-pile installation.

Generally spoken the development activities suffered from the work overload in the irradiation project sector and could not be pursued with the desirable speed.

4. JRC PUBLICATIONS
HFR Programme, January - June 1980

- Programme Progress Report HFR, July - December 1979.
  P. von der Hardt
  COM 3731

- Proceedings of the Third ASTM-Euratom Symposium on Reactor Dosimetry.
  H. Röttger (Ed.)
  EUR 6813 en FS

- Newsletter on Reactor Radiation Metrology, Nr. 13, February 1980.
  H. Röttger (Ed.)

- High Fluence Creep Behaviour of near Isotropic Pitch Coke Graphite.
  M.R. Cundy (JRC Petten), M.F. O'Connor, G. Kleist (KFA Jülich)
  Paper to the Carbon '80 Conference, Baden-Baden, June 1980.

- FRG/US-Zusammenarbeit bei der Charakterisierung von Reaktorgraphiten.
  M.F. O'Connor (KFA Jülich), G.B. Engle, L.A. Beavan, R.D. Burnette (General Atomic Company, San Diego), W.P. Eatherly (Oak Ridge National Lab.), M.R. Cundy (JRC Petten).
  Paper to the Carbon '80 Conference, Baden-Baden, June 1980.

GLOSSARY

ACPM	Advisory Committee on Programme Management
ARTEMIS	Amoeba Rig Test Experiment on kernel Migration, In-Pile Simulation
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AUSTIN	Austenitic Steel Irradiation
BEST	Brennelementsegment
BISO, TRISO	Coated HTR fuel particle types
BONI	Borosilicate glass samples
BWFC	Boiling Water Fuel-element Capsule
CATRI	Carbide fuel TRIO irradiation
CCTV	Closed circuit television
CII	Central Information Index
CL	Confidence Limit
CORROX	Corrosion experiment on Oxide fuels
COSAC	Computerized On-line Supervision of the data Acquisition
CRIMP	Graphite In-pile Creep Machine
DACOS	Data acquisition and control system
DAR	Damage to Activation Ratio
DIN	Deutsche Industrie Norm
DISCREET	Discontinuous in-pile graphite Creep Testing
DUELL, KAKADU	Twin capsules for fuel pin irradiation
ECN	Energieonderzoek Centrum Nederland
EDN	Equivalent DIDO Nickel fast neutron fluence
EEC	European Economic Community
EN	European Norm
ENS	European Nuclear Society
EUROS	European Remote encapsulation Operating System
EWGRD	Euratom Working Group on Reactor Dosimetry
FIT	Fissile Isotope Target
FOM	Institute for Fundamental Materials Research, Jutphaas, the Netherlands
GIF	Gamma Irradiation Facility
HB	Horizontal Beamhole
HEU	High-enriched Uranium
HFPIF	High Flux Pool Side Isotope Facility
HFR	High Flux Reactor
HIFI	High Flux Facility for Isotopes
HP	Trademark for Hewlett-Packard computers
HR	Hydraulic Rabbit facility
HTR	High Temperature Reactor
IAEA	International Atomic Energy Agency
IDEAS	Irradiation damage evaluation of austenitic steel
INTEC	In-pile instrumentation test capsule
IRI	Interuniversitair Reactor Instituut (Delft)
JRC	Joint Research Centre
KFA	Kernforschungsanlage Jülich
KfK	Kernforschungszentrum Karlsruhe
KTG	Kerntechnische Gesellschaft
LEU	Low-enriched Uranium
LOC	Loss-of-Cooling capsule
LVDT	Linear Variable Displacement Transducer

LWR	Light Water Reactor
MONA	Modular NAST
MTR	Materials Testing Reactor
MUCE	Millions of Units of Account
NAST	Na-steel irradiation
NIRVANA	Niobium and Vanadium samples irradiation in Na (sodium)
ORR	Oak Ridge Research Reactor
PDP	Trademark for 'Digital Equipment Corporation' computers
PIE	Post-Irradiation Examens
PIF	Pool side Isotope Facility
POCY	POwer CYcling experiment
PPR	Programme Progress Report
PROF	Poolside Rotating Facility
PRS	Pneumatic Rabbit System
PSF	Pool Side Facility
R & D	Research and Development
REFA	Reloadable Facility
RIF	Reloadable Isotope Facility
SHOT	Stationary High Overtemperature experiment
SINAS	Simplified NAS(T)
SNR	Schneller Natriumgekühlter Reaktor (Kalkar)
S.S.	Stainless Steel
TEDDI	computer programme to evaluate reactor neutron spectrum
TOP	Transient Overpower experiment
TRAMP(STICK)	gamma calorimeter
TRESON	Mesure de Transport d'Energie en pile par méthodes Soniques
TRIO	Irradiation device with three thimbles
USNRC	Nuclear Regulatory Commission (USA)
VDU	Vertical Displacement Unit (for irradiation capsules) *or* Visual Display Unit (for computer operation)
WOL	steel specimen code

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