PROGRAMME PROGRESS REPORT

July - December 1980

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

COMMUNICATION

Operation of the High Flux Reactor

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COMMISSION OF THE EUROPEAN COMMUNITIES
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 second half year 1980, it reached 67 °/o availability at 45 MW. The occupation by irradiation experiments and radioisotope production devices varied between 73 and 83 °/o.

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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.
A 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, utilisation, and management of HFR Petten.

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

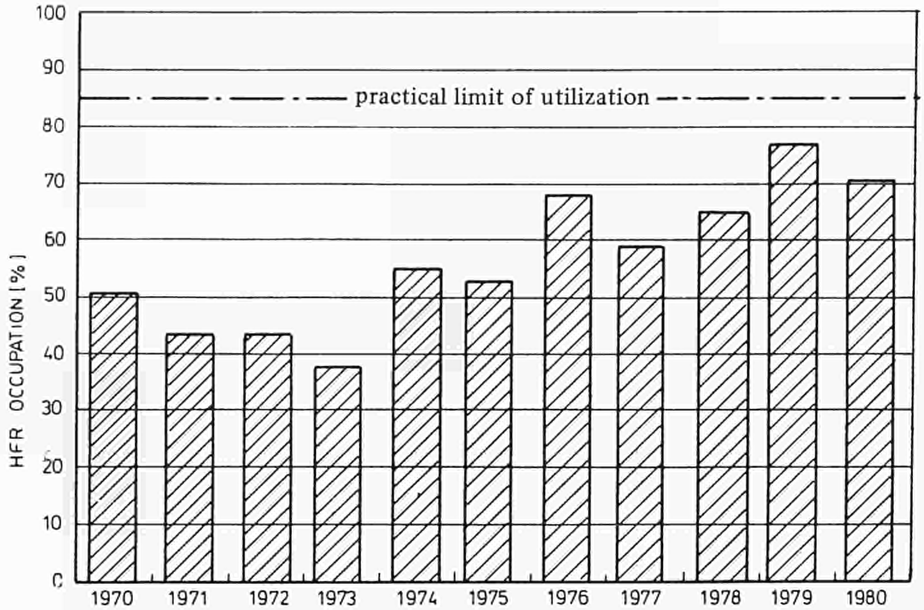
UTILIZATION

WHY ARE THESE EXPERIMENTS CARRIED OUT IN THE HFR?

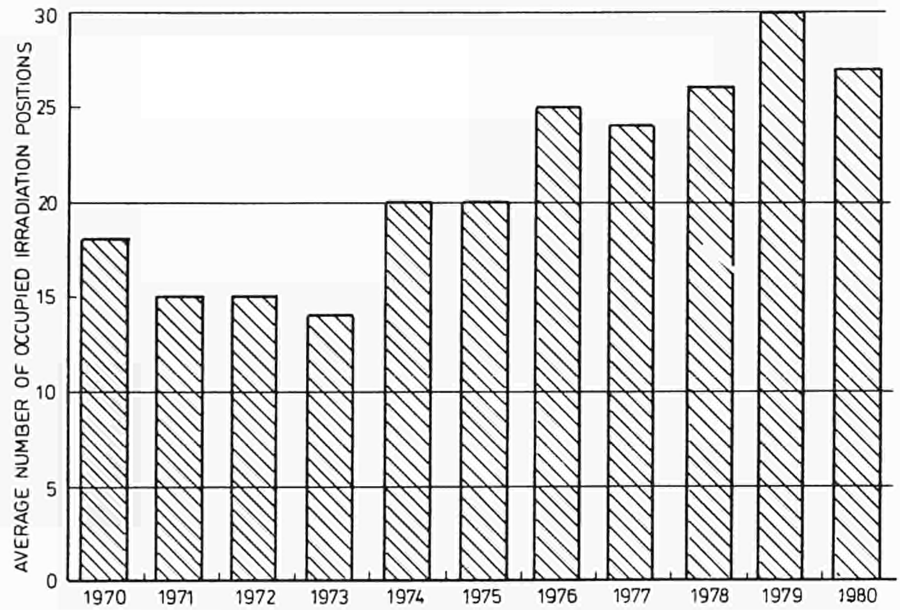
- FAST BREEDER REACTOR STRUCTURAL MATERIALS IRRADIATIONS: SEVERAL HIGHLY RELIABLE IRRADIATION CAPSULES AND SPECIAL HOT CELL EQUIPMENT AVAILABLE
- FAST BREEDER REACTOR FUEL PIN TESTING UNDER ABNORMAL CONDITIONS AND UNDER OPERATIONAL TRANSIENTS: LONG-STANDING EXPERTISE, AVAILABILITY OF THE PSF, AND OF 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 DURING 1980

1.	Materials testing	
	Graphite and HTR fuel	18 %
	Structural materials	16 %
	LWR fuel	12 %
	Fast breeder reactor fuel	9 %
2.	Solid state and nuclear physics	18 %
3.	Radioisotope production facilities	16 %
4.	Miscellaneous	11 %

HFR Petten

STANDARD IRRADIATION FACILITIES

1. NON-FISSILE MATERIALS TESTING

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

Steel, unstressed, Na environment, tensile specimens, ø 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

- SECOND NEUTRON RADIOGRAPHY FACILITY (1981)
- REPLACEMENT OF THE REACTOR VESSEL (1982/83)
- DEVELOPMENT OF REDUCED ENRICHMENT FUEL (1981/84)
- STUDIES FOR A POWER INCREASE TO 60 MW (1982)
- NEW REACTOR AND EXPERIMENT DATA LOGGERS (1980/81)
- MEDIUM ACTIVITY LABORATORY (1984/85)
- NEW DISMANTLING CELL TRANSFER SYSTEM (1980/81)
- ENLARGED COMPUTING FACILITIES (1981)
- NEUTRON BEAM QUALITY IMPROVEMENTS (1982/83)

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

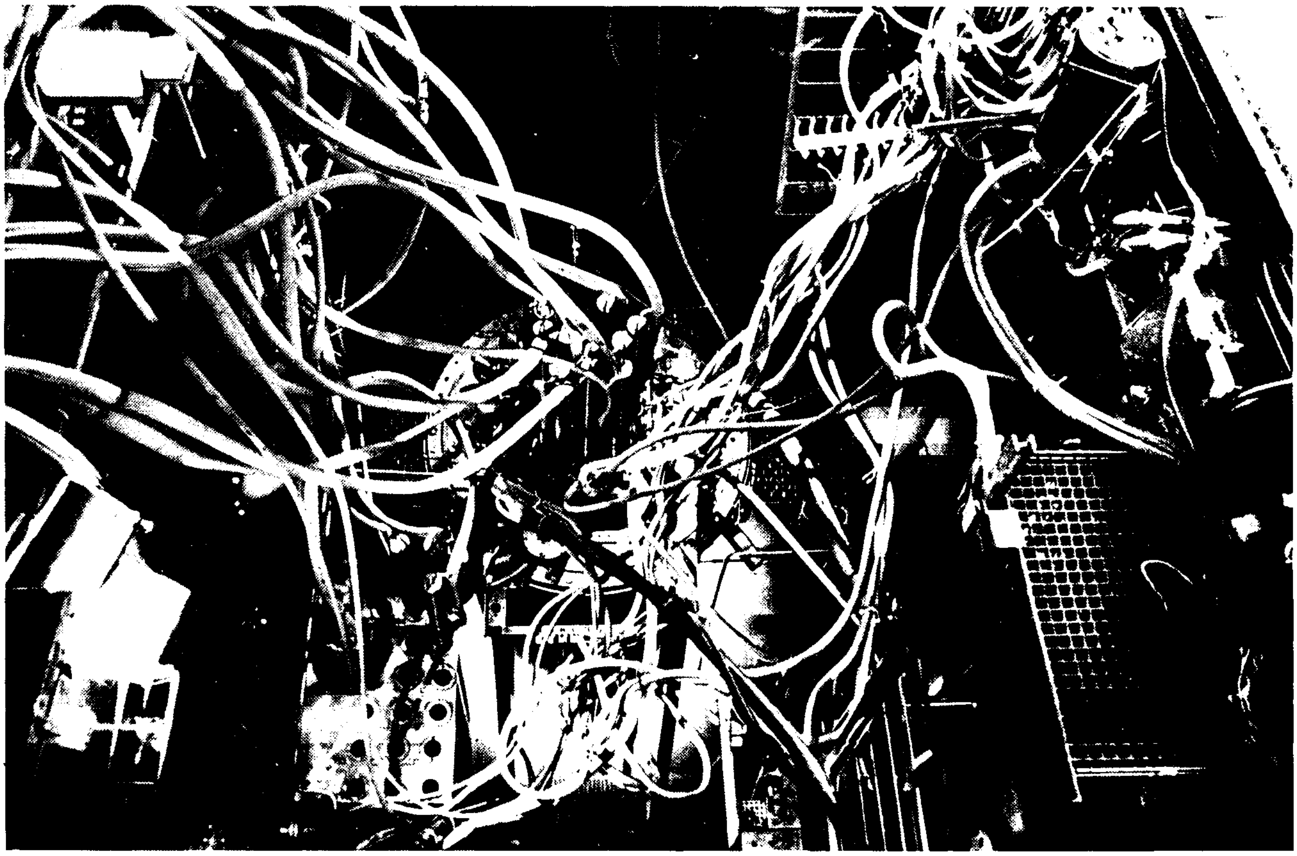


Fig. 1 HFR Petten. View into the reactor pool.

2. PROJECTS

2.1 HFR Operation and Maintenance

2.1.1 Objectives

Reactor operation during the second half year of 1980 aimed at

- high availability, in order to satisfy a consistent irradiation programme,
- inspection and maintenance during the summer outage in order to assure the technical health of the plant, preparation of the main overhauling work during the winter outage.

2.1.2 Methods

A precise operating calendar has been established by the end of 1979. It included for this half year a five weeks period for the summer stop, five reactor cycles of four weeks each and one week winter outage during Christmas and New Year. Technically the operation of HFR Petten follows usual MTR practice and has been described in several earlier publications.

2.1.3 Results

2.1.3.1 HFR Operation

Operation Summary

During the second half year of 1980 the HFR was in operation on full power of 45 MW for five reactor cycles. The yearly summer outage period started on July 1 and lasted till August 3, 1980. Following the last 1980 operational cycle, three days (December 22 to 24) were devoted to training of operations personnel in the handling of reactor power changes and disturbances.
The yearly maintenance shut-down period started on December 22 and will last approximately three weeks. In Table 1 a survey of the reactor operation schedule is given. The relation between shut-down and operation periods is given in Fig. 2, while Fig. 3 presents the relative occupation of the different HFR positions during this half year.
A detailed specification of the experimental utilization of the HFR during cycles 80.07 up to 80.11 is given in Figs. 4 to 8. Control rod position (bankwise) and reactor power during these reactor cycles are given in Figs. 9 to 13.

Table 1 Survey of reactor operation

Dates (1980)	Programme	Accumulated reactor power
01.07 - 03.08	Holiday period	
04.08 - 01.09	Cycle 80 . 07	1129 MWd
02.09 - 29.09	Cycle 80 . 08	1138 MWd
30.09 - 27.10	Cycle 80 . 09	1076 MWd
28.10 - 17.11	Cycle 80 . 10	788 MWd
18.11 - 21.12	Cycle 80 . 11	1369 MWd
22.12 - 31.12	Training and maintenance period	
	Total	5500 MWd

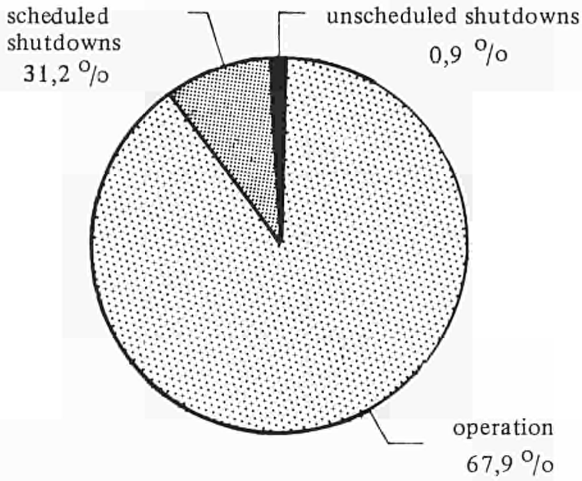


Fig. 2 HFR operation time distribution.

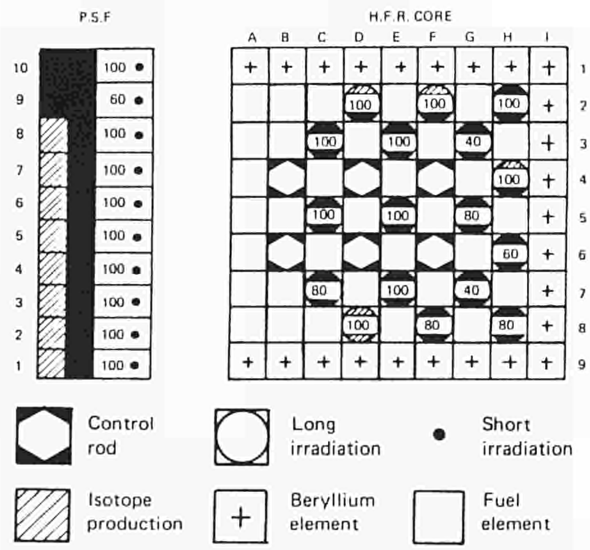
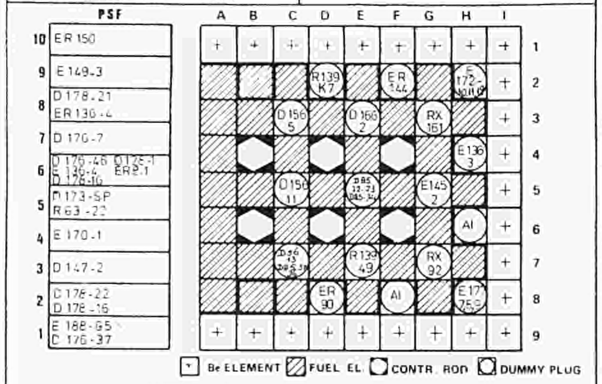


Fig. 3 Occupation of HFR irradiation positions in the second half of 1980 in %.

H.F.R. REACTOR LOADING FOR CYCLE : 80.07

START UP DATE 7-8-'80
END OF CYCLE 1-9-'80
REACTOR POWER : 45 MW



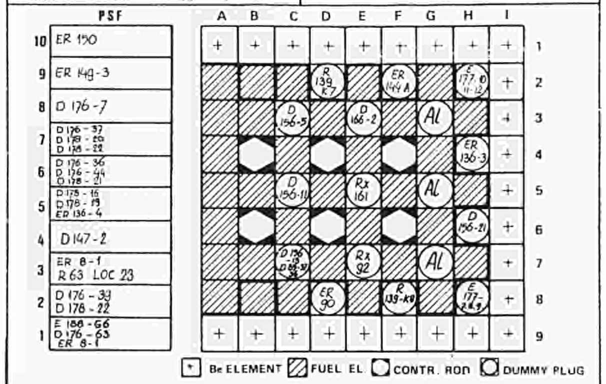
EXP. No.	DESCRIPTION	POSITION	*
D156-5	Graphite creep rig (DISCREET) TRIO 129-24,3 legs	C3	
D156-11	" " " " TRIO 129-9, leg I	C5	
D156-13	" " " " TRIO 129-23, leg I	C7	
D85-37	Intermed. temp.graphite TRIO 129-23, leg II	C7	
D85-36	Low temp. graphite TRIO 129-23,leg III	C7	
R139-K7	Steel irr. in Sodium (SINAS) TRIO 129-7,3 legs	D2	
ER90	Reloadable isotope facility (RIF)	D8	
D166-2	Graphite creep irradiation (CRIMP) REFA 140-15	E3	
D85-23-33	Low temp. graphite TRIO 129-17, leg I and II	E5	
D85-34	Intermedtemp.graphite TRIO 129-17,legIII	E5	
R139-49	Steel irr. in Souium(SINAS) TRIO 129-5, 3 legs	E7	
ER144	High flux facility for isotopes (HIFI)	F2	
RX161	Garma heating measurements (TRAMP)	G3	
E145-2	Austenitic steel irr. in Sodium(AUSTIN) TRIO 129-16, 3 13gs	G5	
RX92	HFR Al. irradiation	G7	
E177-10-11-12	Austenitic steel and tensile samples irr. in Sodium (FANTASIA) TRIO 129-13, leg I, II, III	H2	
ER136-3	Fissile isotope target (in-core FIT) JRC/ECN	H4	
E177-7-8-9	Austenitic steel and tensile samples irr. in Sodium (FANTASIA) TRIO 129-18, leg I, II, III	H8	
D176-37	Power ramp exp. BWFC supp. 21	psf 1	
D178-16	" " " " " 9	psf 2, 6	
D178-22	" " " " " 9	psf 2	
D147-2	Mixed carbide irr. (CAREL) PSF supp. 33	psf 3	
E170-1	Power cycling exp. (POCY) spec. supp.	psf 4	
D173-SP	FBR fuel pin irr. (DUELL), PSF supp. 34	psf 3	
ER8-1	High flux poolside isotope fac. (HFPIF)	psf 6	
R63LOC-22	Loss of cooling exp. (LOC)	psf 5	
D128-1	Fuel stack displacement BWFC supp. 12	psf 6	
E188G5	BWFC gamma test - BWFC supp. 15	psf 1	
ER136-4	Fissile isotope target spec. supp.	psf 6,8	
D176-46	Power ramp exp. BWFC supp. 19	psf 6	
D176-7	" " " " " 14	psf 7	
D178-21	" " " " " 15	psf 8	
E149-3	Borosilicate glass (BONI) spec. supp.	psf 9	
ER150	Neutrography camera	psf10	

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

Fig. 4 HFR reactor loading for cycle 80.07.

H.F.R. REACTOR LOADING FOR CYCLE : 80.08

START UP DATE : 04-09-'80
END OF CYCLE : 29-09-'80
REACTOR POWER : 45 MW



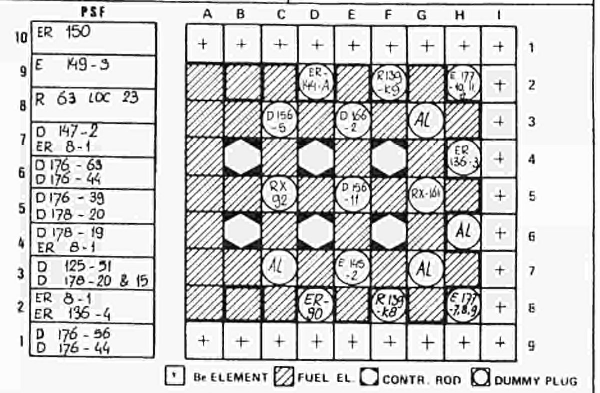
EXP. No.	DESCRIPTION	POSITION	*
D156-5	Graphite creep rig (DISCREET) TRIO 129-24, 3 legs	C3	
D156-11	" " " " " 129-9, leg I	C5	
D156-13	" " " " " 129-23, leg I	C7	
D85-37	Low temp. graphite " 129-23, leg II	C7	
D85-36	Intermed.temp. graphite " 129-23, leg III	C7	
R139-K7	Steel irr. in Sodium(SINAS) " 129-7, 3 legs	D2	
ER90	Reloadable isotope facility (RIF)	D8	
D166-2	Graphite creep irr.(CRIMP). REFA 140-15	E3	
RX161	Gamma heating measurements (TRAMP)	E5	
RX92	HFR Al irradiation	E7	
ER144	High flux facility for isotopes (HIFI)	F2	
R139-K8	Steel irr. in Sodium (SINAS) TRIO 129-6, 3 legs	F8	F
E177-10,11,12	Austenitic steel and tensile samples irr. in Sodium (FANTASIA), TRIO 129-13, leg I,II,III		
ER136-3	In core fissile isotope target	H4	
D156-21	Graphite creep rig (DISCREET) TRIO 129-21, 3 legs	H6	F
E177-7,8,9	Austenitic steel and tensile samples irr. in Sodium (FANTASIA), TRIO 129-18, leg I,II,III		
E188-G6	BWFC gamma test BWFC support 24	psf 1	
ER8-1	High Flux p.i.f. (HFPIF)	psf 1,3	
D176-63	Power ramp experiment BWFC support 24	psf 1	
D176-39	Power ramp experiment BWFC support 21	psf 2	
D178-22	Power ramp experiment BWFC support 9	psf 2,7	
R63-LOC23	Loss of cooling experiment (LOC) Special support	psf 3	
D147-2	Mixed carbide irradiation (CAREL) psf supp. 33	psf 4	
D178-16	Power ramp experiment BWFC support 15	psf 5	
D178-19	Power ramp experiment BWFC support 15	psf 5	
ER136-4	Fissile isotope target (FIT) Special support	psf 5	
D176-36	Power ramp experiment BWFC support 19	psf 6	
D176-44	Power ramp experiment BWFC support 19	psf 6	
D178-21	Power ramp experiment BWFC support 15	psf 6	
D176-37	Power ramp experiment BWFC support 21	psf 7	
D178-20	Power ramp experiment BWFC support 9	psf 7	
D176- 7	Power ramp experiment BWFC support 14	psf 8	
E149- 3	Borosilicate glass (BONI) Special support	psf 9	
ER150	Neutrography camera	psf 10	

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

Fig. 5 HFR reactor loading for cycle 80.08 .

H.F.R. REACTOR LOADING FOR CYCLE : 80.09

START UP DATE : 3 - 10 - '80
END OF CYCLE : 27 - 10 - '80
REACTOR POWER : 45 MW



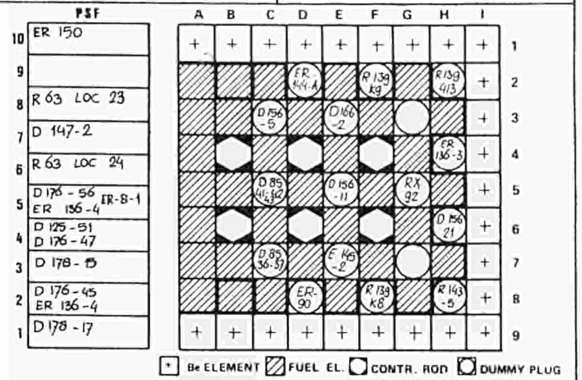
EXP. No.	DESCRIPTION	POSITION	*
D156-5	Graphite creep rig (DISCREET), TRIO 129-24,3 legs	C3	
RX92	HFR Al. irradiation	C5	
ER144	High Flux facility for isotopes (HIFI)	D2	
ER90	Reloadable isotope facility (RIF)	D8	
D166-2	Graphite creep irr. (CRIMP) REFA 140-15	E3	
D156-11	Graphite creep rig (DISCREET) TRIO 129-9 leg 1	E5	
E145-2	Austenitic steel irradiation in Sodium (AUSTIN) TRIO 129-16, 3 legs	E7	
R139-K9	Steel irr. in Sodium(SINAS), TRIO 129-20, 3 legs	F2	F
R139-K8	Steel irr. in Sodium(SINAS), TRIO 129-6, 3 legs	F8	
RX161	Gamma heating measurement (TRAMP)	G5	
E177-10,11,12	Austenitic steel and tensile samples irr. in Sodium (FANTASIA), TRIO 129-13, leg I,II,III		
ER136-3	In core fissile isotope target		
E177-7,8,9	Austenitic steel and tensile samples irr. in Sodium (FANTASIA), TRIO 129-18, leg I,II,III	H8	
D176-44	Power ramp exp. BWFC supp 19	psf 1-6	
D178-20	Power ramp exp. BWFC supp 9	psf 3-5	
D178-19	Power ramp exp. BWFC supp 15	psf 4	
D176-39	Power ramp exp. BWFC supp 21	psf 5	
D176-63	Power ramp exp. BWFC supp 24	psf 6	
D178-15	Power ramp exp. BWFC supp 15	psf 3	
D176-56	Power ramp exp. BWFC supp 24	psf 1	
D125-51	Power ramp exp. BWFC supp 21	psf 3	
D147-2	Mixed carbide irr (CAREL) psf supp 33	psf 7	
E149-3	Borosilicate glass (BONI) spec supp	psf 9	
ER136-4	Fissile isotope target (FIT)	psf 2	
R63LOC23	Loss of cooling experiment (LOC)	psf 8	
ER8-1	High flux poolside isotope facility (HFPIF)	psf 4-7-2	
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 80.09.

H.F.R. REACTOR LOADING FOR CYCLE : 80.10

START UP DATE : 29 - 10 - 80
END OF CYCLE : 17 - 11 - 80
REACTOR POWER : 45 MW



EXP. No.	DESCRIPTION	POSITION	*
D156-5	Graphite creep rig (DISCREET), TRIO 129-24,3 legs	C3	
D85-41	Intermed. temp. graphite TRIO 129-25, leg I	C5	F
42	Intermed. temp. graphite TRIO 129-25 leg II	C5	F
43	Low temp. graphite TRIO 129-25 leg III	C5	F
D85-36	Intermed. temp. graphite TRIO 129-23 leg I	C7	
37	Low temp. graphite TRIO 129-23 leg II	C7	
ER144-2	High flux facility for isotopes (HIFI)	D2	
ER90	Reloadable isotope facility (RIF)	D8	
D166-2	Graphite creep irr. (CRIMP) - REFA 140-15	E3	
D156-11	Graphite creep rig. (DISCREET) TRIO 129-9 leg I	E5	
E145-2	Austenitic steel irradiation in Sodium, (AUSTIN) TRIO 129-16, 3 legs	E7	
R139-K9	Steel irr. in Sodium.(SINAS) TRIO 129-20, 3 legs	F2	
R139-K8	Steel irr. in Sodium.(SINAS) TRIO 129- 6, 3 legs	F8	
RX161	Gamma heating measurements (TRAMP)	G3	
RX 92	HFR Al. irradiation	G5	
R139-413	Steel irr. in Sodium (SINAS) - REFA 170-8	H2	F
ER136-3	In core fissile target (FIT)	H4	
D156-21	Graphite creep rig (DISCREET), TRIO 129-22,3 legs	H6	
R143-5	Natrium steel irradiation (MONA)	H8	
D176-56	Power ramp exp. BWFC supp. 24	psf 5	
ER 8-1	High flux poolside isotope facility (HFPIF)	psf 5	
D176-45	Power ramp exp. BWFC supp. 14	psf 2	
D178-17	- - - - - 15	psf 1	
D176-47	- - - - - 21	psf 4	
D178-15	- - - - - 9	psf 3	
D125-51	Trolley test. BWFC supp. 21	psf 4	
ER136-4	Fissile isotope target (FIT) spec. supp.	psf 5,2	
R63 LOC24	Loss of cooling experiment (LOC)	psf 6	
D147-2	Mixed carbide irr. (CAREL), psf supp. 33	psf 7	
R63LOC23	Loss of cooling experiment (LOC)	psf 8	
ER150	Neutrography camera	psf 10	

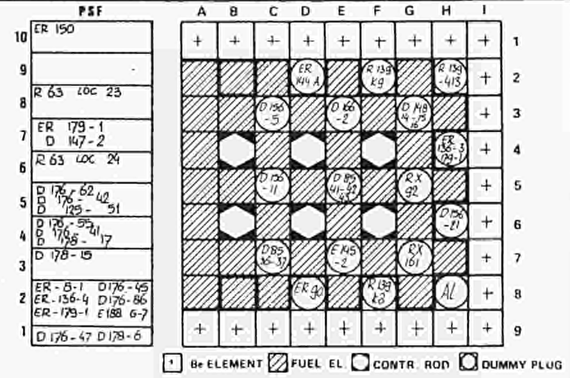
*REMARKS. F: first loading P: plug type exp. no vessel entry S: short irradiation

Fig. 7 HFR reactor loading for cycle 80.10 .

H.F.R. REACTOR LOADING FOR CYCLE : 80.11

START UP DATE : 19-11-'80
END OF CYCLE : 21-12-'80

REACTOR POWER : 45 MW



EXP. No.	DESCRIPTION	POSITION	*
D156-5	Graphite creep rig (DISCREET) TRIO 129-24,3 legs	C3	
D156-11	Graphite creep rig (DISCREET) TRIO 129-9, leg I	C5	
D85-36	Intermed. temp. graphite TRIO 129-23, leg I	C7	
37	Low temp. graphite TRIO 129-23, leg II	C7	
D166-2	Graphite creep irr. (CRIMP) REFA 140-15	E3	
D85-41	Intermed. temp. graphite TRIO 129-25, leg I	E5	
42	Intermed. temp. graphite TRIO 129-24, leg II	E5	
43	Low temp. graphite TRIO 129-25,leg III	E5	
E145-2	Austenitic steel irr. in Sodium TRIO 129-16, 3 legs (AUSTIN)	E7	
R139-K9	Steel irr. in Sodium (SINAS) TRIO 129-20, 3 legs	F2	
R139-K8	Steel irr. in Sodium (SINAS) TRIO 129- 6, 3 legs	F8	
D148-14-15-16	Mixed carbide irr. (CATRI) Special TRIO,3 legs	G3	
RX92	HFR Al. irradiation	G5	
RX161	Gamma heating meas. (TRAMP)	G7	
R139-413	Steel irr. in Sodium (SINAS) REFA 170-1	H2	
D156-21	Graphite creep rig (DISCREET) TRIO 129-22, 3 legs	H6	
ER8-1	High flux p.i.f. (HF-PIF)	psf 1	
ER70	Poolside rotating facility (PROF)		
ER90	Reloadable isotope facility (RIF)	D8	
ER150	Neutrography camera	psf 10	
ER144A	High flux facility for isotopes (HIFI)	D2	
ER136-4	Fissile isotope target (FIT)	psf 2	
ER136-3	" " " (incore FIT)	H4	
ER179-1	Fissile isotope target (MOLY)	H4,psf 2,7	
R63LOC23	Loss of cooling exp. (LOC)	psf 8	
R63LOC24	" " " " "	psf 6	
D147-2	Mixed carbide irr. (CAREL) BWFC supp. 33	psf 7	
D176-55	Power ramp exp. BWFC supp. 24	psf 4	
D176-62	" " " " " 24	psf 5	
D176-47	" " " " " 21	psf 1	
D176-42	" " " " " 21	psf 5	
D125-51	" " " " " 21	psf 5	
D176-45	" " " " " 14	psf 2	
D176-41	" " " " " 14	psf 4	
D176-86	" " " " " 25	psf 2	
D178-17	" " " " " 15	psf 4	
D178-6	" " " " " 15	psf 1,2	
D178-15	" " " " " 9	psf 3	
E188G7	BWFC gamma test " " 25	psf 2	

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

Fig. 8 HFR reactor loading for cycle 80.11 .

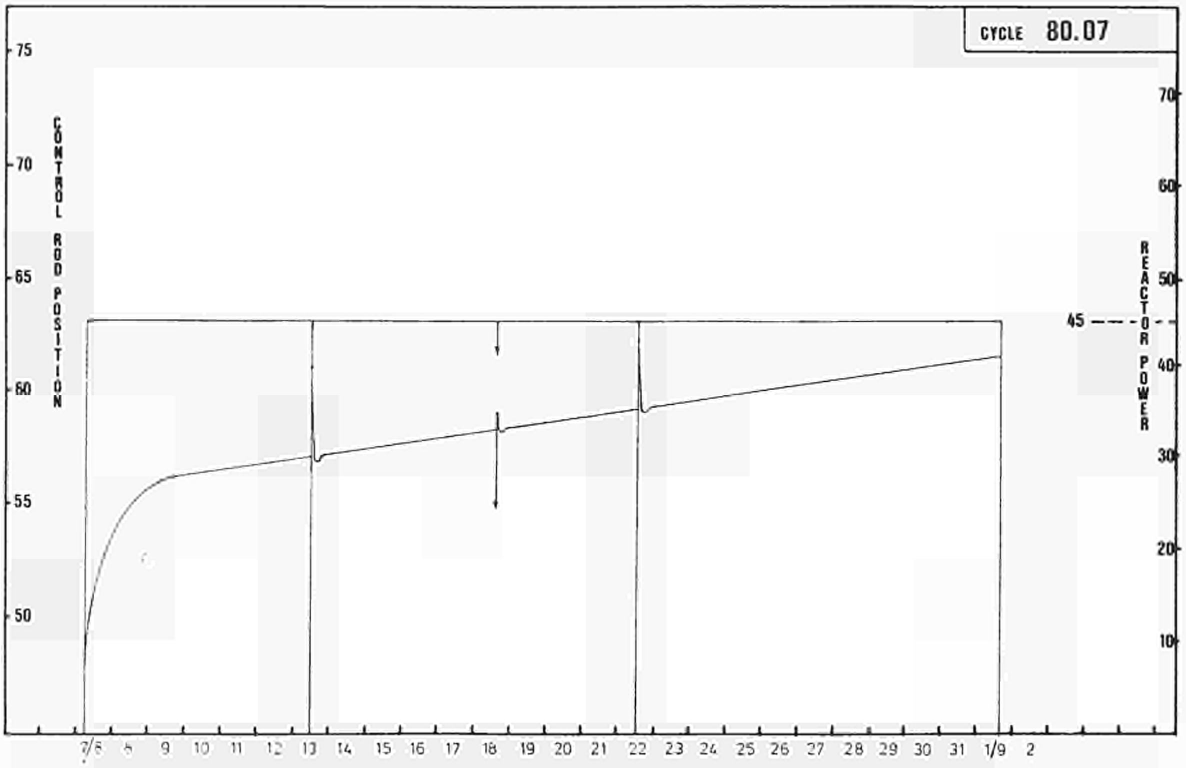


Fig. 9 HFR reactor power and control rod position vs. time, cycle 80.07.

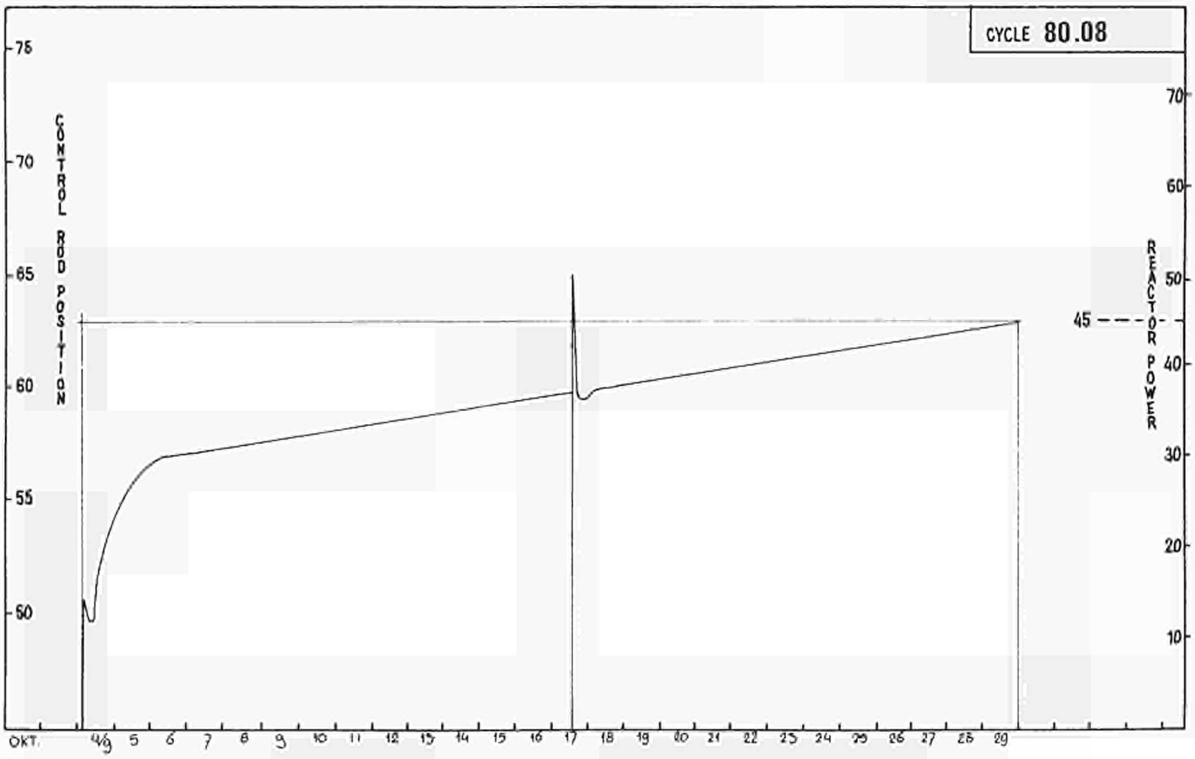


Fig. 10 HFR reactor power and control rod position vs. time, cycle 80.08 .

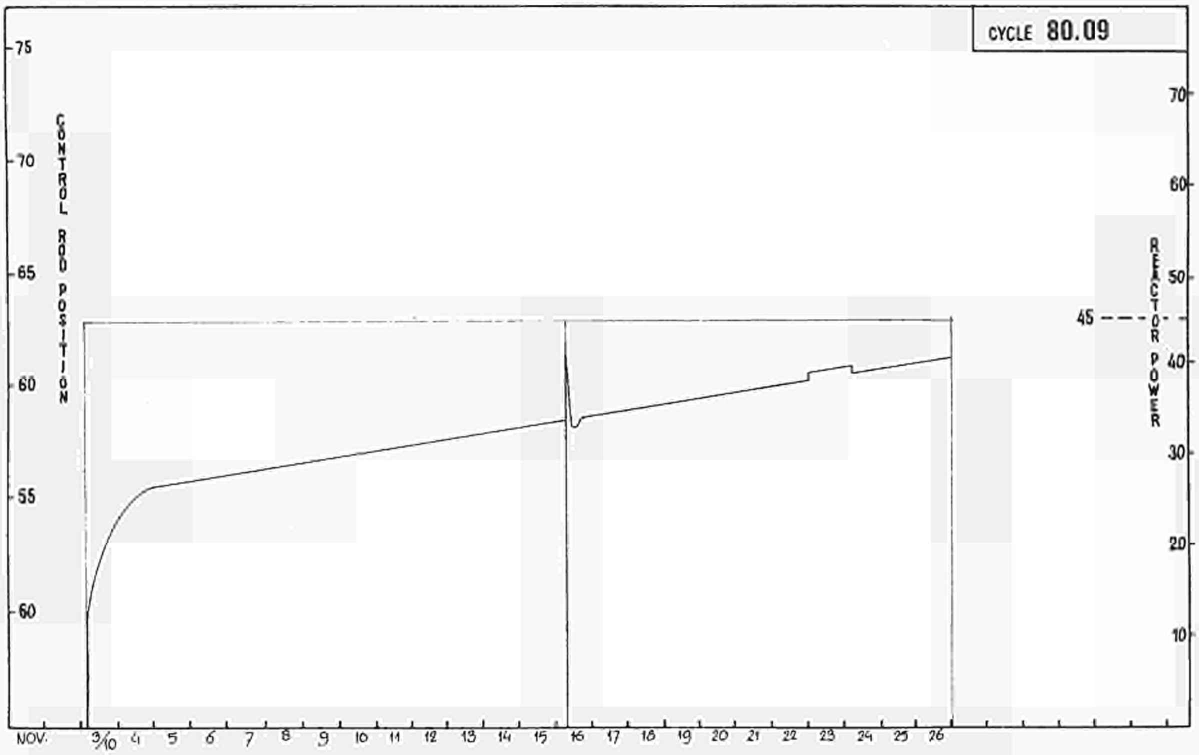


Fig. 11 HFR reactor power and control rod position vs. time, cycle 80.09 .

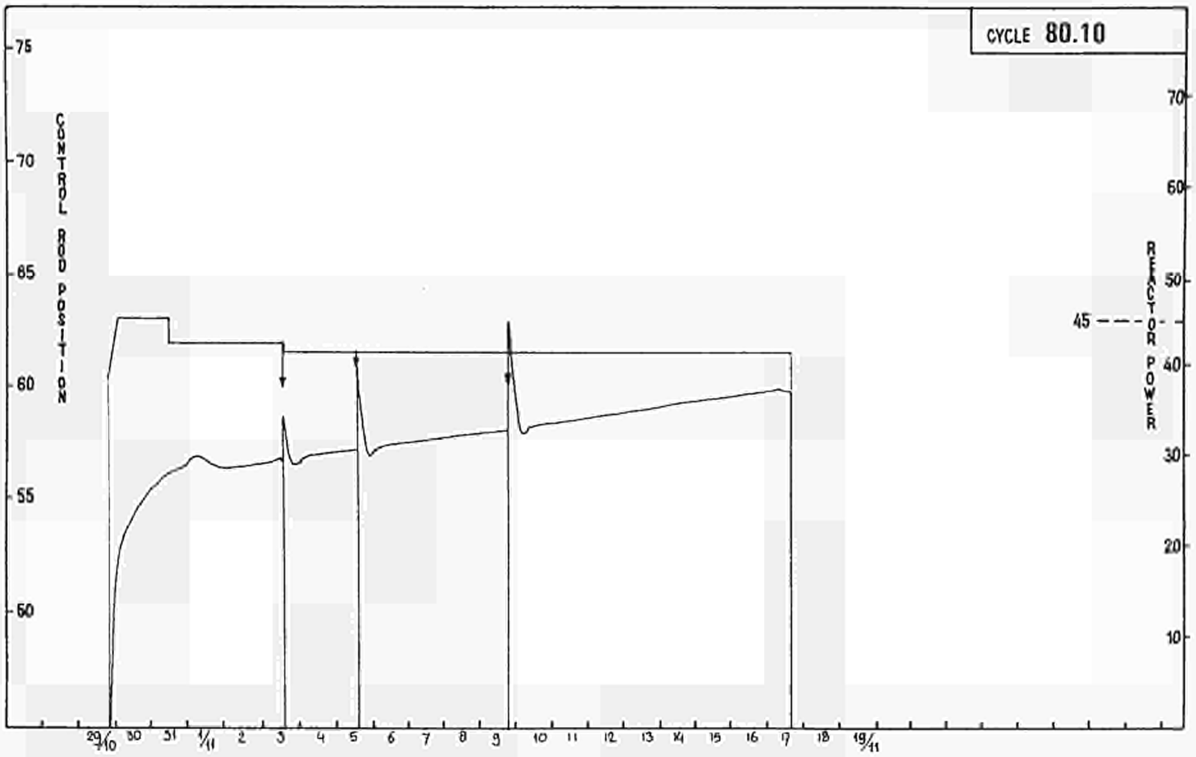


Fig. 12 HFR reactor power and control rod position vs. time, cycle 80.10 .

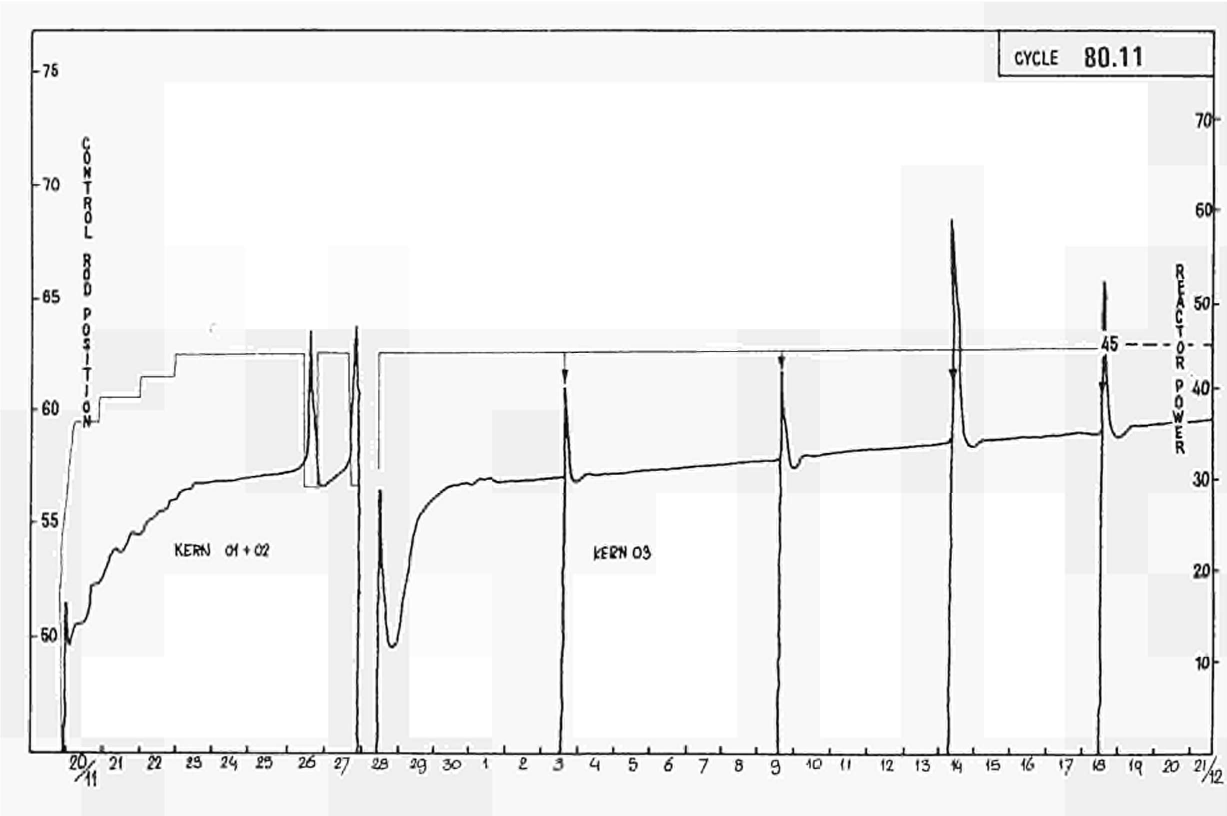


Fig. 13 HFR reactor power and control rod position vs. time, cycle 80.11.

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 12 unscheduled scrams, 1 unscheduled power decrease, while 2 scrams and 3 power decreases were scheduled in accordance with various experimental requirements.

One manual scram, necessitated by data logger malfunctioning and subsequent unloading of an irradiation experiment (D148), resulted in xenon poisoning and, thus, 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		1st half 1980	2nd half 1980
Integrated reactor power	MWd	6738	5500
Average reactor power	MW	44,8	44,08
Operating time	%	82,6	67,9
Unscheduled shutdown time	%	0,8	0,9
Unscheduled scrams	–	21	12
Unscheduled power decreases	–	1	1
Planned scrams	–	5	2
Planned power decreases	–	5	3
Fuel consumption	g ^{235}U	8416	6870
Stack radioactivity release	Ci(^{41}Ar)	275,7	191

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

Date	Hour	Duration hrs./min.	Disturbance type	Reactor system or experiment	Cause	Code	Core no.
4/9	09.51	01.35	MS	R139-K8	Manual reactor shut down for repair of Vertical Displacement Unit R139-K8	ME	80.08.1
26/11	11.41	02.49	MPD (30 MW)	Reactor core	Small foreign object on fuel element in core position G4	MR	80.11.1
27/11	12.42	00.05	MPD (30 MW)	D148.02	In reaction to unexplained temperature alarms	ME	80.11.1
27/11	13.16	00.27	MPD (0 MW)	ER 136.03	Loading of facility	ME	80.11.2
27/11	17.57	18.13	MS	D148.02	Removal of experiment and core reloading	ME	80.11.3

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

I : Instrument failure
O : Human 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.	Disturbance type	Reactor system or experiment	Cause	Code	Core no.
13/8	11.57	00.23	AS	E170.01	Probable cause, power disturbance during maintenance work	OE	80.07.1
18/8	14.15	00.21	APD (15 MW)	D147.02	Technical disturbance during pre-operational preparation of experiment	OE	80.07.1
22/8	11.28	00.32	AS	R63.22	Cooling water tube coupling accidentally disconnected during in-pool manipulations	OE	80.07.1
17/9	14.50	00.17	AS	Power supply	Mains failure	PS	80.08.1
16/10	01.00	00.25	AS	R63.23	Cooling flow disturbance due to in-pool manipulations	OE	80.09.1
3/11	14.53	00.20	AS	R63.23	During check out of experiment a wrong switch activated	OE	80.10.1
5/11	13.53	00.18	AS	R63.23	During cleaning work a switch accidentally activated	OE	80.10.1
9/11	19.48	00.16	AS	Reactor	Low primary pressure due to water leakage caused by malfunctioning of ER136 loading plug	FE	80.10.1
27/11	14.37	00.23	AS	D148.02	In reaction to unexplained temperature alarms *)	IE	
3/12	16.43	00.16	AS	Reactor interlock	Scram button touched accidentally	OR	80.11.3
9/12	16.44	00.16	AS	Flux channel 2	Water in protection hose of ionization chamber	IR	80.11.3
14/12	09.34	00.23	AS	Primary system	Low primary pressure due to water leakage, caused by malfunctioning of ER136 loading plug	FE	80.11.3
18/12	18.52	00.17	AS	D147.02	Bending of cooling water hose during in-pool manipulation	OE	80.11.3

AS : Automatic scram
APD : Automatic power decrease
MPD : 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

*) The perturbances caused by experiment D148.02 on November 27, 1980, have been identified as instrument failures (malfunctioning of a new data logger).

2.1.3.2 Maintenance and Modifications

Major activities carried out during the summer holiday period and during the winter maintenance outage are summarized hereafter.

a) Mechanical Installations

Reactor vessel and associated structures

- Visual inspection of reactor interior: No specific observations.

Cooling systems: Heat exchanger (HX)

- The observations, in October, of increased noises during normal full-power, full-flow operation at one of the main heat exchangers (HX 3) has led to a thorough investigation of the state of health of the three main heat exchangers.
  The heat exchangers are of the straight tube type (see Fig. 14), with aluminium ("alclad") pipes between two double pipe plates. The primary cooling water enters the HX at the outside of the pipes at one end of the heat exchanger and is reversed several times by four baffle plates and leaves the HX at the other end. Secondary cooling water flows inside the pipes with flow reversal in each of the end cover compartments.
  The rattling is apparently caused by mechanical contact between the vibrating pipes and one or more of the baffle plates.
  Slight throttling and redistribution of the primary flow over the three HX's led to elimination of this rattling noise but as the mechanical reliability of the HX's was somewhat in doubt several investigations have been and are being made.
  By removal of the primary piping connections and by drilling small holes in the outer shell of the most suspect HX the play between baffle plates and some of the outer pipes could be measured. For this purpose a special feeler gauge had been made and tried out at a model set-up. Visual inspection by means of an endoscope and feeler gauge measurements indicated that the play had considerably increased e.g. from 0,4 to 1,2 mm at the flow inlet and to 0,7 mm at the flow outlet section of the HX. In the center section, where only visual inspection could be carried out, no increase in play could be observed.
  By means of a mechanical bar which was inserted into the pipes from one of the end cover compartments (Fig. 15) vibration measurements were carried out up to full primary flow. Though clear indications of vibration were obtained no conclusions as to the exact nature and location of the rattling noise have yet been reached.

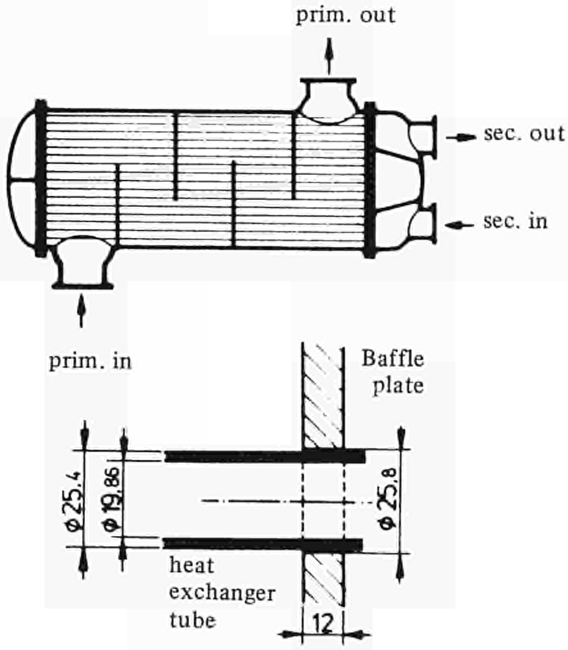


Fig. 14 Schematic view of HFR main heat exchangers.

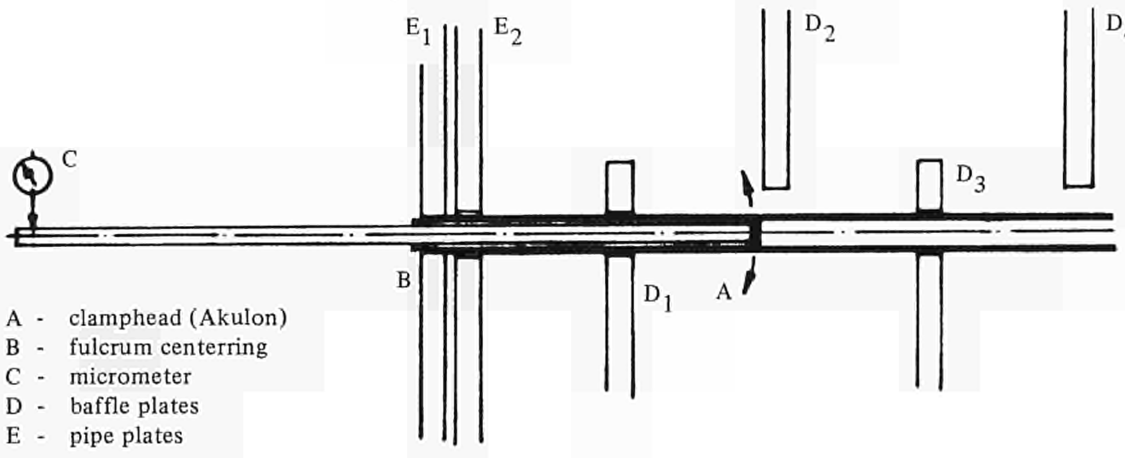


Fig. 15 Set up of heat exchanger pipe vibration measurements.

Similar measurements at HX 2 did not show up any clear pipe vibrations. However, also in this heat exchanger rattling noises have been observed and an increase in play between pipes and baffle plates has been measured.
Future measurements will be directed towards more detailed assessment of vibration characteristics and origin, and towards a non-destructive check on possible pipe wall damage.
As precautionary measure primary cooling flow has been slightly reduced in order to eliminate rattling.

o Internal inspection of the underground section of the secondary cooling circuit between secondary and primary pump building has only revealed some local erosion and corrosion effects. Repairs were made to the walls of the concrete sections. Some components in the joints which isolate the reduction piece from the adjacent pipe sections have been replaced.
Some difficulties were experienced in getting the pipes drained at the lowest part of the circuit, but hopefully this has not affected the quality of the concrete repair work.

o Adjustment of automatic secondary inlet filter rotating chain in order to reduce mechanical overload problems.

o Cleaning of chlorine dosage input piping to secondary cooling circuit in order to restore full nominal flow characteristics.

o Installation of protection tube around "LFR" connection joint in secondary cooling outlet circuit. Protection of this joint, at which several non-HFR waste water circuits are connected to the secondary outlet tubing, is necessary because of the dynamic behaviour of the sand dune profile in this area.

o Repair of secondary pump outlet valve no. 1. Due to internal wear and vibrations, the locking pin between spindle and gate had loosened and flushed away, resulting in the valve gate having lodged in its bottom position. Inspection of outlet valve no. 2 did not reveal similar problems.

o Replacement of rubber gaskets of pool door no. 2.

Off gas and ventilation systems

o Flow resistance and absorption tests carried out at off-gas filters.
Absorption capacity was satisfactory (> 85 $^{o}/o$ for methyl iodide) but due to high flow resistance (280 mm WG) the complete no. 1 filter unit, consisting of 2 absolute and 4 carbon filter units, was replaced. Subsequent measurements on individual filter units revealed that the increase in filter resistance could be attributed almost fully to the absolute filters (Δp increase from 11 to 215 mm WG at 2250 m^3/h).
In future cases the - rather expensive - charcoal filter units will not be replaced unless their absorption efficiency for methyl iodide is reduced to below 80 $^{o}/o$ (fresh filters: 99 $^{o}/o$). The corresponding absorption efficiencies for elementary iodine are obviously very much higher.

o Off-gas blower and filter section in reactor hall basement isolated from remainder of basement by double stone walls. Tests have indicated that the resulting poorer ventilatation conditions have not led to much higher blower temperature (increase < 5 ^{o}C)

Buildings

o Containment building leakage tests carried out with considerably improved measuring set-up and possibility for reference leakage check (see par. 2.1.3.5).

o Verification of proper roundness of track of reactor building rotating crane. Maximum diameter deviation determined between + 7 mm and – 4 mm, which is acceptable and does not deviate from preceeding checks.

o Internal inspection of reactor building water lock. Both 8" and 12" connection pipes require anti-corrosion treatment during forthcoming extended maintenance shut-down.

b) Instrumentation

Reactor Instrumentation

o Replacement of signal cable and protection hoses on start up channel 2, period channels 1 and 3, safety channel 2, the automatic control channel and the linear channel.

o A study has been started to investigate the possible installation of a redundant second safety channel system, fully separated and independent from the present systems. The installation of SPN (self powered neutron) detectors as near as possible to the reactor core has been envisaged, connected to amplifiers in the sub pile room with direct action on the control rod magnets.

Process Instrumentation

o Temperature measurements.

One of the three thermowells in the reactor outlet cooling water system has broken off, probably due to mechanical vibration-induced fatigue. The broken part of the detector well was found in the decay tank. All thermowells in the primary cooling water circuit have been replaced by a modified type with a welded flange connection and a larger diameter of the mechanically most vulnerable section.

o Hot drain tank level measuring system

After some starting problems the acoustic level measuring system on tank no. 1 proved to be accurate and reliable. Similar measuring systems for the remaining tanks have now been ordered.

Miscellaneous

o Installation of a new air tight penetration block for coaxial cables in the reactor building wall.

c) Electrical Installations

o Supply Systems

- Replacement of burnt transformer unit in VZO-2 (failure-free) power supply unit (caused by a defect controller).
- Local characteristics of high voltage transformers determined. Overload aspects discussed with utilities.
- Relocation of power supply boxes in reactor building and several laboratories.

o Reactor systems and components

- Primary pump field break switch coils rewound, new parts manufactured, wiring renewed. As these switches are of a now obsolete design, commercial spare parts are no longer available.
- Power switches for emergency pumps and emergency lighting systems inspected and overhauled.
- Warm and hot drain pumps interlock systems modified, permitting remote cut-off in case of reactor containment isolation (TMI-2 lesson !).
- Overhaul of secondary cooling pump no. 4, CO_2 blower and Reverse Osmose Installation flush pumps.

o Miscellaneous provisions

- New halogen type spotlights installed adjacent to storage pools.
- Various improvements and extensions.
- Vehicle airlock switching system overhauled; all clamping strips renewed.

d) General irradiation facilities

o Isotope production and activation analysis facilities

- Mechanical switch in underground pneumatic rabbit conveyor piping replaced by a photo cell unit.

o Cooling water supply and distribution systems

- Clean up of PSF cooling circuits. Radiation levels reduced from 20 - 40 to $<$ 2 mR/hr.
- New low flow system ordered.
  The system consists of a central switch and indication panel with the possibility to connect the flow contacts to the reactor safety system in either the scram or the APD action mode.
  Status indication of the switches and flow contacts is foreseen on the local panel and on the HFR control room panel.

o Gas supply and distribution systems

- "Oxisorb" and associated valves at main gas service station replaced. Preparations for installation of new humidity sensor.
- New TRIO gas-supply panels connected to standard alarm systems.

o PSF support and trolley facilities

- 4 new fast trolleys assembled and installed, 4 defect trolleys renewed and repaired.
- Prototype "push-pull" trolley assembled, installed and tested (see paragr. 2.3.3.7).

o Data acquisition and processing systems

- Two of the new standard datalogger cabinets have now been assembled and will be installed during the December/January maintenance shut-down. At the same time the signal connection box (COBO) will be replaced.
  For the further implementation of the new datalogger generation see time-schedule, Fig. 16.

The central data collection system "DACOS" has been taken in use with one data logger for irradiation experiments and one for the reactor parameters. The software of the DACOS system has been tested for the extension with more data loggers. A special program has been written to generate a survey of the actual PSF situation on the screens. The distance to the core and the type of the irradiation experiment can be displayed.
The hardware of the computer system has been extended with a third disc-unit (type RLo1) and with a second 8 lines multiplexer. In total 16 in/output lines are now available for data loggers and visual display screens.

In addition to the main task of data collection it has been decided to extend the DACOS system in such a way that also a large part of the off-line data analysis can be performed by the system. Therefore a larger central processor has been ordered and will be installed in March 1981.
This new computer will be a VAX 11/750 with two disc units of 28 Mega bytes. A drum plotter will also be connected to the system, to provide plots of temperatures vs. irradiation time, etc.

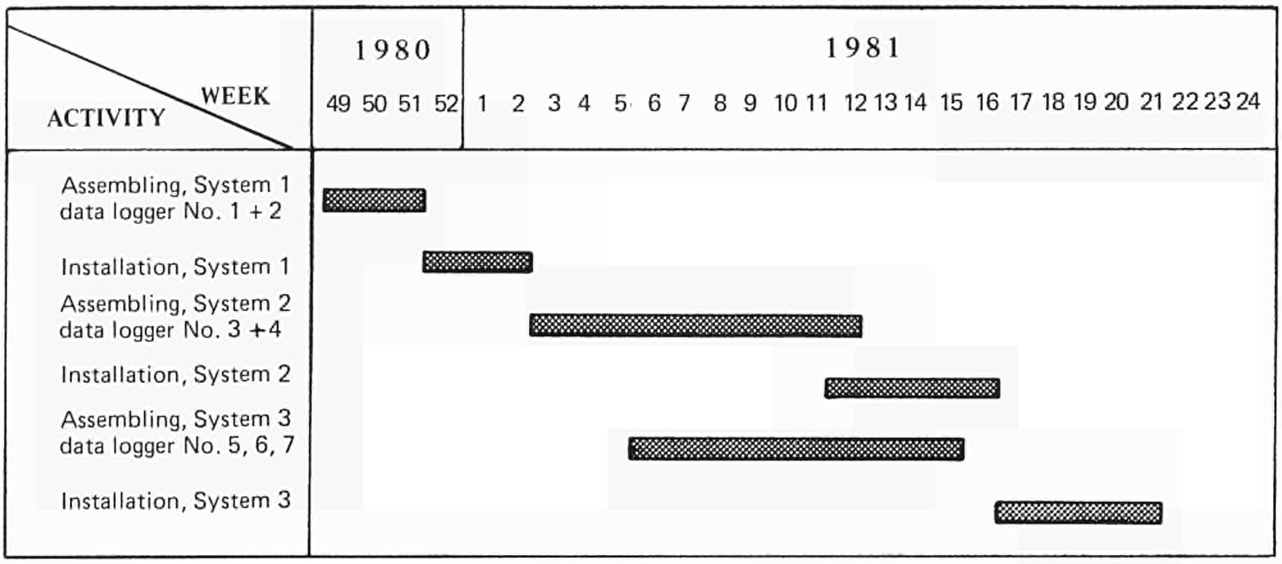


Fig. 16 Data logger installation time schedule.

2.1.3.3 Nuclear and Technical Support and Development

a) 1980 flux density characteristics

The yearly flux density mapping irradiation series has been performed on September 2, preceding HFR operation cycle 80.08.

Vertical thermal and fast flux density distributions have been measured with cobalt and nickel wires in all fuel assemblies and 10 experiment positions (C3, C5, D8, E5, E7, F2, G5, G7, H6 and H8). The wires were irradiated during 1 hour at an HFR power of 500 kW (control member setting 500 mm). During a second irradiation of 1 hour at 500 kW, measurements were performed in all 10 PSF positions at distances of 25, 50, 85 and 135 mm from the core box wall. Cobalt and nickel foils were used as neutron activation detector. The control member setting during the second irradiation was 565 mm.

In order to realize a control member setting comparable with the settings during normal HFR operation and to realize an accurate power calibration, the first irradiation was preceded by an irradiation, partly at 45 MW, partly at 500 kW, and a waiting time thereafter of 6 hours.

Special attention was given to the flow rate of the primary cooling water and the indication of the linear channel in order to determine an accurate conversion factor from a reactor power of 500 kW to 45 MW. In all positions at centre line fuel and in core position D5 and experiment position E5 in 17 positions, small foils of 20 $^{o}/o$ enriched uranium were introduced as second thermal neutron flux density detector.
The results will be presented in the 1981/82 version of report EUR 5700 e (HFR characteristics).

b) Adverse neutron flux condition at certain beam tubes

High density irradiation facilities, such as steel irradiation experiments, require in-core experiment positions with low gamma heating in order not to exceed maximum sample temperature limits. For this reason core positions H2 and H8 are often selected for these experiments.
The presence of steel irradiation experiments in H2 and H8, however, results in a considerable flux depression in certain beam tube experiments (see Fig. 17). In order to investigate whether this adverse effect can be compensated somehow, the effect of several alternative loading configurations has been studied.

From all alternatives considered, the core with experiment positions in E1 and E9 and Be reflector assemblies in H2 and H8 is, in principle, feasible. The improvement in thermal flux densities in the beam-tubes HB1, 2, 5 and 6 is about 30 $^{o}/o$. In the new experiment positions E1 and E9 the nuclear heating is even lower than that in H2 and H8 while at the same time the fast and thermal flux densities are higher.
However, it would be technically complicated to make the positions E1 and E9 accessible for standard experiments. Another disadvantage is the reduction of about 8 $^{o}/o$ in the flux densities in the isotope irradiation positions D2 and D8 and in the PSF.

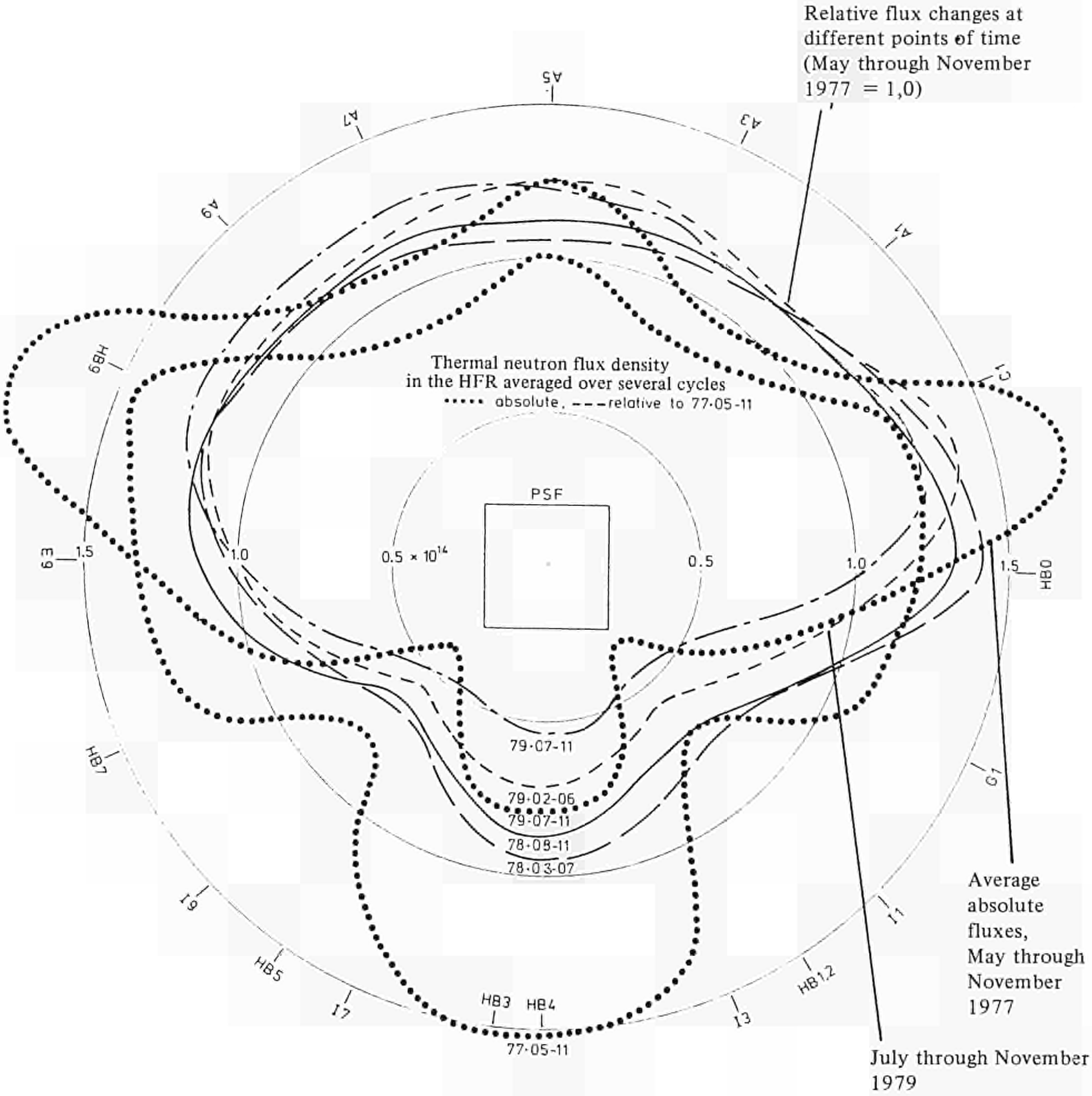


Fig. 17 HFR beam tubes. Polar diagram of thermal neutron flux densities showing variations with respect to the 1977 situation.

c) *Nuclear heating characteristics of the HFR-core*

The nuclear heating measurements with TRAMPSTICK has been continued for different core positions. In Fig. 18 the vertical distribution of the nuclear heating is given for core positions E5, G3, G5 and G7. It has been observed that the measured value of the nuclear heating is sensitive to the orientation of TRAMPSTICK. A 120 degree rotation gives different results especially in the lower regions of the core. This indicates that, depending on position, local nuclear heating gradients exist which are of the order of 0,2 Watt per gram per centimeter.

The nuclear heating has also been measured above the core. The distribution is the same for aluminium, graphite and stainless steel samples. It shows that in the region up to 30 cm above the core fuel, the distribution can be calculated from the expression $P = P_o\, e^{-0{,}084\, d}$ with P_o being the known nuclear heating value at the upper end of the fuel at the experiment positions and d being the distance in cm above the fuel core.

It is believed that the expression is valid for all experiment positions.

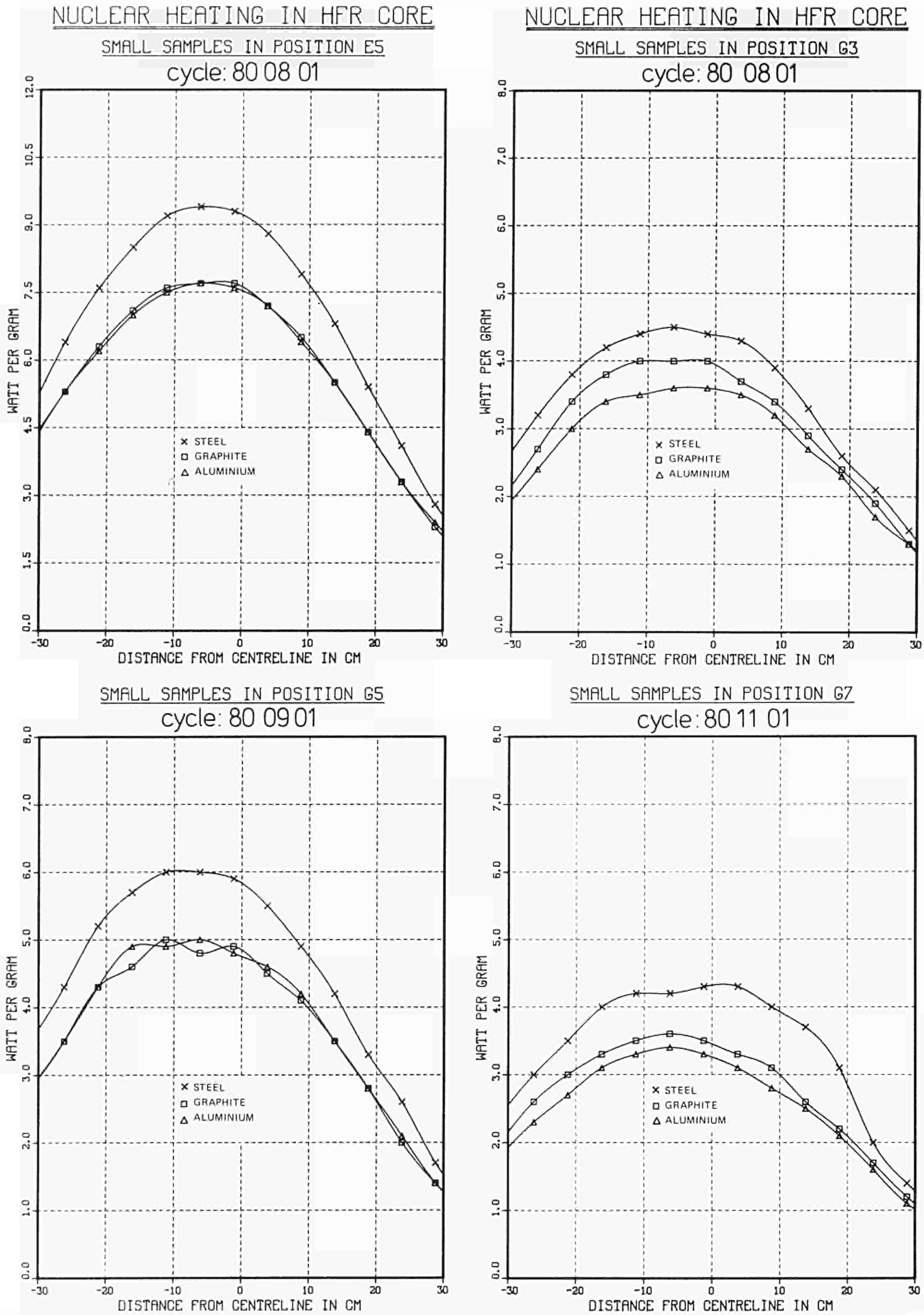


Fig. 18 Nuclear heating measurements in HFR core positions using a small-sample calorimeter ("TRAMPSTICK").

2.1.3.4 Fuel Cycle

a) Uranium supply

About 40 kg of highly enriched materials have been supplied to Europe in October 1980. They are needed for the 1981 fuel element and control rod manufature.

b) Local fuel element management

During the second half year of 1980, 71 new fuel elements have been delivered (see Table 5) of which 24 elements with the increased uranium contents of 405 g ^{235}U.

c) Spent fuel transports

34 depleted fuel elements and 8 depleted control rods have been transferred for reprocessing to the Savannah River reprocessing plant (see Table 5).
The decay power of this transport was 1432 W.
The next transport will include 13 spent fuel 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 Technical University (I.R.I.).
Fig. 19 a and b show the arrival of the Delft element transport in HFR pool no. 2.

d) Compact fuel storage

Specifications for compact storage requirements have been drawn up and several firms will be approached for a quotation for design and fabrication of compact storage racks.

e) Low enriched ("LEU") fuel test irradiations (Louise/Ludwig project)

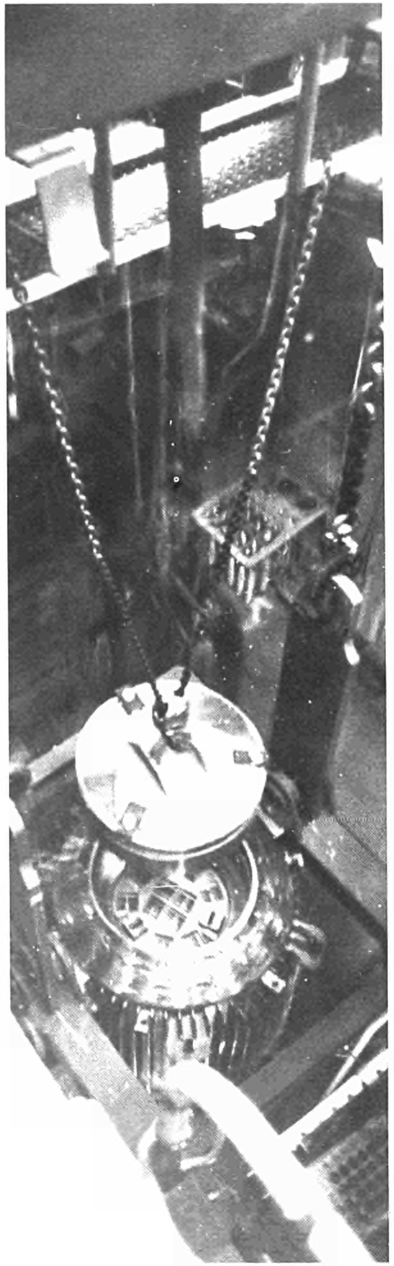
Preparations for the test irradiations of two UAl_x -Al and two U_3O_8-Al plate-type fuel elements, both 20 °/o enriched, have been continued.
They consisted of:

- o Discussion and final approval of the design of the elements. Special features of the design are :
  - same specific U^{235} content, slightly thicker fuel plates then present elements.
  - reduced number of plates in order to maintain present cooling channel width.
  - flux monitors incorporated in non-fuelled end plates.
  - square top section in order to facilitate access for neutron dosimetry and cooling channel inspection.
  - Cd-wires in side plates as burnable poison.
- o Drafting a design and safety report
- o Evaluation of flux dosimetry and cooling channel measurement tooling requirements. A special dosimetry "sword" has been designed for extensive low power measurements. For channel width measurements the acquisition of an ultrasonic channel gauge, developed by EG & G, is being considered.
- o Shipment of flux monitor tubes to the fuel element manufaturers.

The irradiations are now scheduled to start in the summer of 1981.
The international situation of reduced enrichment fuel development for research and test reactors has been reviewed in a meeting at the Argonne National Laboratory, on November 12 - 14, 1980.

Table 5 Fuel element and control rod movements, 1979/80.

	First half 1979	Second half 1979	First half 1980	Second half 1980
Transfer of depleted fuel elements	–	73	32	34
Transfer of depleted control rods	–	11	10	8
Average burn-up of transferred fuel elements	–	51°/o	52°/o	52°/o
Average burn-up of transferred control rods	–	55°/o	51°/o	51°/o
Delivery of new fuel elements	11	24	39	71
Delivery of new control rods	–	–	14	–
New fuel elements available for use at end of half year	26	18	27	62
New control rods available for use at end of half year	16	8	14	6
New fuel elements charged to core	36	32	30	36
New control rods charged to core	8	8	8	8
Fuel elements depleted	34	22	31	37
Average burn-up of depleted fuel elements	53°/o	51°/o	52°/o	51°/o
Control rods depleted	6	7	10	7
Average burn-up of depleted control rods	52°/o	49°/o	49°/o	45°/o



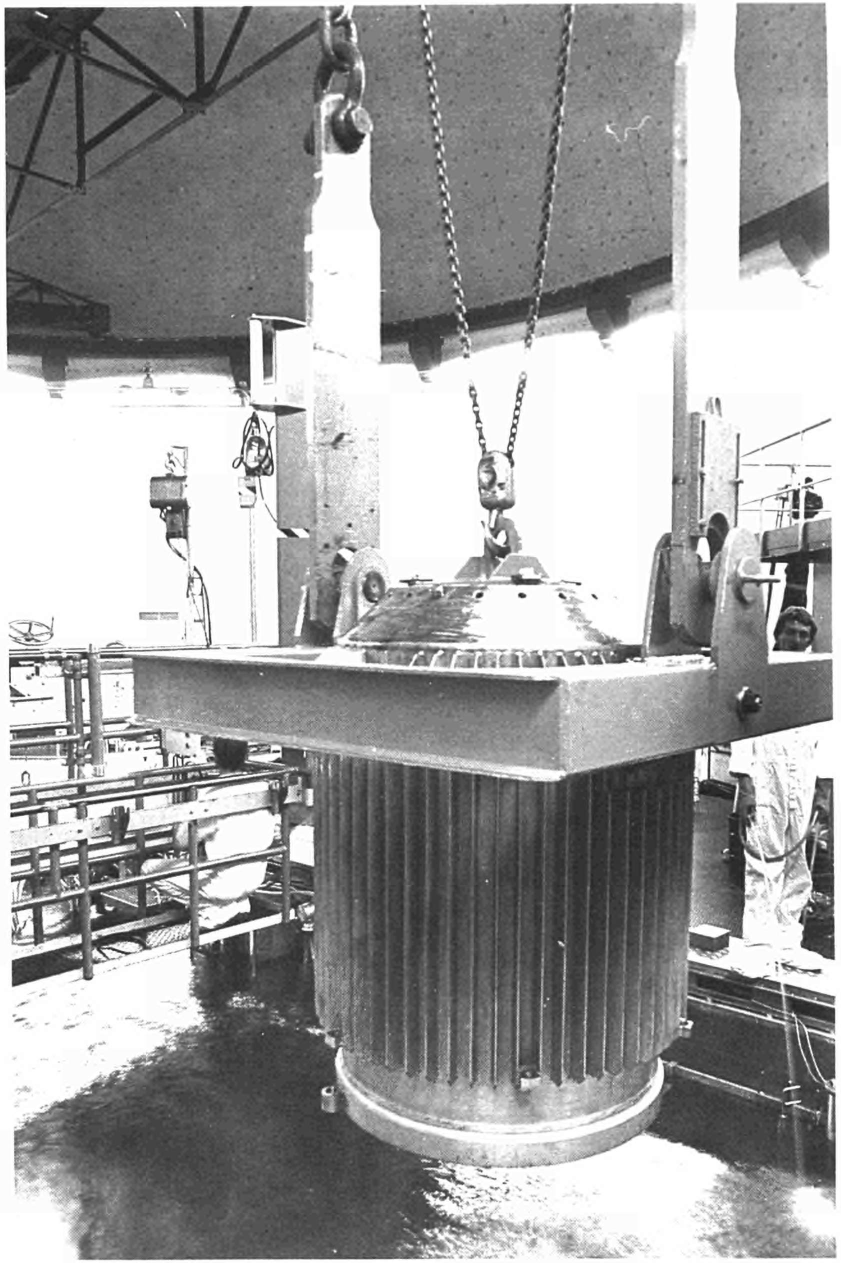


Fig. 19 Reception of spent fuel-elements from the H.O.R. reactor Delft, The Netherlands.

a) Container in HFR-pool no. 2, before element unloading

b) Container after unloading

2.1.3.5 Miscellaneous Tasks

Reactor containment building high pressure leakage rate test

According to an agreement with the licensing authorities a supplementary high pressure leakage test [1] with subsequent calibration run has been performed in July 1980.
Preceding the measurements local soap tests were performed at most penetrations. No major leakages were detected.
The containment was pressurized up to 0,45 bar overpressure. The actual measurements started at 11.54 hr on July 1 (see Fig. 20 and 21). The gross leakage rate which could be derived from the measuring data during the first 24 hours was - 0,0039 $\pm$ 0,0043 $^{o}/o$ per day.

From separate measurements on pressurized systems (cryostats, sweep systems, etc.) in the containment an inflow of 0.038 $^{o}/o$ per day was determined.
Combination of both results and extrapolating to 0.5 bar overpressure yielded an absolute leakage rate of 0,035 $\pm$ 0,004 $^{o}/o$ per day at 0,5 bar (outflow). The results is well below the allowable limit value of 0,1 $^{o}/o$ per day.

After this 24 hr. measuring period (see Fig. 21) a calibration test was carried out. Instead of a present leakage rate, as applied in Aug. '79, a stepwise gas release was applied this time. In this way the influence of interfering day/night effects was eliminated.
The amount of gas released in about 1 hour, as measured with a calibrated gasmeter and calculated with the applied computer program, was 124,0 $\pm$ 1,5 kg and 125,7 kg respectively. The relative difference between both figures (1,5 $^{o}/o$) is well below the maximum allowable deviation (10 $^{o}/o$).
An additional gas release of about 650 kg was used for calibration of the free volume of the containment. Evaluation of the result of this second calibration run has led to a correction of - 1,4 $^{o}/o$ with regard to the free volume (11950 m^3 becomes 11390 m^3).

The new measuring equipment and automatized data sampling and evaluation have functioned in a reliable way during this test. Since this complete test satisfies the official requirements the next integral high pressure leakage test will be performed in 1984.

2.1.3.6 Users' Services

a) *^{60}Co production in the HFR* [2]

Previous calculations showed that production of large ^{60}Co specimens in the HFR offers too many disadvantages in terms of reactivity costs and flux depression. Another possibility of the irradiation of Co grains of 1 mm diameter, 1 mm long and lined up in 48 longitudinal grooves milled into an aluminium carrier tube and covered by a thin walled Al cladding (see Fig. 22), has now also been examined.
A configuration has been assumed in which the carrier tubes are inserted in beryllium filler elements in irradiation positions B1, B9, C1 and C9.
According to calculations the specific activity yield would now be 3,5 to 4 times that of the previous set-up i.e. 100 Ci/g Co after one year (see Table 6).
The anti-reactivity due to the Co irradiation would be about 700 p.c.m., which is acceptable for reactor operation.

b) *Fission power in POCY (E 170)* [3]
(see paragr. 2.2.3.5)

The fission power of POCY has been calculated in order to confirm that a linear power of at least 540 W/cm in PSF 4, at 63 mm distance from the core box wall can be obtained.
The fuel consists of (U + Pu)O_2 with 10 and 20 $^{o}/o$ enriched ^{235}U in a configuration as shown in Fig. 23. The calculations yielded attainable linear powers of 1239 and 1184 W/cm for an enrichment of 20 and 10 $^{o}/o$ respectively, which means that the desired

Table 6 Specific activities in Ci per gram Co after irradiation times "T" in years.

Reflector position	Activity in Ci/g after time "T"					
	T = 1 year	T = 2 years	T = 3 years	T = 4 years	T = 5 years	T = ∞
B1	89,5	168,3	237,3	297,7	350,8	727,6
B9	93,7	176,2	248,4	311,7	367,2	761,8
C1	100,0	188,0	265,0	332,5	391,7	813,0
C9	99,7	187,5	264,3	331,7	390,7	810,6

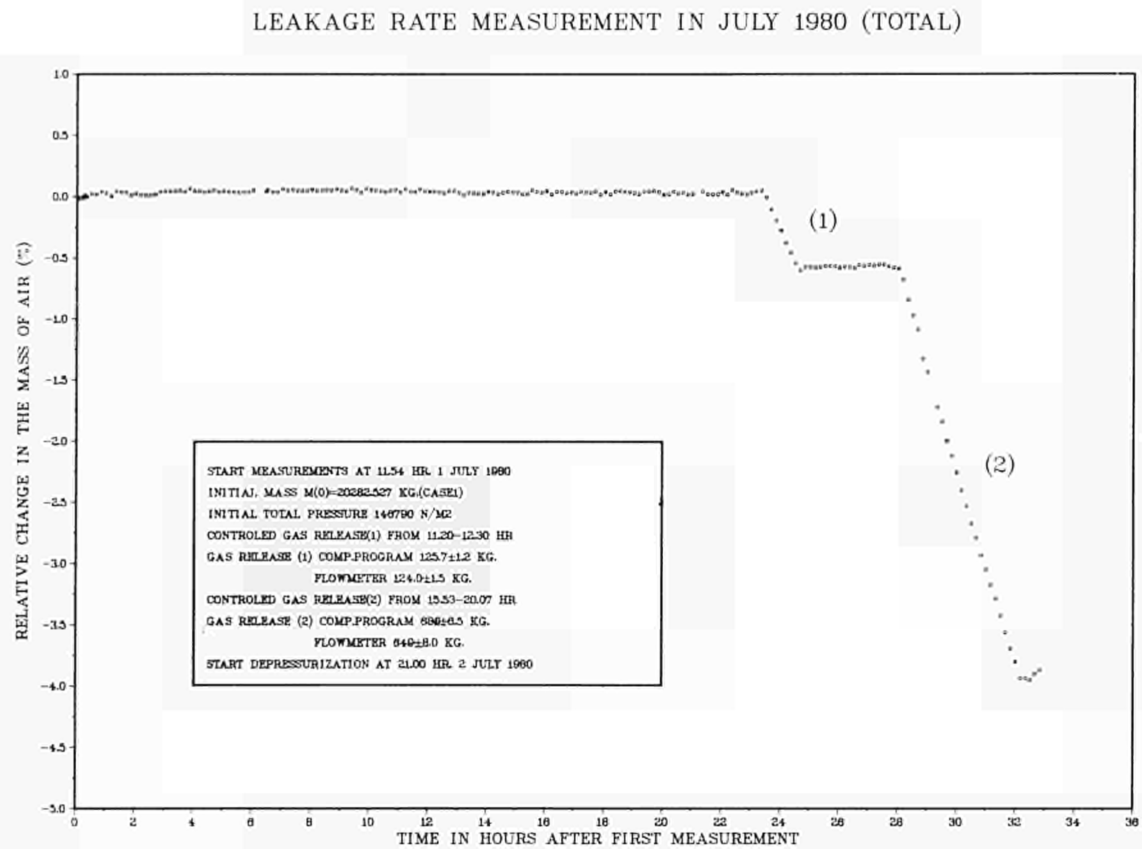


Fig. 20 Containment building leak tests. Survey of the measuring results in July 1980.

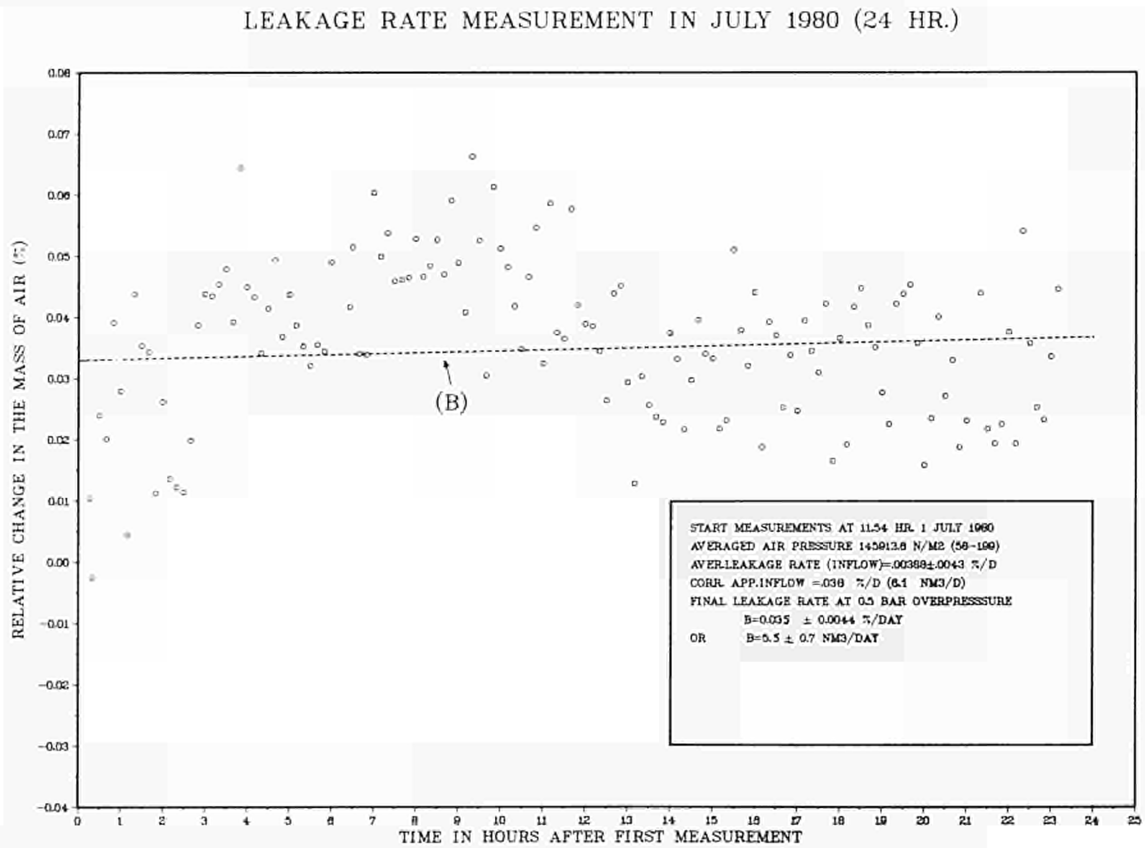


Fig. 21 Containment building leak tests. Results during the first 24 hr. period (undisturbed).

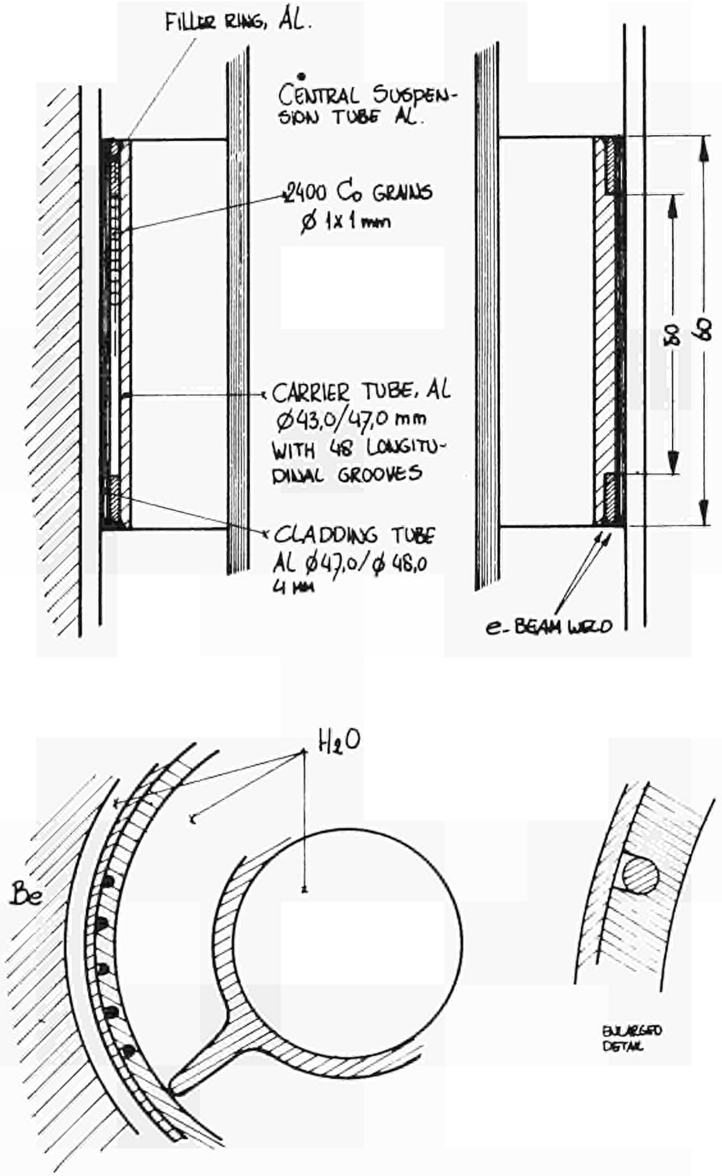


Fig. 22 Irradiation facility for cobalt grains of 1 mm diameter in 48 grooves in an aluminium tube.

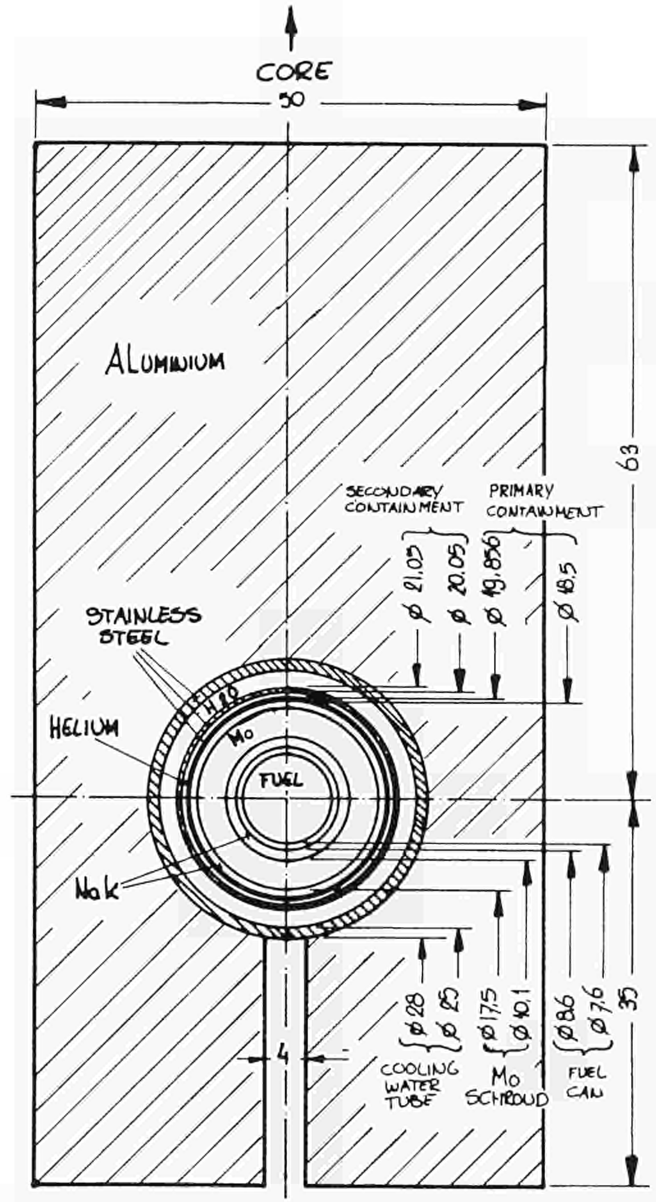


Fig. 23 Irradiation devices E170 ("POCY"). Cross-section of capsule on centerline of reactor core.

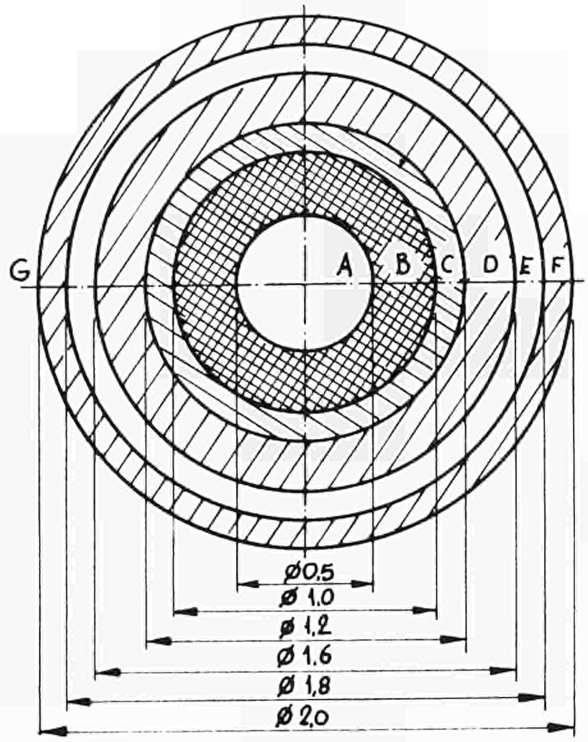
powers can be easily obtained in the mentioned positions.
The required irradiation time for a 10 °/o burn up of ^{235}U is 121 days (= 4,7 reactor cycles) at a power of 450 W/cm and 101 days (= 3,9 cycles) at a power of 540 W/cm.

c) *Fission power in E138.02 (BEST* [4] *, see paragr. 2.2.3.2)*

In the spherical fuel element BEST in the HFR position G7, calculations have been carried out to determine the generated fission power. The fuel consists of 1,4 g ^{235}U and 18,5 g ^{238}U in a spherical graphite sample with a diameter of 5,0 cm. The max. density of (UO_2 + C) is 2 g/cm^3.
The calculated fission power is 3095 Watt per spherical fuel element, with a contribution of about 10 °/o from epithermal neutron-induced fissions.

d) *Fission characteristics of E149 (BONI, see paragr. 2.2.3.6)*

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.
The irradiation target consists of a rod with a composition of 2 wt.°/o UO_2, 90 °/o enriched ^{235}U and simulated fission products. In order to get a homogeneous fission rate distribution the rod is surrounded by a cadmium shield (see Fig. 24).
HIP-TEDDI calculations have been carried out to calculate the number of fissions per cm^3 in BONI during the 1979 and 1980 irradiations periods.
The results are given in Table 7.



(measures in cm)

Material composition :

A	:	He
B	:	UO_2 + simulated fission products (2 °/o UO_2, 90 °/o enriched).
C	:	Cd
D + F	:	Al
E + G	:	H_2O

Fig. 24 Horizontal cross section of BONI experiment with simulated fission products, surrounded by cadmium and placed in the pool side facility.

Table 7 Number of fissions per cm^3 in the BONI rod.

Cycles	PSF position	Distance in mm	Irradiation time in days	Number of fissions per cycle in 10^{18}/cm^3
79.09	3	30	25.44	0.2212
79.10	3	30	24.74	0.2130
79.11	3	30	25.04	0.2208
80.03	1	25	25.70	0.1674
80.04	1	25	23.36	0.1546
80.05	9	25	25.92	0.2432
80.06	9	25	25.82	0.2476
80.07	9	25	25.09	0.2392

e) Optimization of DUELL absorber shield (see paragr. 2.2.3.5)

Various options have been selected and analyzed for a thermal neutron absorber shield which could provide better axial, radial and circumferential flux flattening in the DUELL irradiation facility. The following shielding configurations have been examined:

- without a neutron shield
- a stainless steel shield of 0,7 cm thickness
- a SS shield of 1,2 cm thickness
- a Co-SS shield of 0,7 cm thickness

The results are plotted in Fig. 25 and refer to a distance of 6 cm to the core box wall [5].
Also the influence of the distance to the core box wall was determined. The relative fluxes were computed for 3 distances for case b.

f) "H_2O"-power control of E172 facility ("Corrox")

As an alternative to the power control by means of the cadmium "minirods", the feasibility and effectiveness of changing the water/gas configuration around the Corrox-fuel pins has been studied [6], [7].

The principle idea is to fill up the room between fuel pins and filler element with thin S.S. tubes. These tubes are divided into groups, which can be filled with water or air separately. In order to estimate the possibilities of this concept two HIP-TEDDI calculations have been performed:

- Core 6130, CORROX in H6, all S.S. tubes and interspaces filled with H_2O.
- Core 6131, CORROX in H6, all S.S. tubes and interspaces filled with air.

The calculations have led to the following conclusions:

- o The maximum available power control range is about $\pm$ 12,5 °/o.
  This range covers the differences in flux conditions for the positions H2, H4, H6, H8, G3, G5, G7, F2 and F8.
- o The reactivity change due to filling up the S.S. tubes and interspaces with water is about + 300 pcm.
- o The vertical power distribution will not be affected by H_2O power control.

The power control range might be further enlarged by the application of a Boric acid solution instead of water.

2.1.3.7 Neutron Metrology Methods Development

a) Interlaboratory experiment REAL-80

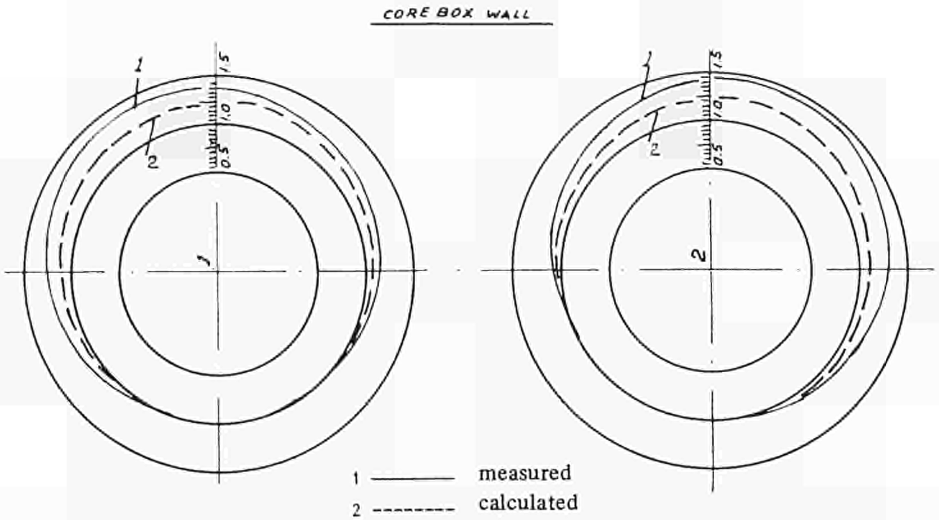
During the workshop session on "Adjustment Codes, Uncertainties and Input Needs" of the 3rd ASTM-Euratom Symposium on Reactor Dosimetry (held in Ispra, 1 - 5 October, 1979), the suggestion was made to organize a follow-up of the previous international activities on the intercomparison of unfolding codes.

It was felt that a study should be made on the uncertainty of integral parameters (displacement rates and activation rates), derived from neutron flux density spectrum information (based on experimental activation rates) by an unfolding procedure.
The exercise will have the code name REAL-80 (Reaction Rate Estimates, Evaluated by Adjustment Analysis in Leading Laboratories).

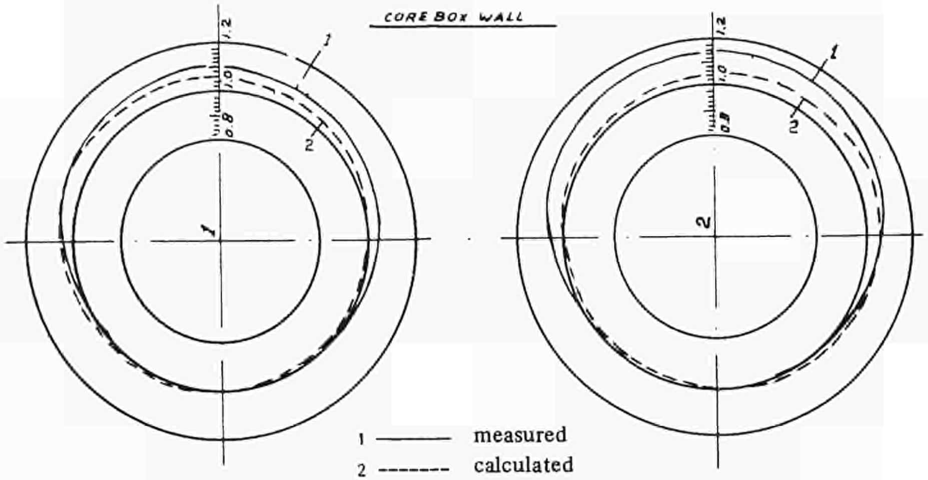
One of the candidate neutron spectra, as typical for a thermal research reactor in the REAL-80 exercise was the spectrum for HFR mid core position E5. The input spectrum was obtained as smoothed results from a two dimensional diffusion code.
Within the framework of REAL-80 calculations have been performed to find estimates for missing physical data for the variance-covariance matrix. The output of some unfolding codes depends on this variance-covariance matrix.
For this reason some calculations have been performed with different matrices to study their influence. The calculations were performed with the Petten version of the program STAY'SL and are reported in [8].
The E5 spectrum has been documented in [9] together with 5 other spectra : CTR, ^{252}Cf, ^{235}U, CFRMF and $\Sigma\Sigma$. For each spectrum flux density values are given in the ABBN group structure, together with flux density values for some special energy groups. Furthermore damage-to-activation ratio (DAR) values are given for graphite, steel and aluminium.

b) Cross section library PALPHA80

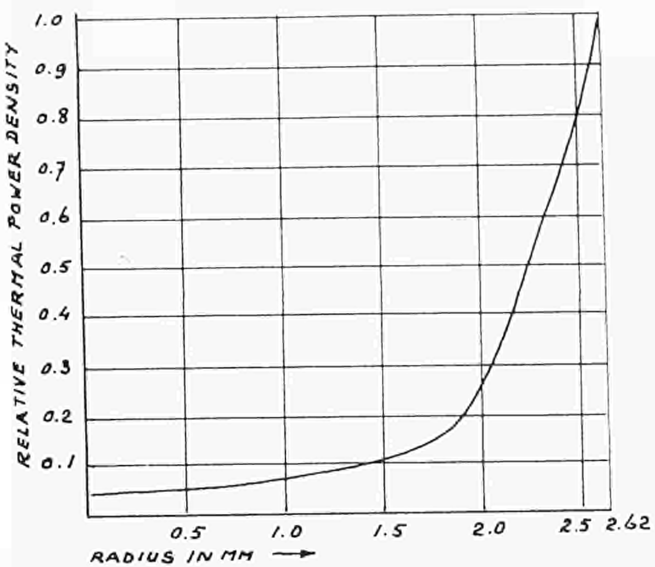
In report [10] a documentation is given of a cross section library called PALPHA80, which has been composed in order to be able to calculate the hydrogen and helium production in materials irradiated in neutron fields. The gas generation due to irradiation also causes damage effects in construction materials.
The library PALPHA80 contains cross section data for 37 elements of material, which produce hydrogen or helium under neutron irradiation. The cross section data have been taken from the ENDF/B-IV dosimetry file with aid of the program ENTOSAN.
The library has been used for the calculation of the number of displacements and the number of hydrogen and helium atoms per target atom for an irradiation of 250 days again in experiment position E5.
The results have been presented in [11].



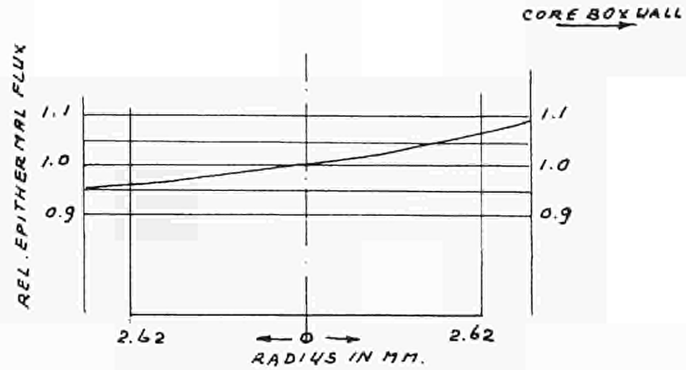
a) Circumferential relative thermal neutron flux distribution in Mo shroud around the DUELL fuel pins in an unshielded Al block of 5 x 5 cm^2 according to flux measurements and calculations.



b) Circumferential relative thermal neutron flux distribution in Mo shroud around the DUELL fuel pins in an Al block of 5 x 3 cm^2 provided with a SS shield of 0,7 cm thickness according to flux measurements and calculations.



c) Relative power density distribution in the DUELL high-enriched fuel pin according to the microflux calculation.



d) Relative epithermal flux (ϕ3) distribution in the DUELL high enriched fuel pin computed by HIP-TEDDI.

Fig. 25 Optimisation of the DUELL directional absorber shield.

2.1.3.8 HFR Vessel Replacement

a) Reactor Vessel Design

The design work on the new vessel is now complete and the final output of design drawings has been submitted by the contractor. A final draft of the design report has been prepared and will be issued, after review early in 1981.

b) Beam Tube Design

Preliminary calculations have been performed to establish the beam tube cooling requirements. The proposed design of the cooling system encloses the beam tube ends in a shroud plate running parallel to the core box wall. The space so formed is forced fed from the pool cooling water system.
Work on the remainder of the contract will be completed by June 1981.

c) Hydraulic Tests

Half scale hydraulic tests will be performed on a model of the new reactor vessel in order to optimise the primary coolant inlet baffle geometry and to establish the core flow distribution. The model and test rig were assembled in October and at the end of the reporting period leak tests had been completed and commissioning of the flow and laser doppler instrumentation were well advanced. The core discharge area has now been included in the model because the vessel design in this region is sufficiently changed from the present design to have an effect on the core flow distribution and the overall pressure drop. The first test run is scheduled for January 1981.

d) Reactor Transient Calculations

The ASME code includes requirements to perform calculations demonstrating the safe response of the reactor vessel under upset, emergency and faulted conditions. These abnormal conditions can be conveniently categorised as follows :

- reactivity faults (e.g. control rod withdrawal at power, cold water injection, experiment failure, over power transient due to start-up accident)
- plant failures (e.g. loss of coolant, electrical power supply failure, NaK or Na filled capsule failure)
- maximum credible accident resulting from partial fuel blockage (see below).

For analysing the HFR core behaviour under transient conditions the computer program AIROS II-A was used. AIROS II-A solves the space-independant reactor kinetics equations and in addition provides for the determination of reactivity by solving the discretized equations which represent the spatial heat and mass transfer models for several fuel channels, including fuel melting, coolant boiling, and vaporization and burn-out. In addition the program has also provisions for flow decay and decay heating. The code calculates power, inverse period and temperatures as a function of time.

In order to assess the safety of a reactor system one has to define a series of upset conditions. Typically these are deviations from normal conditions which are anticipated to occur often enough so that the system should include a capability to withstand the conditions without operational impairment. These conditions include transients from single operation error or equipment malfunction.
The table below gives a survey of the various conceivable reactor accidents which have been calculated.

Case	Description
1	Control rod withdrawal from full power and runaway of all rods.
2	Cold water incident giving a reactor period of 1,3 seconds.
3	A 5 $ / s reactivity insertion rate inflicted by experimental failure.
4	Start-up accident, control rod withdrawal from a cold clean core.
5	Loss of primary cooling flow to 7 °/o of full flow in 1 second.

To investigate the accidents it is further assumed that:

- the nominal reactor power is 55 MW
- the only shut-down mechanism is the safety level trip which is set to scram the reactor at 60 MW
- the mean delay time between the time that the scram- signal is sensed and the release of the control rods is 58 msec.
- a maximum load channel is defined subject to unfavourable nuclear and hydraulic conditions with the following characteristics :

Channel power	110 kW
Core pressure drop	0,72 bar
Coolant velocity	4,19 m/s
Type of flow	single phase

Table 8 gives a summary of the comparational results as obtained with the AIROS code.
From the results the conclusion can be drawn that, under the circumstances postulated, the fuel plate temperature remains well below the melting point of aluminium and the maximum outlet temperature of the cooling flow remains below the water boiling temperature. The results confirm the conclusion stated in the present safety report.

e) Pool Cooling System

Studies are in hand to investigate the uprating and modification of the pool cooling system with a view to providing a larger and more carefully directed flow which will inhibit the migration of ^{16}N to the pool surface and so reduce the activity level. It will also be necessary to demonstrate the compatibility of the new system with the beam tube and vessel wall cooling requirements.

Table 8 Results of computer simulated upset conditions.

Property		Upset condition no. 1	2	3	4	5
Core peak power	MW	60,4	62,9	92,0	188	55
Maximum fuel plate temperature	°C	164	160	188	258	158
Bulk flow temperature	°C	77	76	82	92	76
Outlet temperature	°C	96	95	104	114	101
Inlet temperature	°C	50	50	50	40	50

f) Vessel Replacement Scenario

A special working group has been installed to start the technical work involved in the dismantling of the existing reactor vessel and the installation of the new vessel. The group has recommended full scale tests for the cutting of the thermal column nose and the 8" through tube from the pool side facility to the bottom plug.

For the thermal column removal a special method will be tested which should result in the lowest possible radiation doses for the involved personnel. The method consists of the retraction of the CO_2 circulation tube from the "nose" of the thermal column, the cutting of the remaining two concentric cylinders and the fixation of the highly radioactive nose sections to the reactor vessel by a hardening foam. The nose section can thus be removed together with the reactor vessel and the dismantling of the remainder of the thermal column can be done under much more moderate radiation conditions.

Measurements with shielded and unshielded radiation sensors have been performed [12] to determine the magnitude of the radiation fields that can be expected during the reactor vessel replacement activities.

In Fig. 26 typical radiation fields between reactor core box and pool wall are shown along a normally used (i.e. gas-filled) beam tube (HB4) some 15 days after shut down and unloading of the reactor.

Along a beam tube which has been mostly unused (i.e. water-filled) radiation levels are considerably lower (HB8-bellows: < 1 R/hr).

As the highest radiation source, i.e. the steel bellows section, will be dismantled and removed prior to lowering the water level in the reactor pool, the major source of (radiation) concern will be the radioactivity induced in the outer beam tube section.

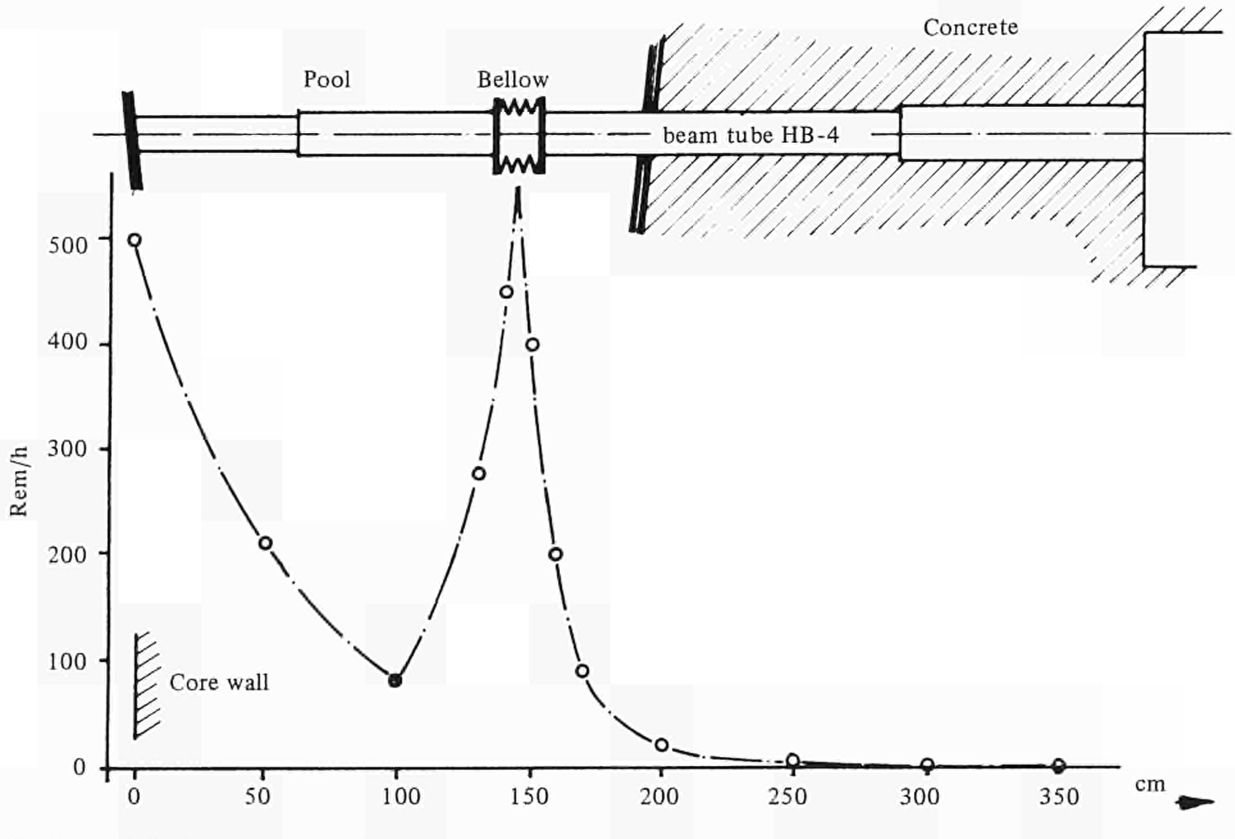


Fig. 26 Radiation level in HB-4.

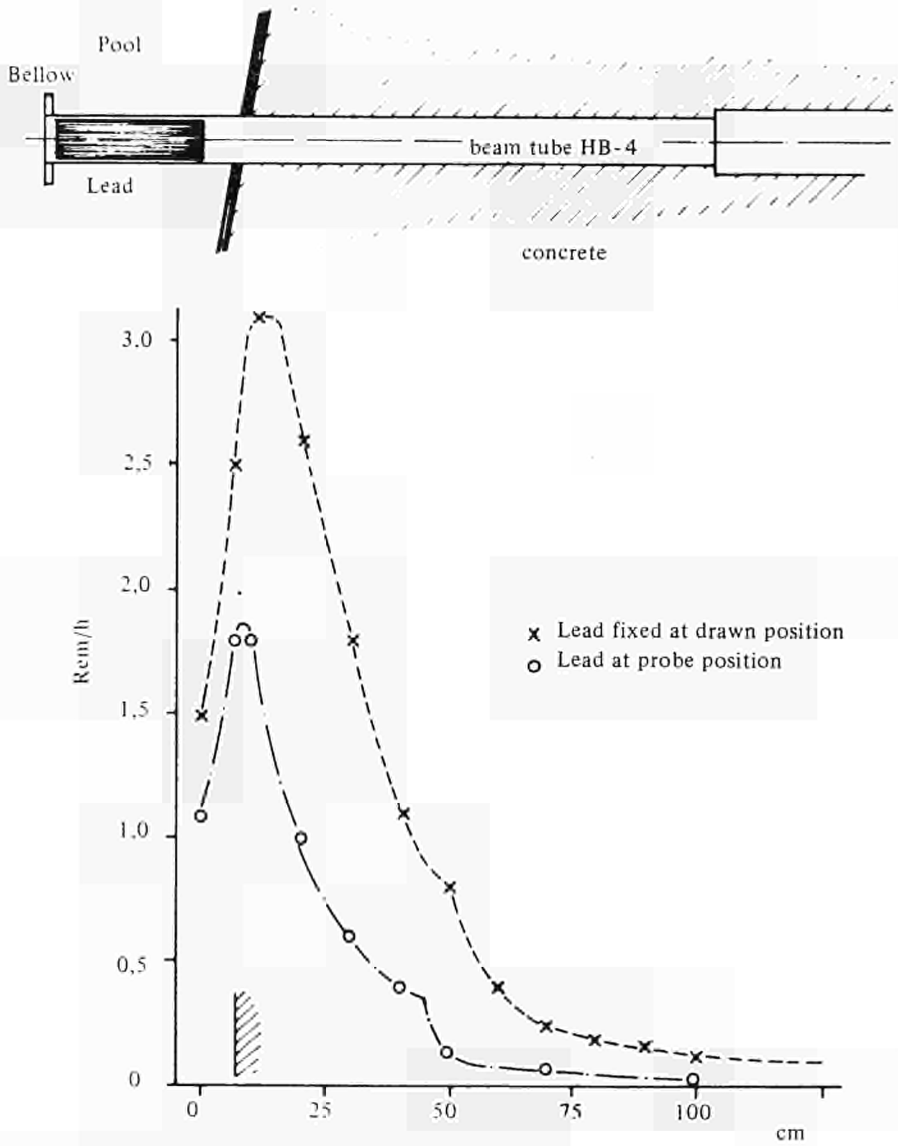


Fig. 27 Radiation levels in beam tube HB-4.

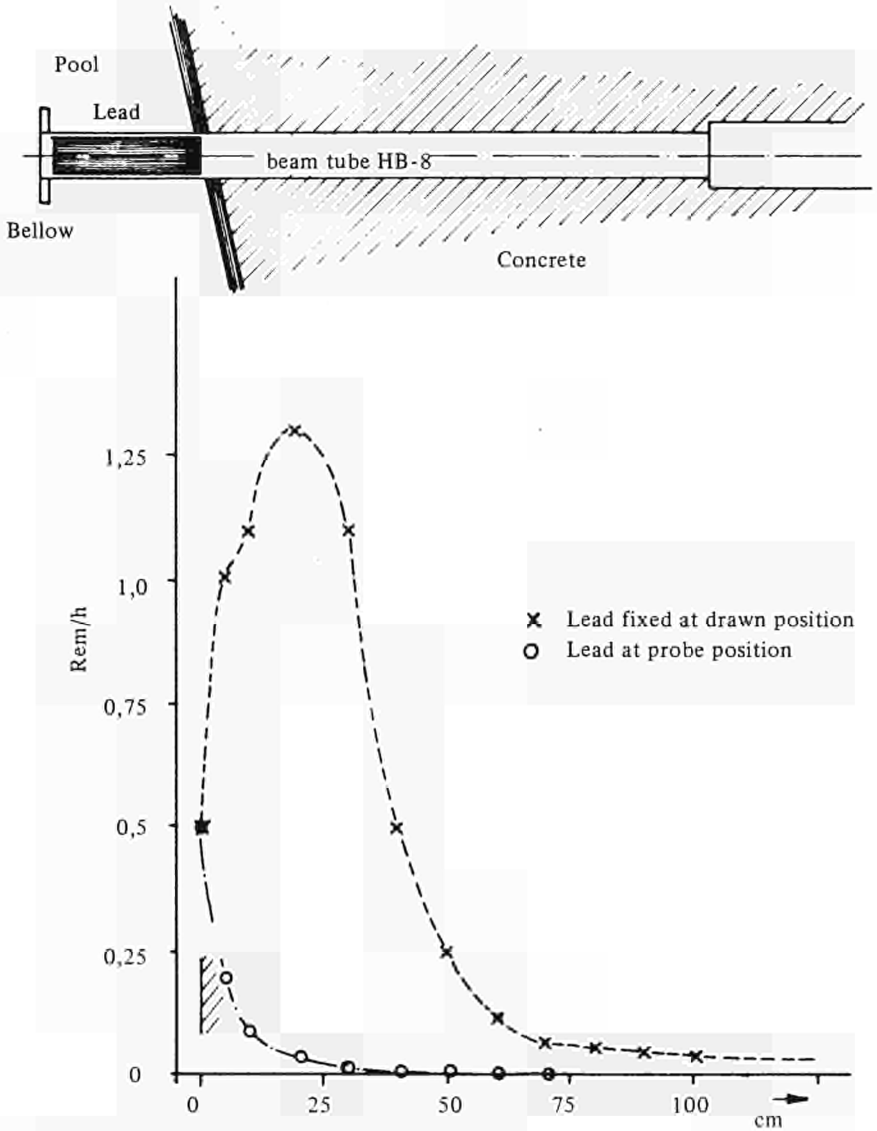


Fig. 28 Radiation levels in beam tube HB-8.

Figures 27 and 28 show typical distribution curves for radiation levels in a used and an unused beam tube with the bellows shielded off by means of a lead plug. Two cases are shown, i.e. with the shielding plug fixed at the outer end and the shielding plug moving inwards with the radiation sensor. It seems that the remaining radiation level is mainly caused by activation of the outer centimeters of the concrete wall. It is expected that lead slabs of a few centimeters thickness, applied around the beam tubes, will be adequate to reduce in-pool radiation to reasonable working levels.

g) Evaluation of an Alternative ("MK II") Vessel Design

As a follow up to the recommendation of the VAT II working group, a contract has been awarded to perform a feasibility study on the Mark II design. In particular the contractor has been asked to prepare general arrangement drawings showing details of the core location and hold down, and the method of control rod alignment. A comparison will be made of the relative merits of the three designs (present vessel, replacement vessels Mark I and Mark II) based on mechanical and thermal design considerations, ease of manufacture and inspectability. Any areas of uncertainty will be identified.

A comparison has been made between the thermal and fast flux characteristics of five HFR vessel options, i.e.

MK 0:	Present vessel and standard core
MK IA :	New design with second pool side facility in use
MK IB :	As above, without second pool side facility
MK II :	Cylindrical vessel with 3 "half" fuel elements between reactor core and vessel wall.
MK II :	Cylindrical vessel with standard fuel elements and excentric core to optimise the neutron fluxes in the PSF.

The study has been reported in [13], [14] and [15].

As a typical result of the study, thermal flux characteristics in the pool side facility are given in Fig. 29 for each of the five cases mentioned above.

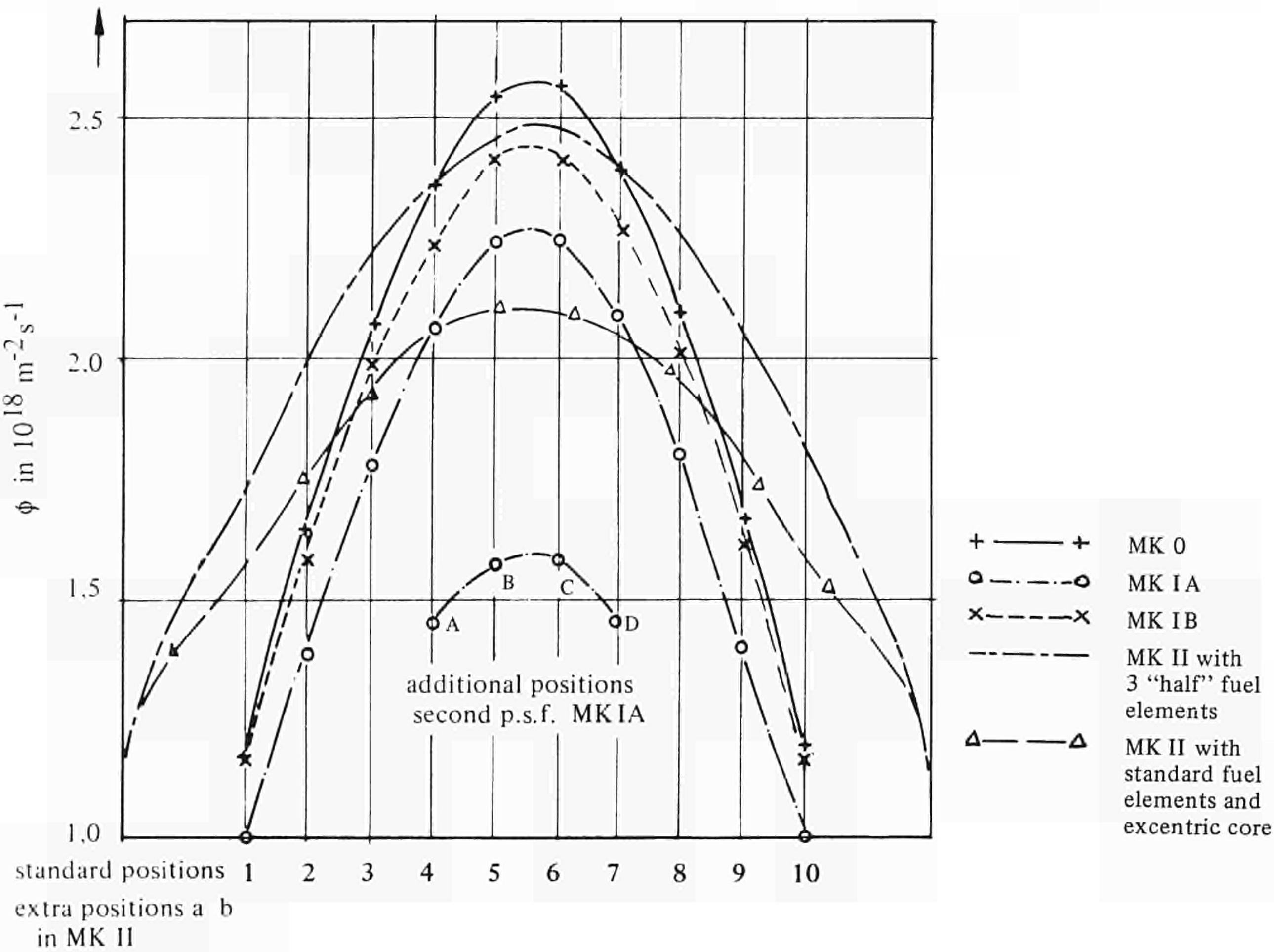


Fig. 29 Thermal flux distribution in the P S F of different vessels.

g) *Aluminium Irradiation Testing and Surveillance*

The ongoing HFR aluminium surveillance programme (experiment RX 92) has demonstrated that neutron irradiation influences the mechanical properties of the present vessel material. A gradual increase in tensile strength accompanied by a reduction in ductility has been demonstrated and though certain stabilization effects seem apparent, saturation values for these properties have not yet been reached. A property, which is not investigated by the RX 92 programme but which may be of considerable importance for the mechanical reliability of the corebox walls, is the so-called K_{IC} or fracture toughness factor. As any future surveillance programme will probably have to include the effect of irradiation upon this property and as little is known at present about the execution and optimum specimen choice for such investigations an irradiation test called "SURP", (RX 189.0) has been prepared which will address these questions.

The "SURP" irradiation facility consists of an aluminium holder which can be placed in a normal filler element of 72 mm diameter. In the middle of this holder the specimen carrier is placed, containing two columns of each six specimens. Cooling takes place by the normal primary cooling flow. The facility will not be instrumented. A draft design and safety report for this irradiation has been made. Material for machinery of the specimen is available. Manufacturing of the samples and execution of the irradiation will take place as soon as ongoing discussions about the sample geometry and configurations have led to a final result.

Published reports, coming mostly from Oak Ridge National Laboratory unequivocally show that the embrittlement and increase in strength of aluminium alloys are to a large extent caused by the transmutation produced silicon due to thermal neutrons. These strength and ductility changes are particularly severe in the solution hardened magnesium containing 5000-series aluminium alloys, because of the formation of Mg_2Si precipitates on a very fine scale. Reduction in ductility is likely to continue as long as dissolved magnesium is available for formation of Mg_2Si.

As stated above the RX 92 results support this view. Tensile tests on the latest batch of samples, irradiated to approximately $\phi_{th} = 5{,}1 \times 10^{22}$ n/cm^2 ($E < 0{,}025$ eV), show uniform and total ductilities of only 0,8 and 1,5 $^o/o$ at 50 ^oC and no tendency of saturation. At this thermal fluence level, which is typical for some parts of the HFR vessel at present, it is estimated that only about half the original Mg content has been consumed by Mg_2Si precipitates.

Microstructural investigation by means of electron-microscopy is in progress in an attempt to correlate quantitatively the mechanical property changes of 5154 to the size and density of the Mg_2Si precipitates.

Based on the literature data, a precipitation-hardened 6000-series aluminium alloy (e.g. 6061, 6063) would be preferable for construction of the replacement vessel.

h) *Vessel Manufacture*

In preparation for selecting a manufacturer, an informal enquiry was launched in October/November with more than 35 European companies. At present five companies have expressed an interest in receiving a formal call for tenders. Contract specifications are currently being prepared which will define the tasks and responsibilities during the manufacturing phase. The main functions envisaged are:

- fabrication (to include preparation of workshop drawings, supply of material, work test and delivery)
- inspection service (to include supervision, control and/or performance of all quality assurance and inspection requirements laid down by the ASME code and by the licensing authority).
- overall project management (to supervise and progress the total project and to provide an interface with the vessel designers).

Contracts for all three tasks will be placed in the first quarter of 1981.

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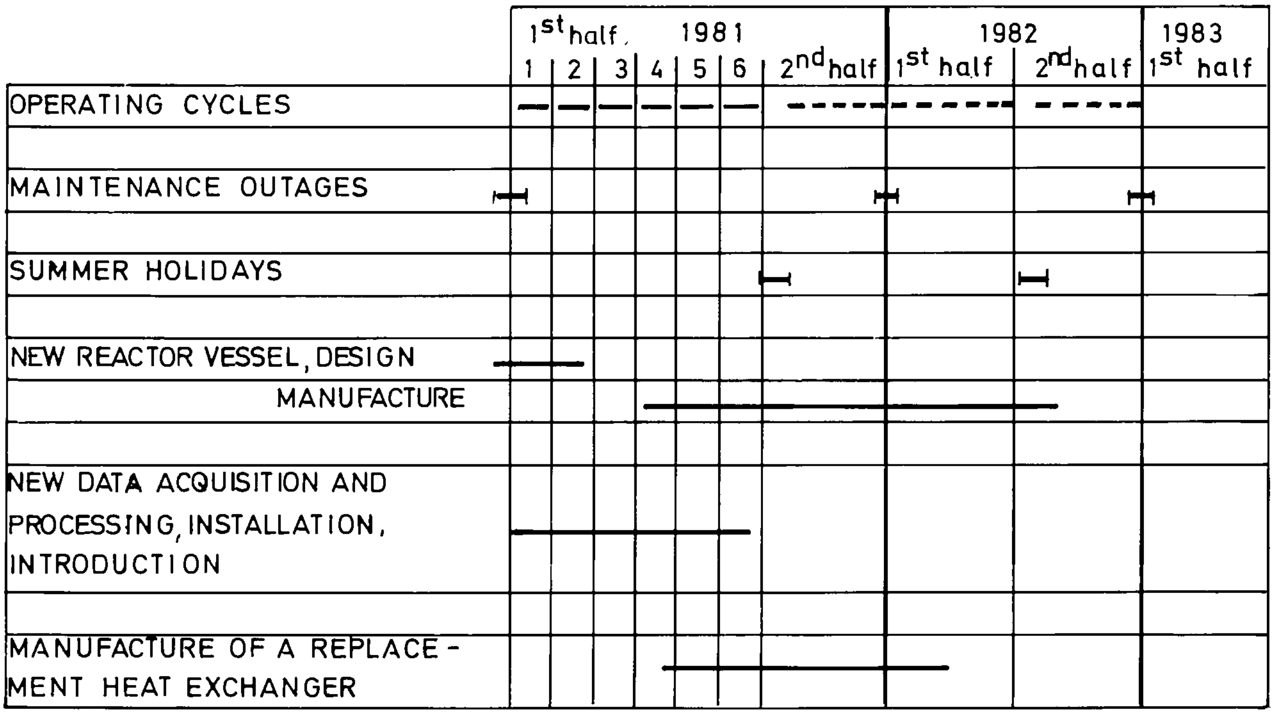
Fig. 30 Reactor Operation. Survey of future activities.



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 June 19, 1980 :

- start of new irradiations 20
- continued irradiation throughout the six months 7
- unloading after end of irradiation 10
- short term transient irradiations 14
- 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), preliminary planning
- Detail design and calculations, detailed planning
- Safety analysis and assessment
- Machining, purchase of material, instrumentation
- Assembly and testing
- Commissioning, loading and connection in HFR
- Irradiation, surveillance, data acquisition
- 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 x 10^{22} cm^{-2} (EDN*) for the most exposed samples.

D85 Series (Fundamental properties programme), see Table 9

D85-23 (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 until August 31, 1980.
With a total of 29 reactor cycles or 732 full power days, D85-23 was the longest lasting graphite irradiation ever performed in the HFR [1].
The maximum fast neutron fluence accumulated was 1,6 x 10^{22} cm^{-2} EDN.
In September, the sample holder was dismantled and the specimens recovered and sent to Jülich for intermediate examens. They will go under irradiation again as D85-38, early next year.
Additional investigations are being executed on the sample carrier itself (aluminium), which, thanks to the above mentioned high neutron fluence, forms an interesting object in the frame of fusion reactor material research.

D85-33 (400 °C):

continued the irradiation of the samples that have been irradiated in D85-23/400. The design was that of D85-25, i.e. an aluminium rod with holes containing about 130 graphite samples [2].

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

It performed well until August 31, when the irradiation ended after 13 cycles or 323 full power days.
During September it was dismantled and the samples sent to KFA Jülich for PIE. These samples will undergo further irradiation, from February 1981 on, as D85-39.

D85-34 (600 °C) :
was the continuation of D85-24/600. In design it was identical to D85-31/32; its irradiation history is similar to 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 continued their irradiation, since cycle 79-07 in a new sample carrier, identical to the previous one.
Together with D85-33, the irradiation ended after cycle 80-07 with a total of 12 cycles or 299 full power days. Similar to D85-23 and -33 it was dismantled during September and the samples were sent for PIE to Jülich [3].
They will continue their irradiation, early in 1981, as number D85-40.

D85-36 (750 °C) :
Its design is slightly changed, but in principle similar to type 33, i.e. D85-35.
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 $1{,}6 \times 10^{22}$ cm^{-2} EDN.
The obtained temperature accuracy was, during the first cycle (80-06), very good. However, during the following cycle an increasing deformation of the temperature profile showed up, which after some weeks, became stable within a margin of $\pm$ 35 °C.
The only possible explanation we have found for this phenomenon can be the fact that residual stresses in the stainless carrier caused an ovalisation of the carrier tube. This tube was, for the first time, not drilled from a solid rod, but a commercial, cold drawn, high precision tube with a ready made I.D. which had to be finished only at its O.D.
During the reporting period the experiment continued its irradiation with the exception of cycle 80-09, when it was interrupted because of another irradiation.

D85-37 (300 °C) :
started together with D85-36 in position C7 on June 6. It contains most of the samples irradiated in D85-26 which, after the scheduled 10 cycles, will have accumulated $\sim 1{,}9 \times 10^{22}$ 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.
The measured temperature distribution is satisfactory and steady.
Like D85-36, with which it shares a TRIO thimble, it had to be interrupted during cycle 80-09, for withdrawal of another experiment placed in the same TRIO.

D85-38/39/40 (300/400/600 °C) :
the follow-up irradiations of D85-23/33/34, have been postponed by three months and are now scheduled to start in cycle 81-02. The sample holders are of the types 14 (85-38 and -39) and 21 (85-40). Type 14 is a new version of the well tested type 11 and actually already under irradiation as D85-43, performing quite well. Type 21 is an updated issue of the previous type 21, irradiated several times quite successfully (D85-31, -32 and -34).
During the reporting period the three sample holders were manufactured and assembled and passed reception tests.

D85-41/42 (500 °C) :
are two identical new sample holders of the type 21 in an improved version. They were assembled in September and loaded with the samples irradiated earlier in D85-32 and -31, respectively.
After the start-up in cycle 80-10, on November 1, both turned out to be slightly too warm, due to some inaccuracy in the thermal calculations. During the major time of the cycle, the reactor had to be operated at reduced power (42 instead of 45 MW) in order to keep the temperatures of D85-41 and -42 within the admitted limits.
During the shut-down (1 week before the scheduled date) the TRIO with these experiments was placed from the "hottest" position (C5) to a somewhat "cooler" place (E5), where during the last cycle of the year, both performed well.
The irradiation is planned to last 11 cycles, i.e. until cycle 81-09. The fast neutron fluence will then be 1,4 and $1{,}6 \times 10^{22}$ cm^{-2} EDN including the previous irradiations.

D85-43 (400 °C)
is the first irradiation of the new low temperature standard sample holder type 14. It has, against the previous type 11, improvements on two features :
- the number of samples that can be irradiated has been increased by 12,5 °/o.
- it makes better use of the neutron flux, reaching $\pm$ 50 mm deeper into the reactor core.

D85-43 is the follow-up experiment of D85-25. It started, together with D85-41 and -42, in position C5 during cycle 80-10, and since cycle 80-11, in position E5, performing quite well in both. It is scheduled to last one year, until 81-09, accumulating 1,4 x 10^{22} cm^{-2} EDN, including D85-25 and, before that, D85-24.

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. About 3,5 x 10^{21} cm^{-2} EDN will be accumulated at the end of two irradiation steps.

D85-18 (700 °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 bits and pieces were manufactured and the sample holder assembled and tested.

The irradiation schedule has been changed and foresees now a start-up not before April, 1981.

D85-19 (900 °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 °C.

D85-20 (1100 °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 °C. It is an all-metal design, made from Nb and TZM. In the 1100 °C version it is equipped with K type thermocouples, sheathed with Nb. At higher temperatures, W/Re thermocouples will be used.

Similar to D85-18 and -19 it will be irradiated only from April, 1981 on.

Table 9 Fundamental properties graphite irradiation programme (D85). 1979/82 survey. [4]

Experiment no. D85-	Irradiation Period	Irradiation Temperature [°C]	Specimen carrier type
18	April - June 81	700	Type 331 (Graphite / S.S.)
19		900	Type 341 (Graphite / TZM)
20		1100	Type 351 (TZM/Nb)
23	Dec. 77 - Sept. 80	300 (400 - 500)	Al - (Al - Cu)
25	Jan. 79 - Apr. 80	400	Type 11 (all Al)
26	Aug. 78 - Dec. 79	300 - (750)	Type 12 (all Al) - (Cu)
31	Jan. 79 - Apr. 80	500	Type 21 (Graphite/S.S.)
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	Type 33 (Graphite /S.S.)
36	Jun. 80 - May 81	750	Type 36 (Graphite / S.S.)
37	Jun. 80 - May 81	300	Type 15 (all Al)
38	Febr. 81 - Febr. 82	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
44	Oct. 81 - Mar. 82	700	Type 331
45		900	Type 341
46		1100	Type 351

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 x $10^{22}cm^{-2}$, 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.

D186-01

Splitting up the original experiment target, a first irradiation will contain only compact tension samples, to be irradiated at 750 °C to a fluence of 6 x 10^{21} cm^{-2} EDN.
During the reporting period, conceptual studies and thermal calculation have been executed (Fig. 31). As the sample column is unusually long (615 mm), great uncertainties exist about the gamma flux level to be expected at the top end of the sample holder, an area of the reactor which until now has never been used for irradiation experiments. The necessity of a dummy irradiation to clear these questions is under discussion.

In-Pile Graphite Creep Studies

D156 (DISCREET)

The top and upper side reflector graphite of the process heat reactor experiences high neutron fluences and relatively low temperatures.
Irradiation creep studies on this material are being performed in tension at 300 °C and 500 °C 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.
The above experiments are all being performed on SIGRI ATR-2E graphite. An indication of the volume and consistent quality of the irradiations is supplied by Fig. 32, giving the free dimensional change of reference specimens from eleven different irradiations at 500 °C.
The scope of the experiments has not been widered to include the bottom reflector material SIGRI ASR-1RS which will be irradiated at 900 °C.

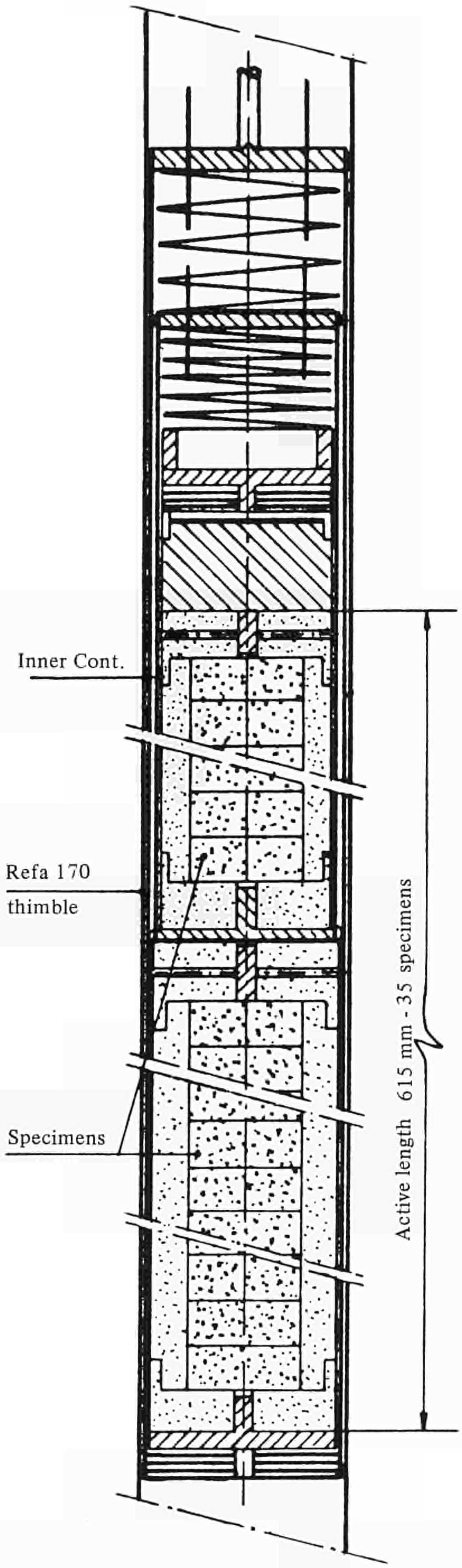


Fig. 31 Graphite irradiation D186. Design principle of the sample-holder.

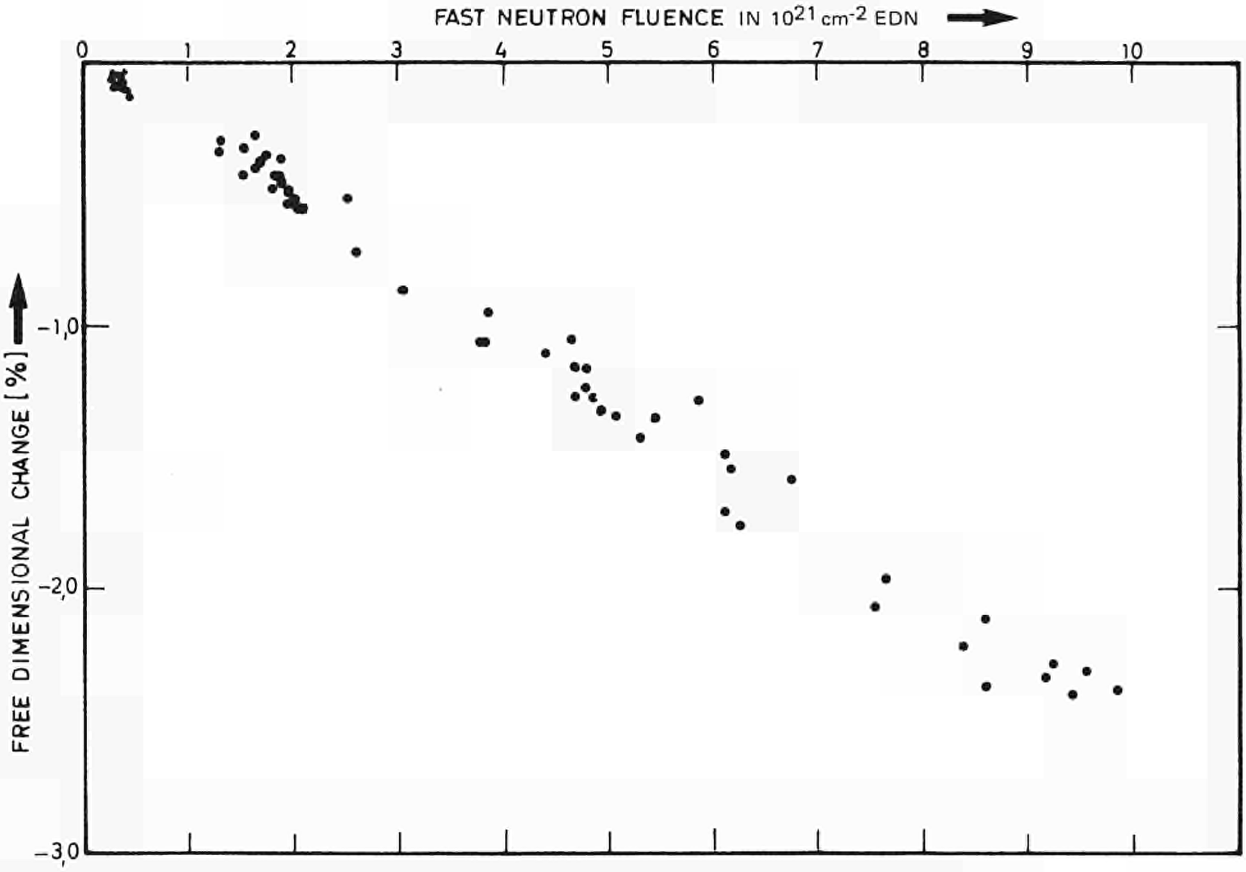


Fig. 32 Free dimensional changes of ATR-2E graphite at 500 °C from D156 experiments.

It is well known that creep strain has an influence on the irradiated properties of graphite, and investigations have begun into the possibility of performing Young's Modulus and coefficient of thermal expansion measurements in the Petten hot cells.

D156-00 Series High Flux 300 °C and 500 °C tensile, 500 °C compressive (ATR-2E graphite)

Irradiation of D156-05 began in cycle 80-06 and has continued successfully throughout the reporting period.

Most of the components for the follow-up experiment D156-06 have been fabricated, however the sample holder tubes were badly machined (oval and bent) and will be remade. No delay to the experiments is envisaged as irradiation is not planned until cycle 81-06.

D156-10 Series Long Term Creep Studies static and 500 °C tensile (ATR-2E graphite)

D156-11 (unstressed) has continued irradiation throughout the period. Earlier plans to stress the samples at some later stage have been reviewed, but irradiation will continue pending a future decision.

D156-13 started irradiation in cycle 80-07 but the experiment was withdrawn after two cycles because a large and incurable leak developed in the bellows system within the TRIO capsule. Dimensional measurements will not be taken and irradiation will be continued as soon as a replacement sample holder can be assembled.

D156-20 Series Low Flux Level, 300 °C, 500 °C, 500 °C tensile (ATR-2E graphite) [5]

D156-21 started irradiation in cycle 80-08 but was interrupted for cycle 80-09 due to a large leak in the TRIO thimble, which released gas into the reactor primary water cooling system.

The three sample holders were transferred to a new thimble and irradiation continued without incident for the remainder of the period. Components for the follow-up experiment D156-22 are currently being delivered.

D156-30 Series High Flux Level 3 x 900 °C tensile (ASR-1RS graphite)

The design of these sample holders will follow closely those of the current series of experiments, the only significant modifications being those made necessary by the higher irradiation temperature and by differences in nuclear heating rates. Minor changes have been made to the fixation of the lower load pin to provide additional security during transport. The in-pile section of the thermocouples will be niobium sheathed to reduce the rate of corrosion at the elevated temperature. A temperature map from the two dimensional heat transfer calculations [6] is shown in Fig. 33.

Components for the first three sample holders have been ordered and delivery is expected in the first half of 1981. Irradiation is planned for cycle 81-11.

D156-30 900C TENSILE CREEP RIG

1	0	0	0	0	0	0	0	0	0	0
2	0	538	545	551	551	548	540	191	55	50
3	0	472	453	437	400	324	285	143	54	50
4	0	450	432	416	379	305	269	135	53	50
5	0	464	464	464	411	314	271	134	53	50
6	0	460	451	414	354	292	262	133	53	50
7	0	459	449	413	353	291	261	132	53	50
8	0	460	460	460	407	312	270	134	53	50
9	0	443	426	411	375	302	267	134	53	50
10	0	439	422	408	373	304	271	142	54	50
11	0	457	411	370	367	362	232	219	56	50
12	0	497	482	434	376	301	232	221	55	50
13	0	531	532	526	510	474	342	212	56	50
14	0	571	570	570	569	568	546	311	54	50
15	0	619	618	618	618	617	596	373	56	50
16	0	708	707	707	706	705	677	439	58	50
17	0	811	811	811	810	809	700	467	60	50
18	0	928	928	928	928	927	865	527	62	50
19	0	953	953	953	952	949	891	552	63	50
20	0	940	940	940	939	936	873	573	63	50
21	0	922	921	921	911	909	868	583	64	50
22	0	916	916	916	907	905	862	557	66	50
23	0	914	913	913	905	903	857	578	67	50
24	0	910	909	909	907	905	858	582	67	50
25	0	910	910	910	909	904	859	585	68	50
26	0	913	912	912	911	907	861	588	68	50
27	0	913	912	911	910	907	860	591	68	50
28	0	917	916	916	907	905	858	600	69	50
29	0	916	915	915	906	904	855	579	70	50
30	0	914	914	913	904	902	852	576	70	50
31	0	908	907	907	905	902	851	576	70	50
32	0	906	906	905	904	899	850	577	70	50
33	0	905	904	904	903	898	846	588	70	50
34	0	904	904	903	901	898	845	589	71	50
35	0	909	909	909	899	897	846	590	71	50
36	0	911	910	910	900	898	846	590	71	50
37	0	910	910	910	900	898	847	591	71	50
38	0	905	904	904	902	899	846	590	71	50
39	0	905	904	904	903	898	847	590	71	50
40	0	906	906	906	905	900	849	586	70	50
41	0	907	906	906	904	901	849	586	70	50
42	0	914	914	914	905	903	852	586	70	50
43	0	917	917	917	908	906	859	596	68	50
44	0	918	917	917	909	907	860	595	68	50
45	0	912	912	911	910	907	860	593	68	50
46	0	912	911	911	910	906	860	592	68	50
47	0	913	912	912	911	907	866	593	65	50
48	0	912	911	911	909	907	864	590	65	50
49	0	912	912	912	904	902	863	586	64	50
50	0	911	911	911	903	901	860	579	64	50
51	0	913	913	913	904	902	859	572	63	50
52	0	924	923	922	920	915	862	560	63	50
53	0	951	951	951	951	951	887	531	62	50
54	0	966	965	965	964	963	879	474	61	50
55	0	967	967	966	966	965	876	432	71	50
56	0	968	968	968	967	966	879	478	61	50
57	0	969	969	969	968	967	883	508	61	50
58	0	137	135	133	127	115	89	63	55	50
59	0	64	64	63	63	61	58	56	55	50
60	0	0	0	0	0	0	0	0	0	0
	1	2	3	4	5	6	7	8	9	10

Fig. 33 Temperature map for D156-30.

Sample holder shielded transport container

Due to the high activity levels on the irradiated samples it became necessary to develop a shielded container [7] which would convey sample holders containing active graphite from the hot cells (G5, G6) to the TRIO thimble in the reactor pool, without intermediate handling. First tests on the container (with D85 sample holders) were successful and only minor modifications will be necessary for future transports.

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 x 10^{21} cm^{-2} EDN at 850 - 900 ^{o}C.

D166-02

Irradiation continued satisfactorily throughout the reporting period. The creep data to the end of cycle 80-10 are shown in Fig. 34. Operation of the computerised data logging and control system (ref. [8] and Fig. 35) has been more reliable and is now acceptable.

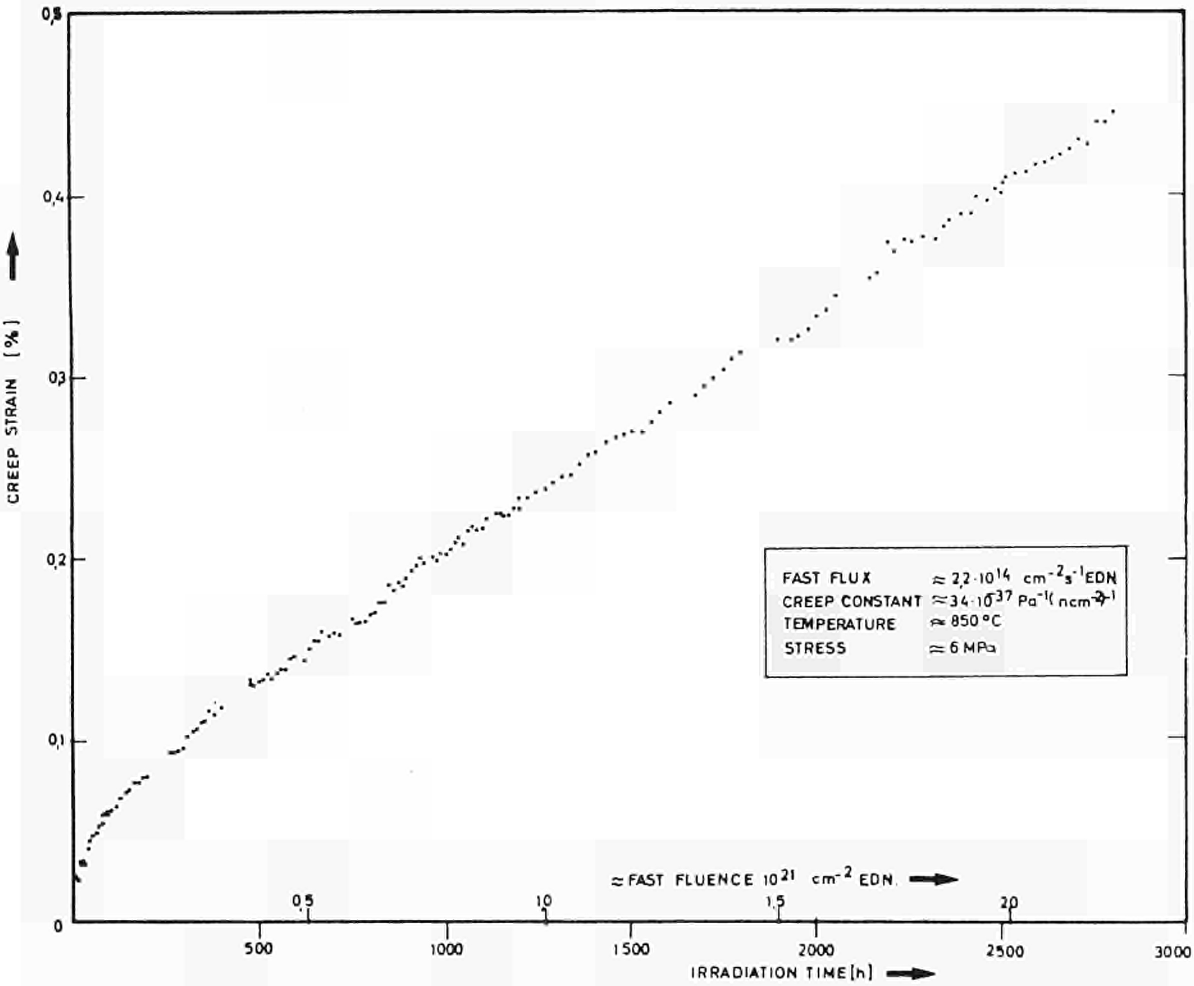


Fig. 34 Graphite creep experiment D166. Creep strain vs. irradiation time.

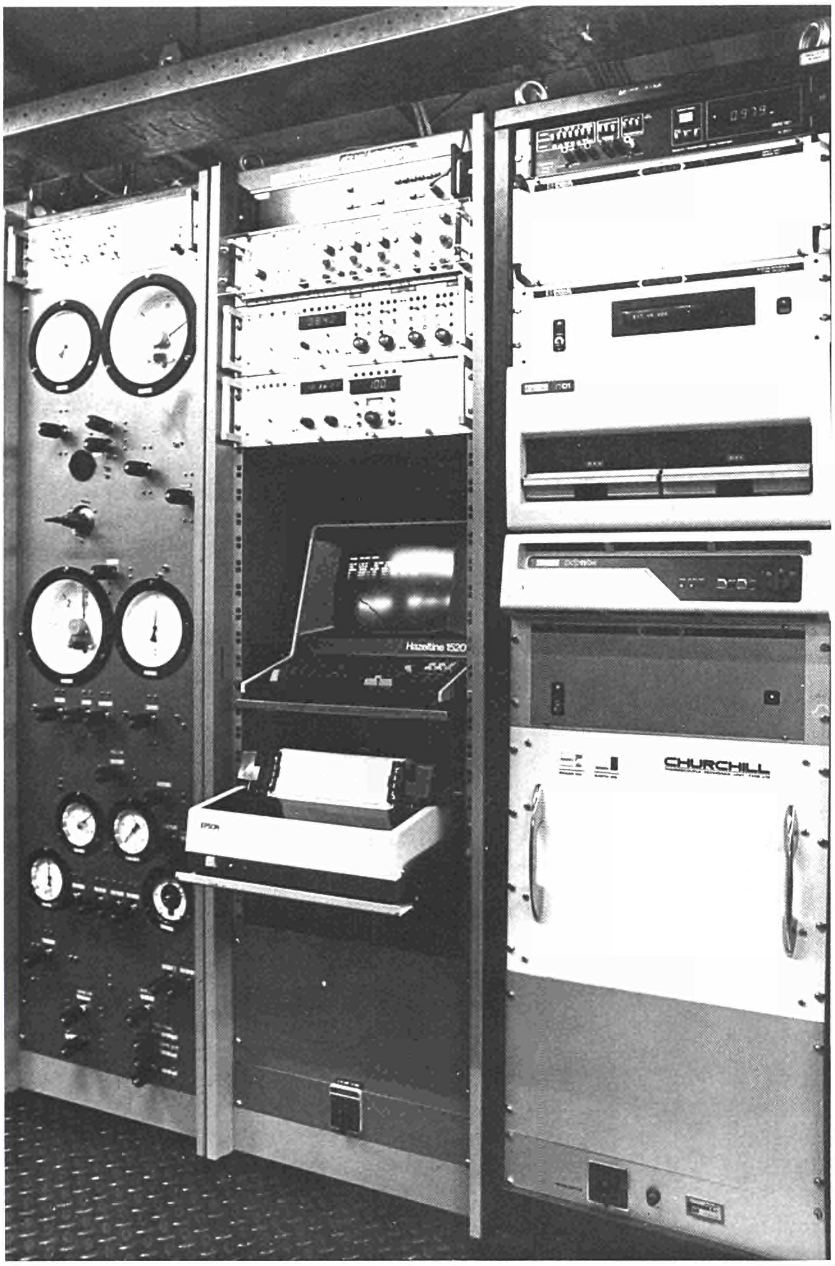
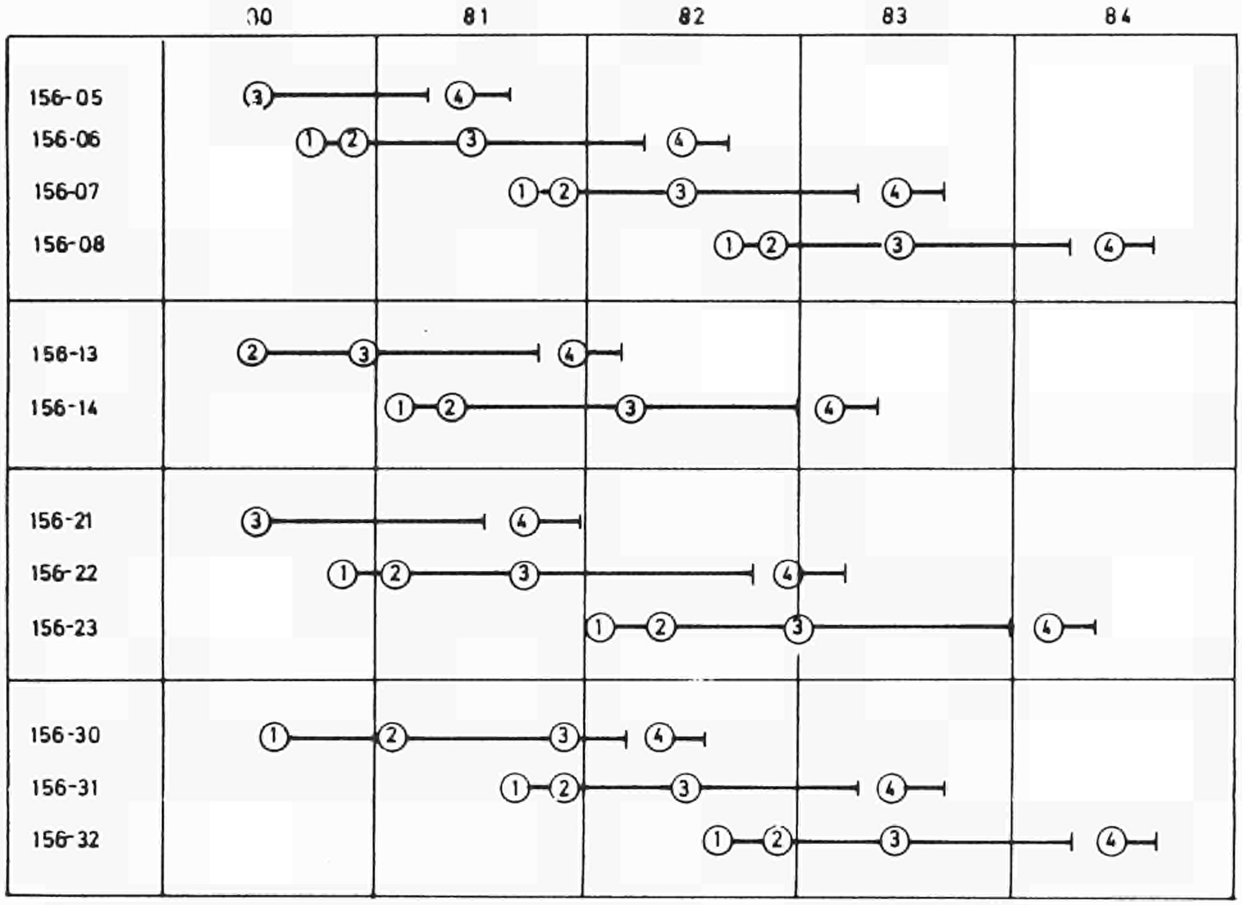


Fig. 35 Graphite creep experiment D166. Control console.

Fig. 36 D 156 GRAPHITE CREEP EXPERIMENTS SURVEY OF FUTURE ACTIVITIES



LEGEND

(1) DESIGN AND CALCULATION

(2) MANUFACTURE AND COMMISSIONING

(3) IRRADIATION, CYCLE REPORTING

(4) DISMANTLING, EVALUATION

CIRCLES INDICATE START OF ACTIVITY

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Progress Report P/F1/80/33 (Oct. 80)

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5. M.R. Cundy, G. Kleist, D. Mindermann, M.F. O'Connor
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Technical Memorandum IT/80/1847 (Dec., 1980)

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Computer Supervised Data Logging System for In-Pile Creep Experiments.
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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) which enable continuous fission gas analysis and temperature control to be carried out [5].

Coated Particles

D162 ARTEMIS, Irradiation of Coated Particle Fuel for Failure Mechanisms Investigation [6].

A second "gradient-experiment" will be carried out in 1981 for testing LEU coated particle fuel at abnormal thermal conditions. The design will be similar to the successcully performed first experiment. An irradiation proposal for KFA is currently under preparation.
A report on the irradiation facility has been presented to the 26th plenary meeting of the IDWG, this year [4].

D175 PETTICOAT, Irradiation of LEU Coated Particle Fuel at 1000 ^{o}C

Design and operating approval were given for the TRIO-131 irradiation facility by EAC (Experimental Assessment Committee) and RSC (Reactor Safety Committee). Specimens of standard quality coated particle fuel of LEU fuel cycle will not be available before the second half of 1981. Meanwhile, a mock-up experiment for neutron dosimetry and spectrum measurements will be performed in a TRIO irradiation facility. This experiment is expected to provide a reference data set on nuclear quantities and damage fluxes in azimuthal and vertical distribution for in-core 'graphite' experiments.

This mock-up consists of three all-graphite sample holders, spiked with dosimeters at three radii and over a length from - 300 mm to + 1500 mm with respect to centre line core. Design of the sample holders will start in January 1981. Sponsor of this experiment is KFA Jülich. The dosimeters and evaluation of these will be provided by ECN Petten.
Irradiation will be carried out during 2 hours at 100 kW in a central core position.

Special test irradiation E138-02 Prototype rig

A special in-pile rig has been designed and manufactured for the purpose of fully realistic operational testing of the new sweep loops (see paragr. 2.3.3.9). A "fuel ball' rig design has been selected using standard THTR elements as fissile targets.
During the reporting period the irradiation facility has been assembled [2], commissioned and connected as scheduled to the sweep loops. Design and safety report has been issued [1]. The design and operating approval has been given. "Hot" operation will start with cycle 81-01 [3].
Irradiation temperature will be measured in this experiment by thermocouples positioned either onto the fuel elements surface of into bores in the fuel free zone of one fuel element in each capsule. This is shown in Fig. 37, a radiograph of one assembled capsule.
The scope of this measure is to obtain consistent reference results on temperature readings of thermocouples positioned onto the surface of the fuel element. Similar experiments in the past showed that after high neutron doses the differential shrinkage of the spherical specimen and the graphite structure created additional gas gaps which could have influence on the contact between thermocouple and fuel element.

Fuel Elements

Irradiation of spherical HTR fuel elements for Nuclear Process Heat Reactor Development

D138-3 A third spherical fuel element irradiation experiment will be sponsored by KFA.
A proposal is currently being prepared for a three capsule rig design. Irradiation temperature will be about 1000 ^{o}C. Burn-up neutron fluence correlation of the LEU fuel containing elements will be similar to these specimens of the D175 coated particle irradiation experiment. Start of the irradiation is planned for 1981.

A 138-mock-up sample holder, sponsored by KFA Jülich, for extended neutron dosimetry measurements will be irradiated during 200 h at about 100 kW in a Refa 170 irradiation facility. The dosimeters will be provided and evaluated by ECN Petten. Results should give a reference mapping of damage fluxes for a full size HTR fuel element (≈ 60 mm Ø), irradiated in a reflector core position. The design is planned to start early in 1981.

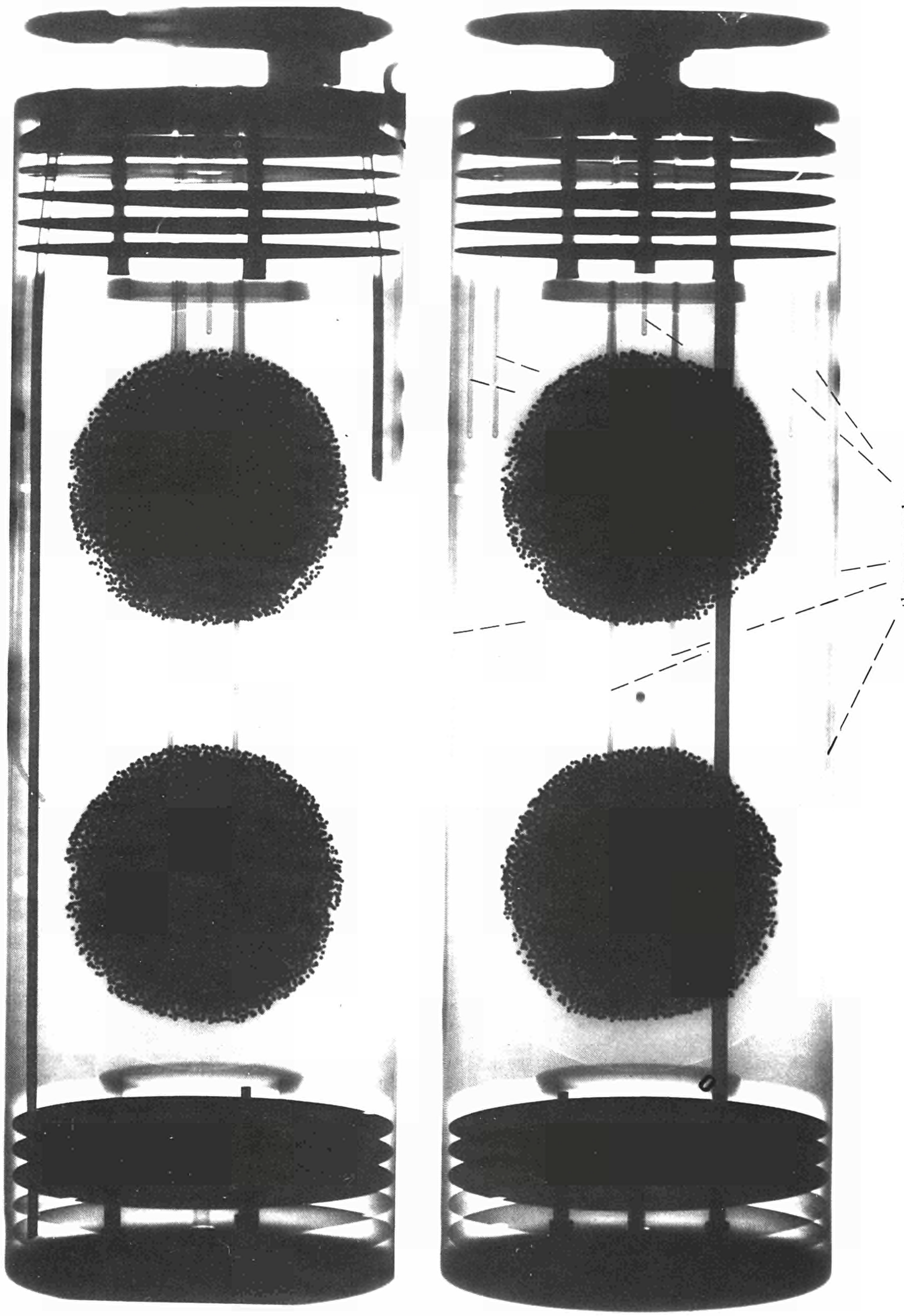
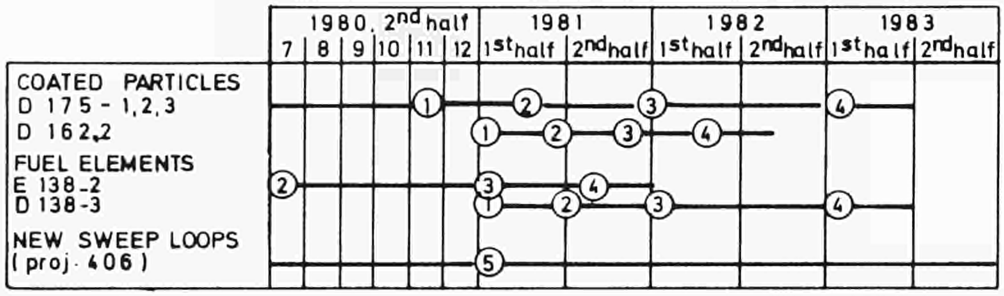


Fig. 37 Project E138.02. Two radiographs of capsule A (one turned by 90° compared to the other) after assembly.

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

References to paragraph 2.2.3.2

1. R. Conrad
Irradiation of low enriched spherical HTR fuel elements in the HFR Petten. Design and Safety Report
Technical Note P/F1/80/36 (Nov. 1980)

2. M. Beers
Montagebericht E138-02.
Technical Memorandum IT/80/1837 (Nov. 1980)

3. J. De Bueger, R. Conrad
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4. R. Conrad
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5. D.W. Klage, R. Conrad
Application of a Programmable Logic Controller (PLC) for a HTR fuel irradiation facility in the HFR Petten.
Paper presented at 26th plenary meeting of the "Irradiation Devices Working Group" at GKSS Research Centre, Geesthacht, F.R. Germanry, 8 - 10 Oct. 1980.

6. R. Conrad, C. Merlini, A.W. Mehner
Safety aspects of advanced coated particle fuels for High Temperature Reactor concepts.
Paper submitted for presentation at the "Topical Meeting on Reactor Safety Aspects of Fuel Behaviour", Sun Valley, Idaho, USA, 2 - 6 Aug. 1981.

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 reactor safety programmes 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", "AUSTIN", "FANTASIA", "SINAS") feature sodium-filled specimen carriers with a large number of thermocouples, and operating at 550 $^{\circ}$C or 650 $^{\circ}$C.
More recently, developments have commenced towards dynamic in-pile facility (creep, fatigue, crack propagation).

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 specimen.
Reactor loading of NAST-14 is now planned for February 1981.

R143 "MONA", ECN project 1.425

R143-5 Manufacturing, assembly and testing of the MONA-5 capsule has been completed and

PROJECT	SPECIMEN	RIG	DRAWING	POS	$\Phi_{0,1}$ 10^{24} m^{-2}
KFK	Φ9	TK5	32461 C	E7	15
		TK6	"	F2	"
		TK7	"	D2	"
		TK8	"	F8	"
		TK9	"	F2	"
		TK10	"	F8	"
		TK11	"	D2	"
1422	Φ8	25	33223 A	E7	90
	10	26	40424	F8	20
	Φ8		33223 B		
	Φ9		32212 C		
1424	60	C47	32745 B	H2	4
	60	C50	40734	H2	4
		C51	40731	H2	4
		C52	"	H2	4
	Φ9	T49	34533	E7	7
	28	C411	39158	H2	4
		C412	"	"	"
		C413	"	"	"
IFO					
1425	Φ8	N14		H8	
R 120	Φ9	N15		"	
R 143		M4		"	
		M5		"	50 } 10^{-4}
		M6		"	5

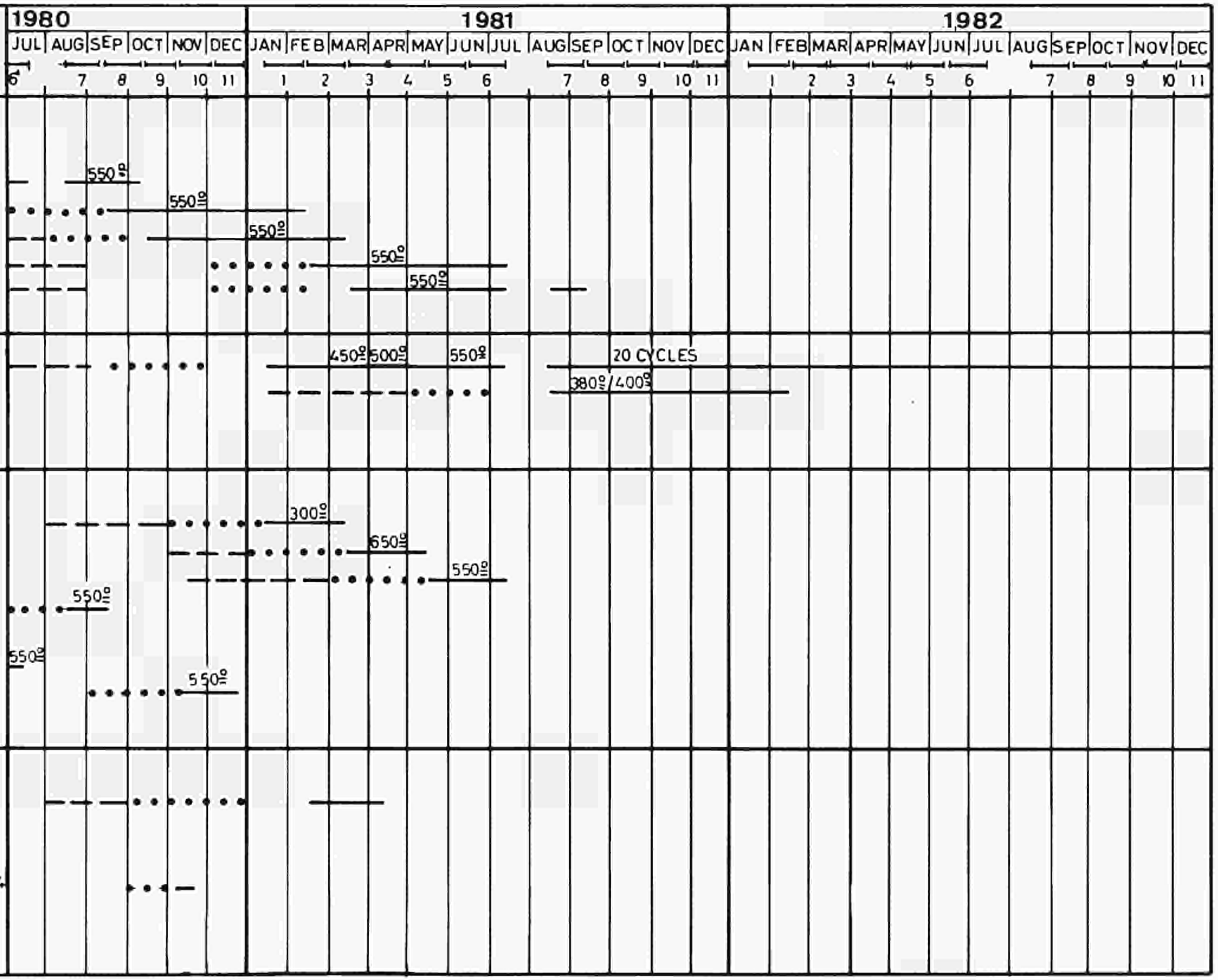


Fig. 39 Steel projects R120, R139, R143. Time schedule, updated 1.7.1980.

LEGEND:
T TRIO
C REFA
N NAST
M MONA

MACHINING — — —
ASSEMBLY • • • • • •
IRRADIATION ———

irradiation has been carried out in core position H8 as scheduled on November 11, 1980. The achieved characteristics were:

- neutron fluence ~ 1×10^{21} m^{-2}
- irradiation time 1000 sec. at 42,5 MW
- sample temperature 550 °C.

Evaluation of the thermal behaviour of the capsule shows that seven out of nine specimen in the rig have been irradiated within a ± 35 °C range.

The irradiation report of MONA-5 has been prepared (ref. [1]).

R139 TRIO and REFA capsule irradiations "SINAS"

These experiments belong to a large mechanical property 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. 39).

The following activities have been pursued during the reporting period :

- Irradiation reports have been completed and issued on the terminated experiments TK4, -5, -6 and -7, and on T49, C47, C411 and C412.
- the TK series' second extension has been successfully started with TK8 and -9 in September and October 1980 respectively.
- T25 is now ready for irradiation in January 1981 and T26 is scheduled in July 1981.
- T48 and C413 have been successfully irradiated in the second half of 1980 and C50, -51 and -52 are being completed for irradiation in the first half of 1981.
- a systematical review of experimental parameters of all R139 irradiations gives examples of the high degree of temperature efficiency and reliability of the experimental equipment (see Figs. 40 through 45).

Irradiation reports of the capsules R139-47, R139-411, R139-49 and R139-412 have been prepared and edited (refs. [2], [3], [4] and [5]).

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 Safety Programme.

The irradiation started in October 1978 and has since then 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.

At the end of cycle 80-11 the experiments reached a maximum damage dose of about 11 d.p.a.

The compatible software for the processing of data from punch tape was purchased. Data processing will be carried out, from cycle 80-09, using the desk top computer HP9845B.

E167 TRIESTE, Steel Creep Rig

This is a series of irradiation experiments for the JRC Ispra Fusion Reactor Materials Programme.

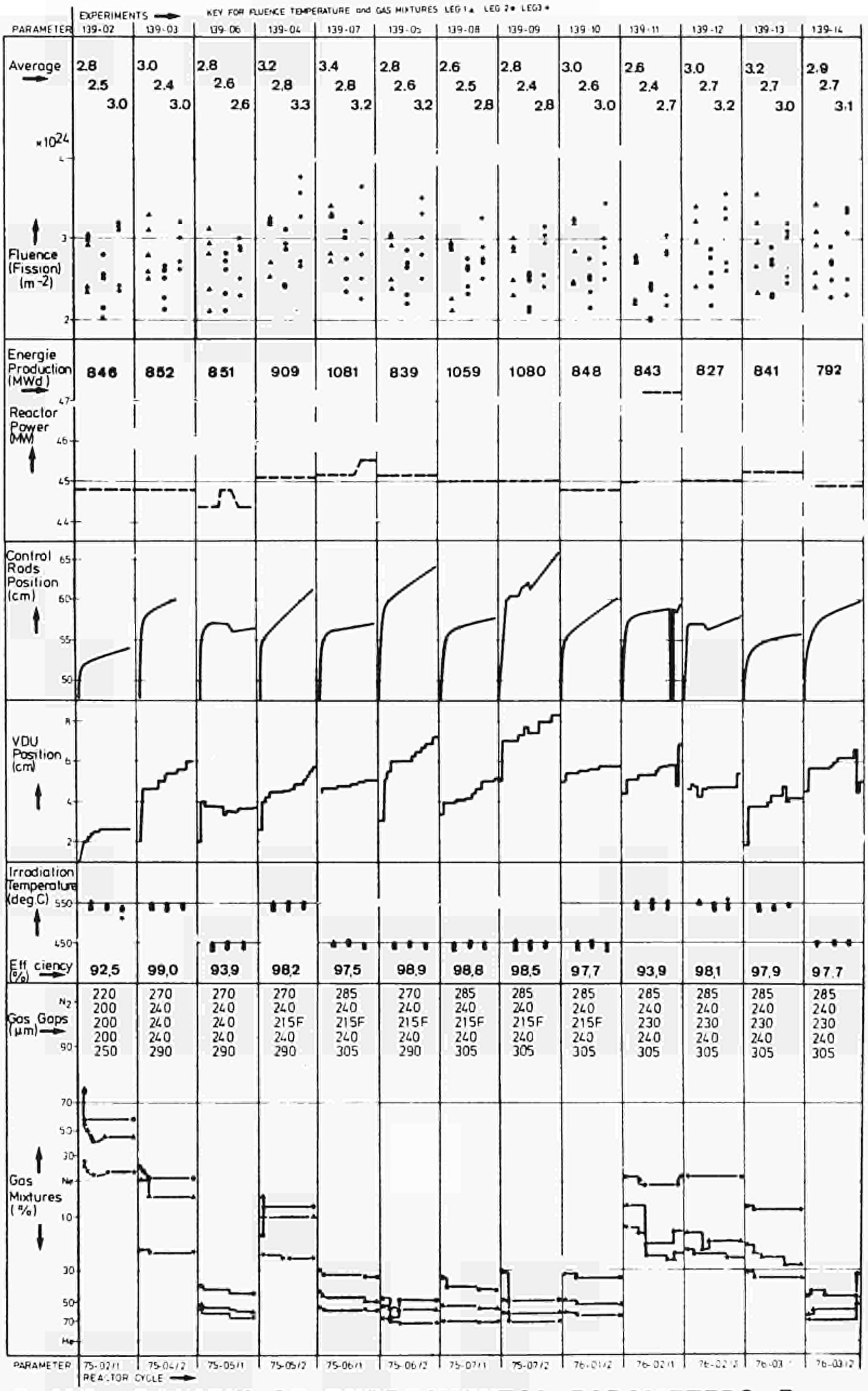
The handling and load-unload procedures of the irradiated specimen stems had to be further examined in view of HFR hot cell working conditions and cell equipment available. Fabrication of dummy sample columns and ancillary equipment had to be delayed because further studies on dimensional measuring techniques by means of neutron radiography became necessary. Detailed experiment design will be continued during the next period.

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

To evaluate the neutron enhanced degradation of fracture toughness characteristics in austenitic stainless steel materials for the JRC Ispra Reactor Safety Programme, the irradiation experiment E177-FANTASIA has been designed, in which more than 250 samples (tensile and 3PB specimens) are irradiated at 350 °C and 550 °C.

Five fluence steps between 10^{19} and 10^{21} $n.cm^{-2}$ are envisaged. The first six sample holders of this experiment E177/1 - 6 have been dismantled, and the 3PB and tensile samples are prepared for transport to the Applied Mechanics Division of JRC Ispra.

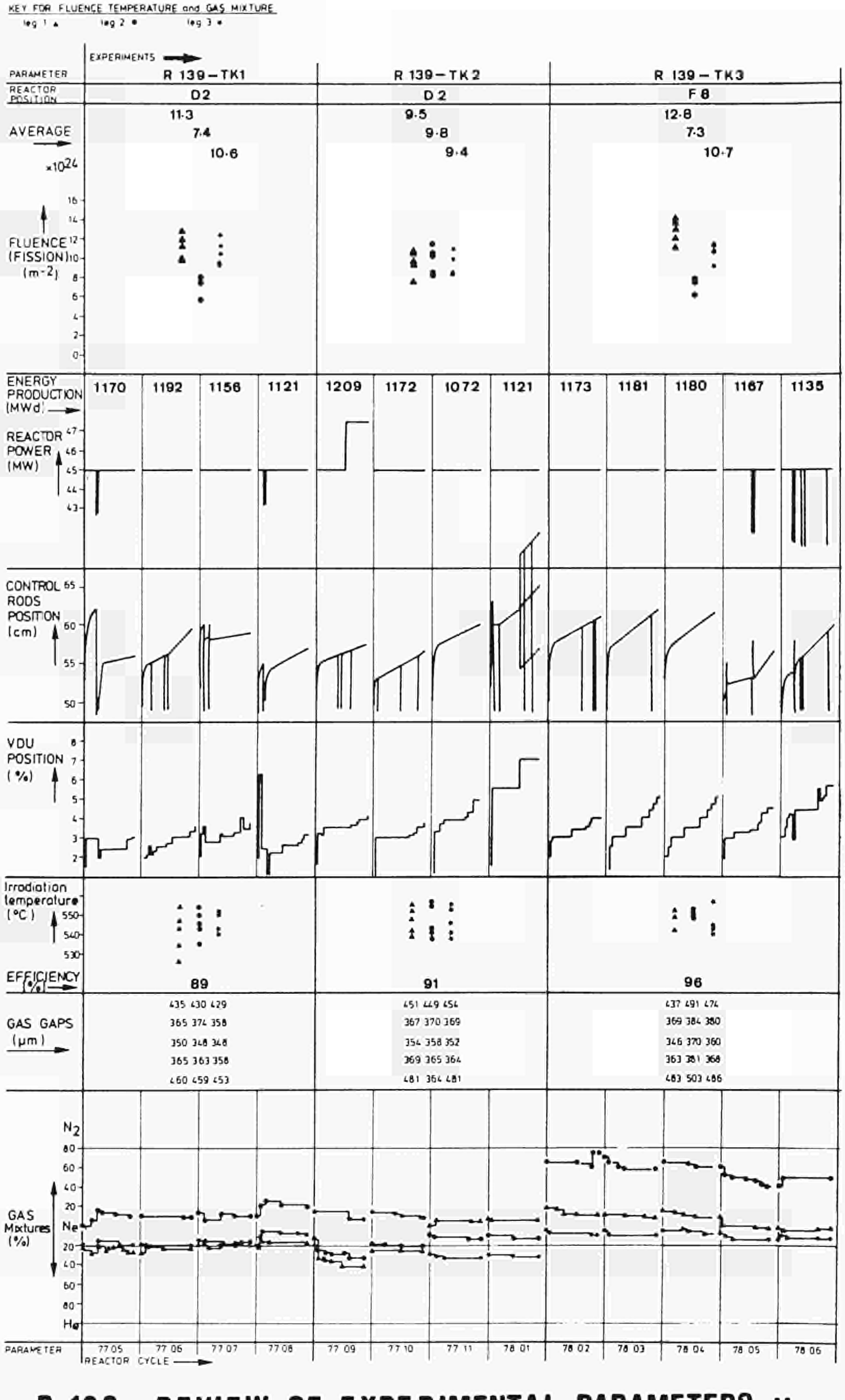
Irradiation was continued as scheduled in the HFR cycle 80-07 with the three sample holders E177/7 - 9 in HFR position H8 and the three sample holders E177/10 - 12 in position H2. Both irradiations were carried out for three cycles to reach a target fluence in the order of $4{,}5 \times 10^{20}$ $n.cm^{-2}$. Irradiation temperature of the steel samples was kept at 350 °C for the sample holders E177/ 7 - 9, respectively 550 °C for the next irradiation step. After irradiation the six sample holders E177/ 7 - 12 were prepared for dismantling. The following sample holders E177/13 - 24 for the next four irradiation steps are presently being manufactured. Irradiation is scheduled for HFR cycles 81-07 till 82-05.



139-02	139-03	139-06	139-04	139-07	139-05	139-08	139-09	139-10	139-11	139-12	139-13	139-14
220	270	270	270	285	270	285	285	285	285	285	285	285
200	240	240	240	240	240	240	240	240	240	240	240	240
200	240	240	215F	215F	215F	215F	215F	215F	230	230	230	230
200	240	240	240	240	240	240	240	240	240	240	240	240
250	290	290	290	305	290	305	305	305	305	305	305	305

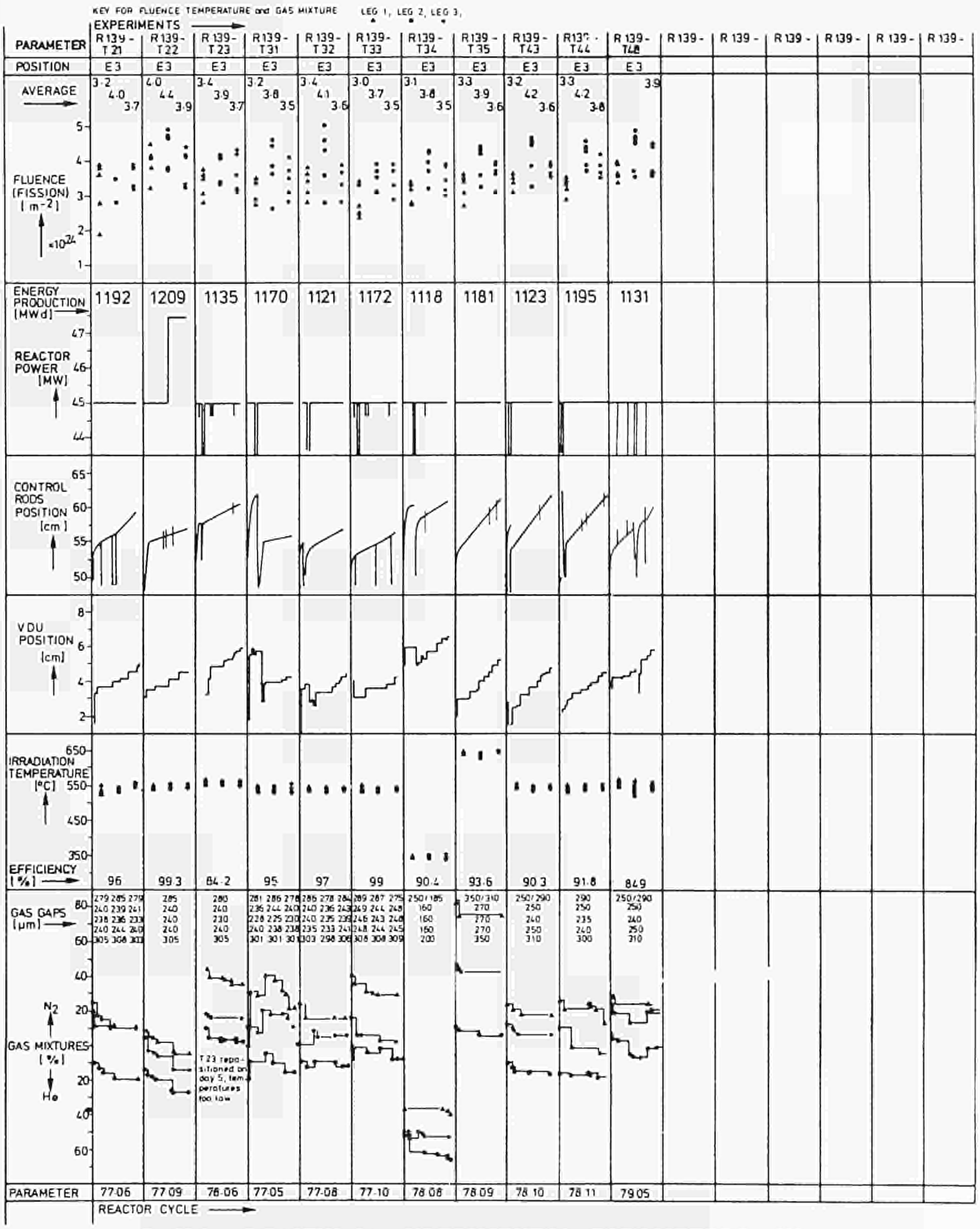
R 139 - REVIEW OF EXPERIMENTAL PARAMETERS I

Fig. 40



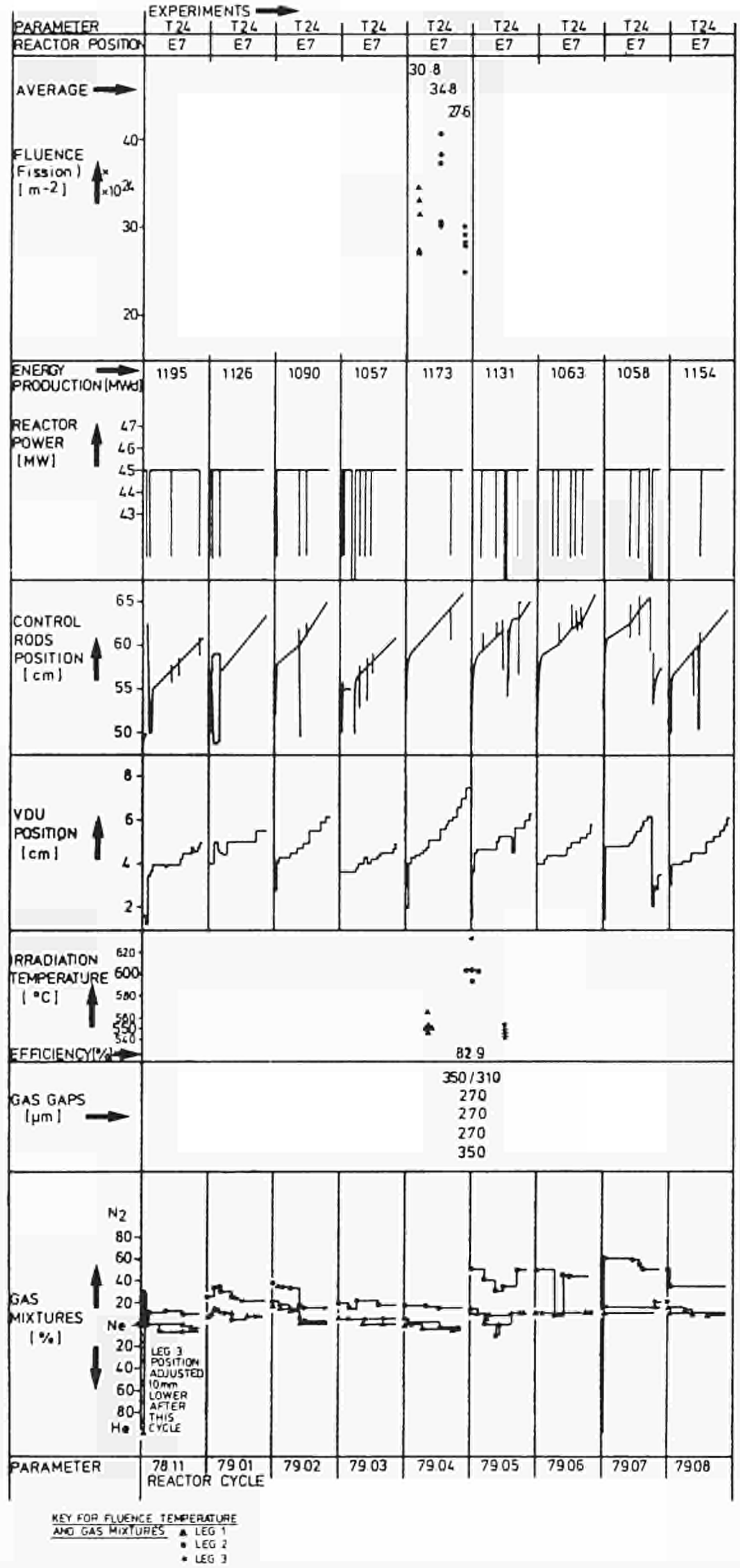
R 139 - REVIEW OF EXPERIMENTAL PARAMETERS II

Fig. 41



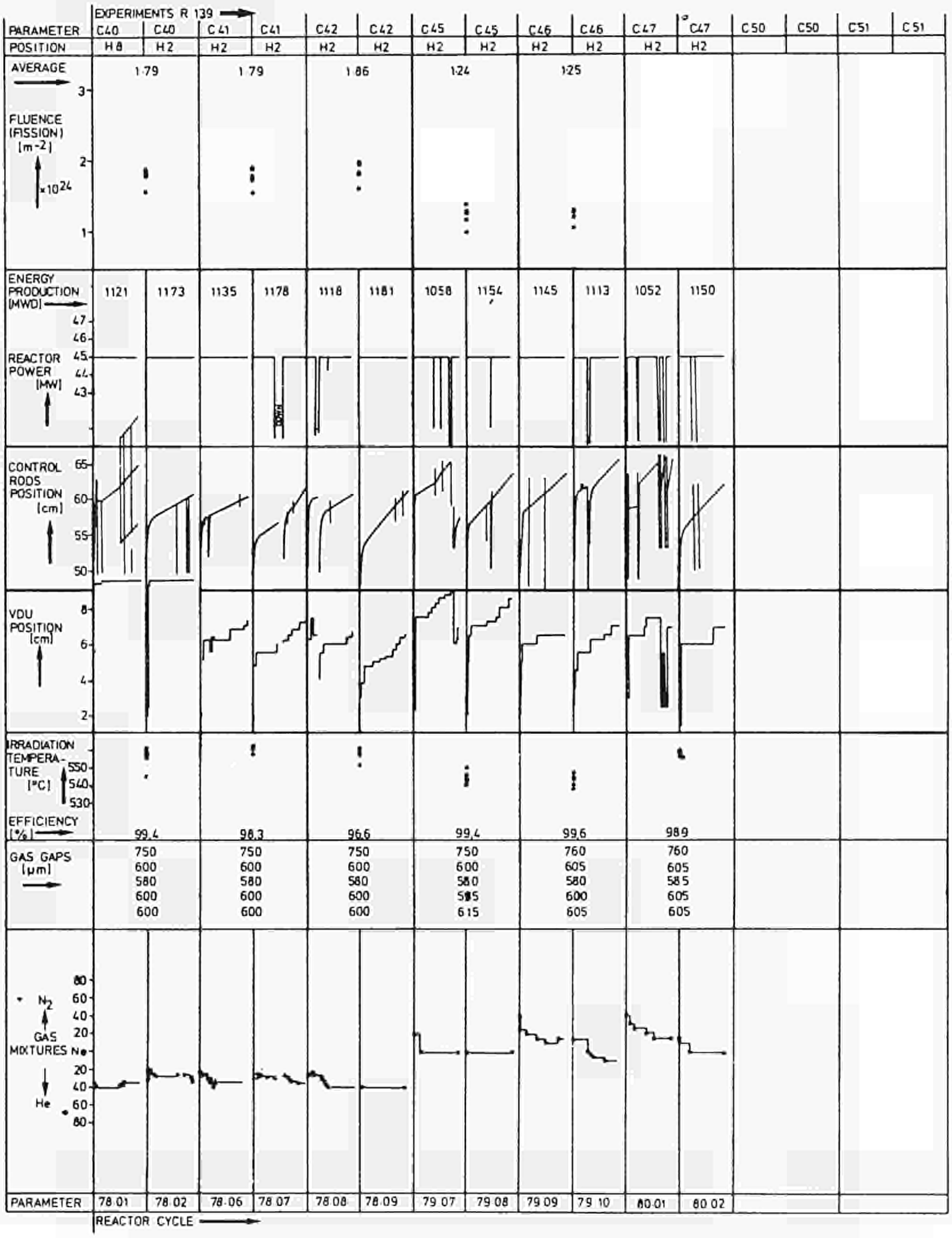
R 139 - REVIEW OF EXPERIMENTAL PARAMETERS III

Fig. 42



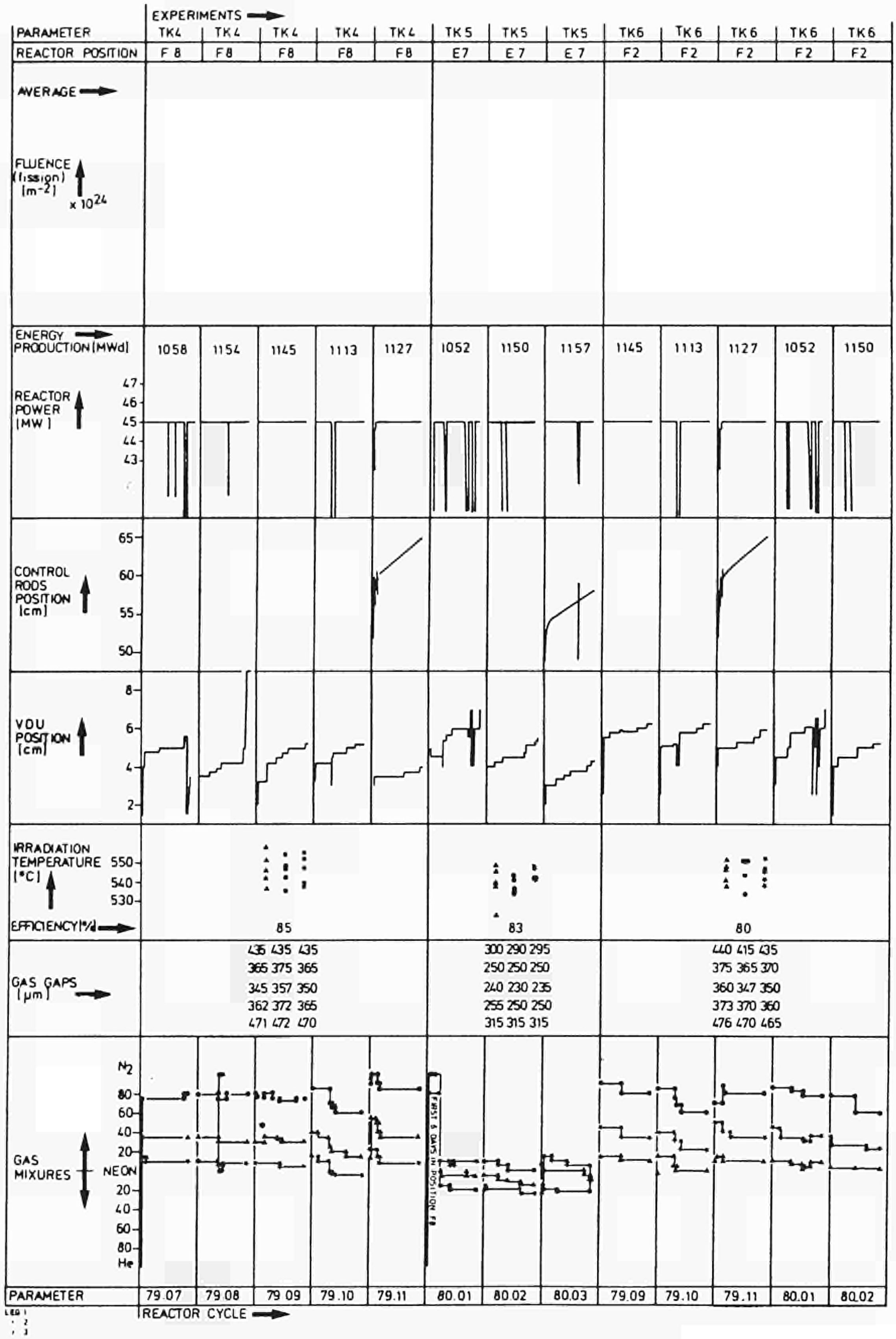
R 139 REVIEW OF EXPERIMENTAL PARAMETERS IV

Fig. 43



R 139 - REVIEW OF EXPERIMENTAL PARAMETERS V

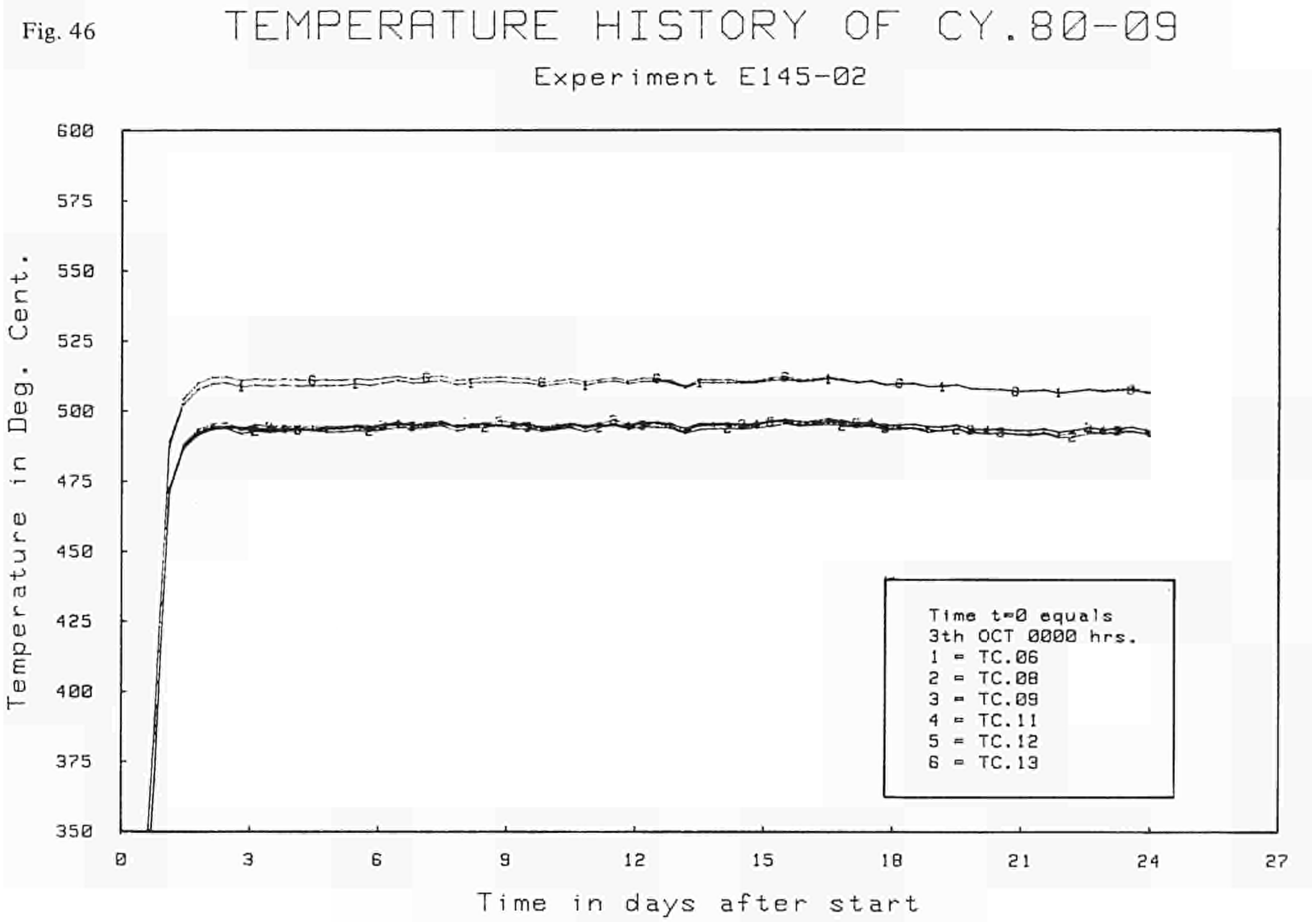
Fig. 44



R139-REVIEW OF EXPERIMENTAL PARAMETERS VI

Fig. 45

Fig. 46



D190 STORY. Short Time Irradiations of Pressure Vessel Steel Samples

Within the surveillance programme for LWR pressure vessel materials a number of high flux, low temperature, short time irradiations have to be carried out.

A special reactor run is envisaged for the experimentation which allows rather simple sample holder arrangements. In this run six sample carriers will be irradiated simultaneously in different in-core positions up to six fluence levels so that the complete irradiation programme can be carried out within four days full power reactor operation. Since the un-instrumented sample containers are directly kept by the core filler element construction, load-unload procedures will be performed at reduced HFR power.

The sample holders carrying six bar-shaped sharpy and six dumbbell-shaped tensile samples each are presently being designed.

b) Other Structural Materials

R158 "HOBBIE", Zircaloy Creep Experiment, ECN Project 1085

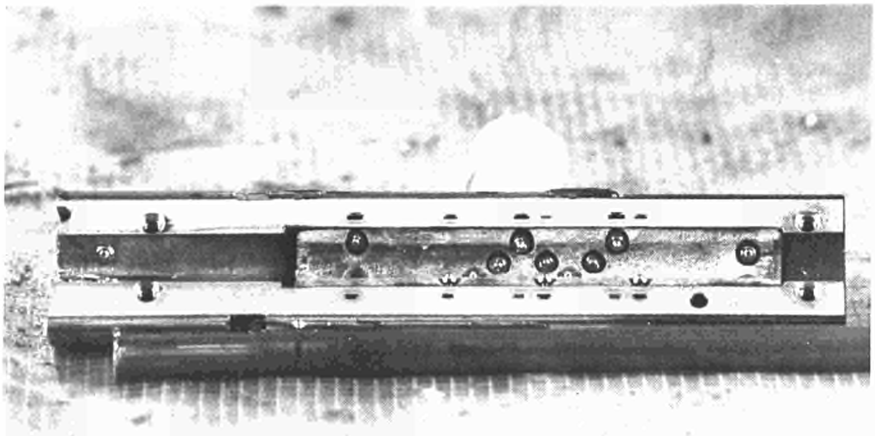
In spite of considerable technical difficulties due to a leak in one of the helium circuits (see previous progress report) the experimental objectives of the irradiations could be achieved and good creep data were obtained. The facility was unloaded after HFR cycle 80-06, and dismantled (Fig. 47).

The irradiation report has been completed, [6], one neutron metrology report remaining as the only project activity in 1981.

The end of 1980 signified the end of the contract period during which the US-NRC, technically represented by ORNL, and ECN have worked together in performing the Zircaloy cladding creep-collapse programme.

An overall report, concerning the technical and scientific aspects of the entire irradiation series, has been drafted in cooperative effort between Oak Ridge and ECN collaborators and will be published in 1981.

The eddy-current deformation sensors and associated instrumentation will remain available in Petten for other irradiation technology applications. A presentation of the HOBBIE technology and results at the Geesthacht meeting of the Irradiation Devices Working Group met considerable interest.



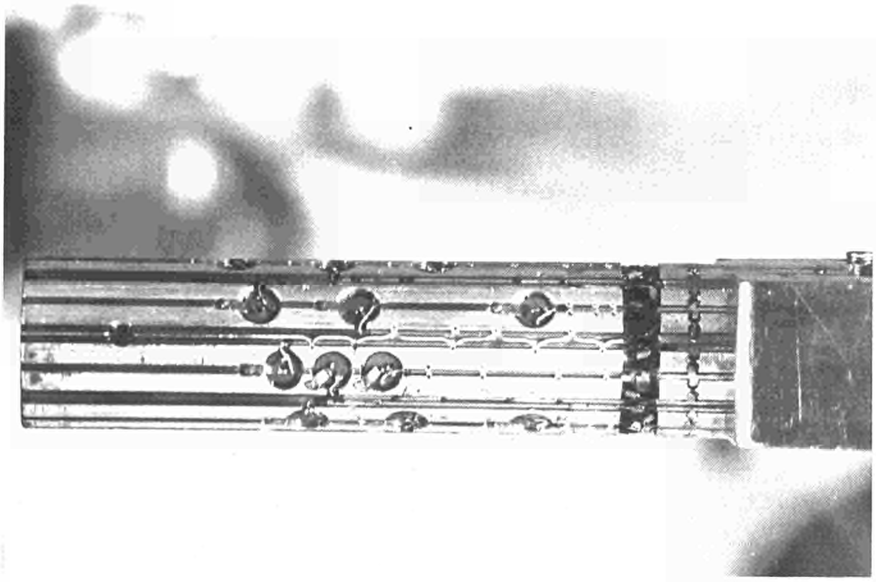


Fig. 47 Experiment R148-08 (HOBBIE). Deformation monitoring device after dismantling in the Hot Cell.

References to paragraph 2.2.3.3

1. D. Vader
Bestralingsverslag MONA-5.
RA Memo 80-45.

2. D. Vader
Irradiation of steel specimen in sodium, using a Refa facility. In-pile performance of R139-47.
RA Memo 80-21.

3. D. Vader
Irradiation of steel specimen in sodium, using a Refa facility. In-pile performance of R139-41.
RA Memo 80-30.

4. D. Vader
Irradiation of steel specimen in sodium, using a Refa facility. In-pile performance of R139-49.
RA Memo 80-34.

5. D. Vader
Irradiation of steel specimen in sodium, using a Refa facility. In-pile performance of R139-412.
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Irradiation report of the HOBBIE-8 capsule, Zircaloy fuel cladding irradiation for creep-down experiments.
ECN 81-010 (Dec. 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 10).
The technical problems reported in the last progress report had been solved. Within the reference period, 11 in-situ ramps and 7 other tests were performed. On 7th of October the 100th ramp test with a pre-irradiated fuel pin was completed. Comparison between first time schedule from 1976 and performed tests shows no delay in the intended irradiation programme.

On two fuel pins, irradiation after ramp testing in the previous period was continued for re-ramping after a burn-up increase of about 3 MWd/kg (U). In prototype tests with "fresh" fuel and a new displacement device (trolley) feasibility of continuous constant power change rate of 0,1 W/cm.min was demonstrated.
On 10 pre-irradiated fuel pins, pre-ramp examinations at the hot cells at Petten was completed. On 15 ramp tested fuel pins, post-irradiation examination was terminated.
A typical result from these examinations is given in Fig. 48 for a defective fuel pin.
Assembly of 3 capsule carriers and 10 irradiations devices was completed.

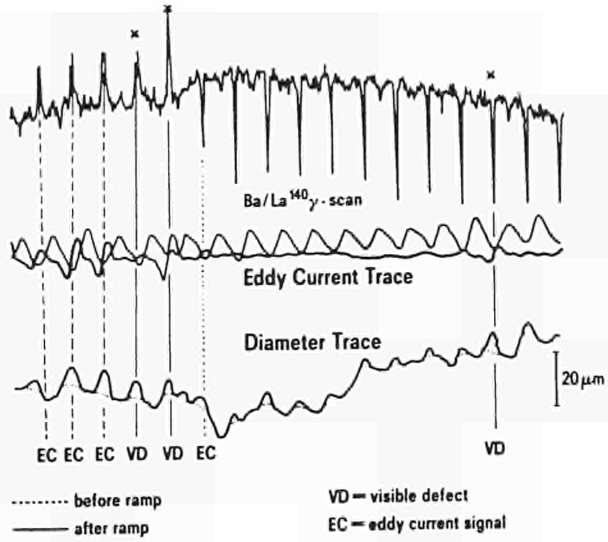

Fig. 48 Post-ramp irradiation examination of LWR fuel pins. Typical non-destructive testing results of a defective pin.

About 25 ramp tests per year are now anticipated for the period to 1985. Pre-irradiation of segmented fuel rods is continued at PWR and BWR power reactors. Fuel rods with a pre-irradiation time of four years will become available during begin of 1981.
In connection with the 100th ramp test a review meeting with all participating organizations and a symposium (refs. [2], [3]) were held at Petten on December 3, 1980.

Table 10 Number of power ramp test on LWR fuel pins.

Year	Ramp test types			Prototype resp. special tests	Total/Year, ( ) - with pre-irradiated fuel pins
	Start-up SU	In-situ IS	Modified in-situ ISM		
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	--	17	1	13	31 (27)
Total	6	73	27	37	143 (121)

D128. In-pile measurements on LWR fuel

D128-01

This experiment had to be stopped in the preceding period due to a leaking seal of the pressure vessel.
A repair procedure was developed and equipment for remote exchange of the defective component in the HFR pool has been prepared. Repair will be performed early in 1981.

D128-02

After termination of all post-irradiation examinations the fuel pin is temporarily stored in the LSO storage pool. Fig. 49 gives an impression of the appearance of the fuel pin after removal of the main capsule structure.

D128-03, -04

Two new experiments with sensors for fuel pin pressure and central temperatures are in fabrication and will become available for irradiation in the second half of 1981.

R174 "POTRA", In-core Power Transient Irradiation Facility, ECN project 8290.23

A description of this facility has been given in the previous progress report. All fabrication drawings for the in-pile part have been prepared and manufacturing can start early 1981.
Handling provisions and tools are under design; engineering and design of the out-of-pile BF_3 gas supply and control system will start early in 1981. The preliminary design and safety report has been assessed by the various internal review committees. Design approval has been obtained under the proviso that the nuclear operation of the facility will be reassessed as soon as detailed characteristics of the irradiation programme have been selected.

U187 Bumping of pre-irradiated fuel pins

An irradiation programme for Battelle Pacific Northwest Laboratory (BNW) sponsored by nuclear industry and organizations of practically all western countries, on high burn-up effects with emphasis on fission gas release has been agreed upon in August, 1980. Eight pre-irradiated fuel pins for this programme were delivered by KWU. Pre-bump preparation at the hot cells at Petten on the fuel pins were started. Fabrication of components for irradiation devices and equipment commenced.
The bumping irradiations - slow transient experiments with about 100h total irradiation time - are scheduled to start in February 1981. For the tests the same equipment as for the power ramping programme is used.

References to paragraph 2.2.3.4

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Experimental Strategy of Fuel Performance Testing with Respect to PCI.
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2. S. Krawczynski (KFA)
Stand und Tendenzen der Untersuchungen an LWR-Brennstoffstäben im Rahmen der amerikanischen und deutschen Sicherheitsforschung.
Presentation, JRC Petten, 3 Dec., 1980.
3. H. Stehle (KWU)
Verhalten von LWR-Brennelementen im Hinblick auf PCI unter besonderer Berücksichtigung der "Petten-Ergebnisse".
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Die elektrische MSR-Instrumentation der BWFC-Anlagen A und B.
Technical Note P/10/80/25, July 1980.

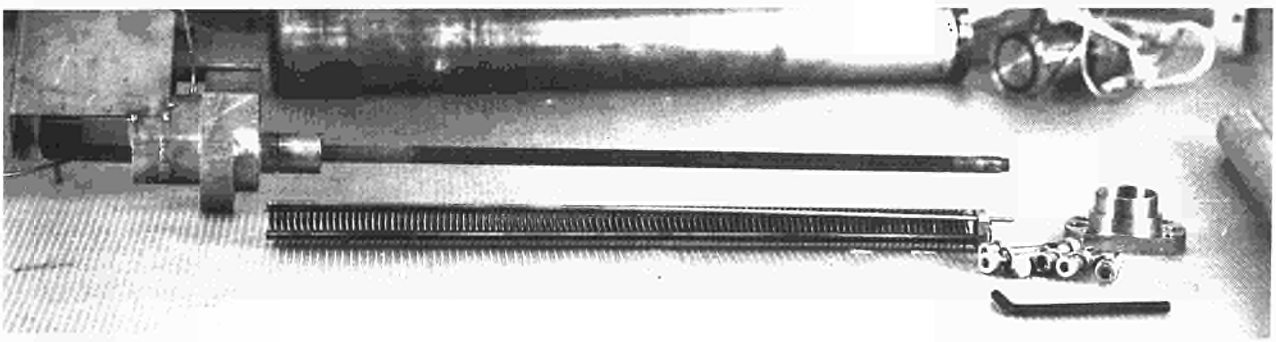


Fig. 49 Experiment D128-02. Post-irradiation removal of the fuel pin in the Hot Cells.

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 R124 in support of SNR 300.

b) transient test 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 SNR300.

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 High Temperature Experiments, ECN project 1.413

The irradiation programme has been terminated in 1979. During the present period the last irradiation report (experiment R54-F45) has been completed [1].
The draft for an external report has been prepared covering lay-out and operating experience of the SHOT irradiation facility.

R63 LOC, Fast Reactor Loss-of-Cooling experiment, ECN project 1.413

This irradiation programme concerns the behaviour of UO_2 fuel pins with a length of 25cm, of SNR300 specifications, 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 power	550 W/cm
Na-temperature	500 - 550 ^{o}C

while fuel burn-up (0 - 65 MWd/kg UO_2), internal fuel pressure (0 - 75 bar), shroud tube (open or closed) and loss-of-cooling time (10 to 20 sec) are varied for different experiments.

R63-20/-21

After the irradiation, capsules L20 and L21 had been dismantled, the fuel pins with surrounding sodium containment were returned to the reactor for neutron-radiography. Neutrographs, made directly after the irradiation, were of inferior quality due to leakage of the cooling channel.

R63-22

From 11-8-'80 to 1-9-'80 capsule L22 has been pre-irradiated up to a burn-up of 4,4 MWd/kg UO_2. On September 1, the cooling transient has been performed with a loss-of-cooling time of 17,5 seconds.
The capsule has been disconnected from the irradiation device. After neutron radiography the capsule has been transferred to the ECN dismantling cell.

R63-24

The pre-irradiation of capsule L24 started at 3-11-'80. The irradiation proceeded without any troubles. On December 21 the transient irradiation has been executed. The LOC time was 13,5 sec. and the pressure in the fuel pin was 10 bar.

R124 "TOP", Fast Reactor Fuel Pin OverPower 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.

- Data handling system

Programming and testing of the TOP-DHS went on successfully. The system was transferred from the ECN Electronics Department to the computer-room in the reactor building in November 1980.
Also the instrumentation-panels for the trace-heater and sodium pump supply units have been installed.
The construction of the control keyboards for the gas supply cabinets has been delayed. Delivery is now expected in January 1981. Thereafter all connections will be made, and the whole out-of-pile system extensively tested. Delays in the final design of the capsule necessitated replanning of the first irradiation to September 1981.

- BF_3 system

Due to the delayed delivery of an electromagnetic valve the replica of the BF_3 system could not be tested with BF_3. The valve arrived in December, and the tests will be carried out in January 1981.

The BF_3 distilling apparatus has been modified and tested as described in the previous report and is now fully operational.

Design and safety report

The draft report has been completed with some result of thermo-hydraulical- and stress-calculations still to be added. The report will be issued early in 1981.

Experiments in support of R124

R163 Instrumentation Test Capsule (INTEC)

The measuring results for the second irradiation cycle (80.01) have generally confirmed that, apart from the defective Schaevitz-transducer, all test instrumentation functioned in a reliable manner with only minor irradiation-induced signal drift.
As reported earlier the Schaevitz-transducer in the INTEC-facility behaved erratically and the intention to apply this instrument in the TOP-irradiation facility has now been abandoned. Instead, a Sybrook transducer, type LD-2.5/600 modified version, with dimensions specially adapted to TOP-application (Fig. 50) has now been selected.
The INTEC in-core facility has been transferred to the dismantling cells for post-irradiation investigation.

R164 BF_3 test capsule

No further irradiations were carried out. The evaluation of all results has been completed and will be reported shortly.

R165 Power measurement capsule

No further irradiations have been carried out. The test results obtained in HFR cycle 80-01 were evaluated and reported.

b) Operational ("mild") Transient Experiments (ref. [4])

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 pre-conditioning 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^2 and fuel can temperatures from 400 ^{o}C (bottom) to 600 ^{o}C (top), [2].
Post-irradiation ceramography, however, revealed pronounced radial gradients which has now been reduced by directed external neutron absorbers (Fig. 51). The resulting temperature patterns in the fuel have been analysed by the experimenter [3].

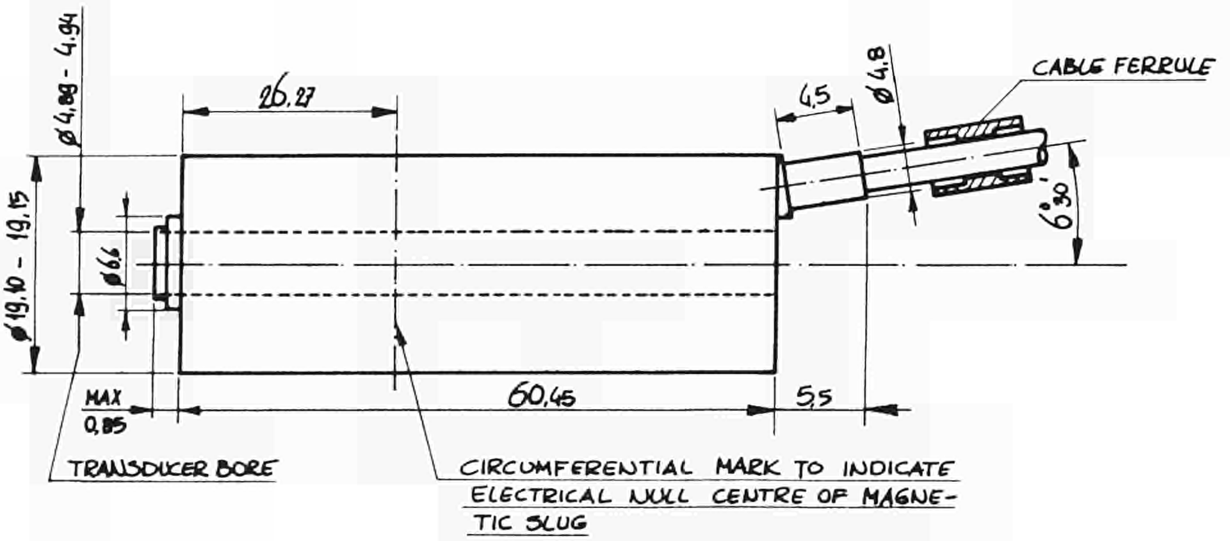


Fig. 50 Dimensions of the "SYBROOK" displacement transducer for application in the TOP capsule.

During the reporting period, the following activities have taken place :

- The outer absorbers have been tested by a dosimetry experiment at low reactor power with the fuel pins to be irradiated in DUELL 18/19. This measurement has confirmed the validity of the HIP-TEDDI calculations (see paragr. 2.1.3.6), showing that the outer absorbers reduce the circumferential neutron flux gradient from 40 °/o to about 10 °/o. The result seemed to justify further amelioration, calculated to be less than 5 °/o for the absorbers as fitted to the DUELL B carrier.

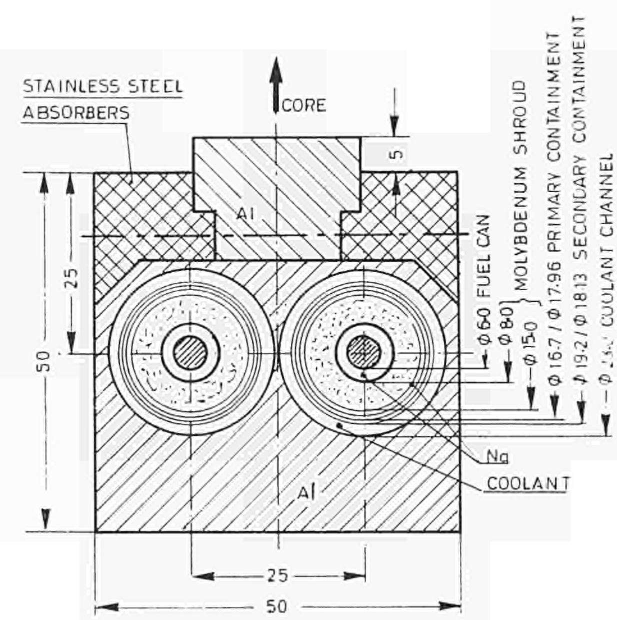


Fig. 51 D173, DUELL B. Typical cross section of the irradiation device.

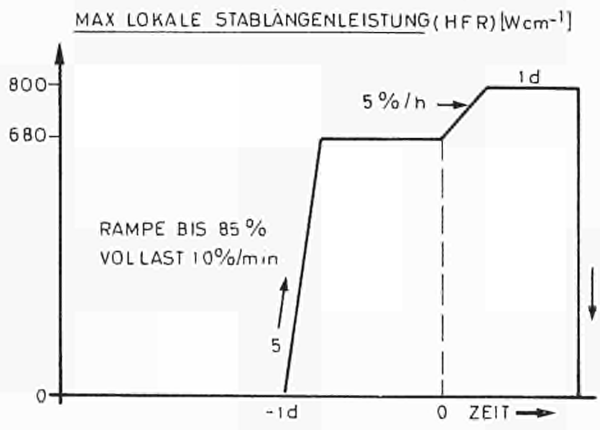


Fig. 52 Start-up ramp power vs. time profile.

- DUELL 18/19 is a B-type experiment, with outer absorbers, 6 mm O.D. fuel pins, high U-235 enrichment and 800 W/cm maximum peak power to attain FBR-fuel centre temperatures of about 2500 °C. The experiment is now ready for irradiation in January, 1981. A cross-section and power schedule are shown on the adjacent figures.

- for the DUELL C series, with 7,6 mm O.D. fuel pins, low U-235 enrichment and maximum peak power of 600 W/cm, the irradiation capsules are assembled, ready for fuel pin loading. Suitable absorbers shall be attached to the DUELL C carrier. Irradiation is scheduled in the first half of 1981.

- non-destructive post irradiation examination has been completed at ECN of the remaining six fuel pins of the DUELL-A series, and the pins shipped to KfK for final analysis. Preliminary conclusions say, that none of the pins have failed during irradiation and that the power transients have not induced measurable deformation or corrosion of the pins. During hot-cell handling all pins suffered a displacement of the fuel stack within the canning for no obvious reasons.

classification of DUELL experiments :

Series	Pin O.D. (mm)	Max. Peak Power (W/cm, fission)	Enrichment	Absorbers
DUELL A	6,0	565	HEU	None
DUELL B	6,0	800	HEU	Strong
DUELL C	7,6	600	LEU	Medium

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 initially forecasted the first irradiation in October, 1980.

The programme has again been considerably delayed by
. the uncertainty in fuel pin selection for testing,
. the flux gradient observed in the DUELL series,
. the request for axial markers on the fuel pins.

The working party decided in October, 1980, to test two of the existing KNK II/1 fuel pins, to apply absorbers similar to DUELL B and to mark the pin by 20-micron scratches equally spaced at 50 mm intervals along one meridian of the outer canning surface to detect local length changes by PIE.
Two new irradiation capsules have been assembled, made from cold drawn tubes, with ameliorated secondary containment seals. The capsules are ready for fuel pin loading (Fig. 53). Those assembled one year earlier, which suffered certain drawbacks, will be used for hot-cell tests.

A glove-box assembly has been set up to safely and reproducably mark the fuel pin axially, to accurately measure the marks and finally load the marked pin into the pre-assembled capsule (Fig. 54).
The equipment is ready to be operated in January, 1981.
The KAKADU carrier made earlier this year, has been modified to take the new shape required for the DUELL-absorber design. Irradiation is now scheduled for February, 1981.

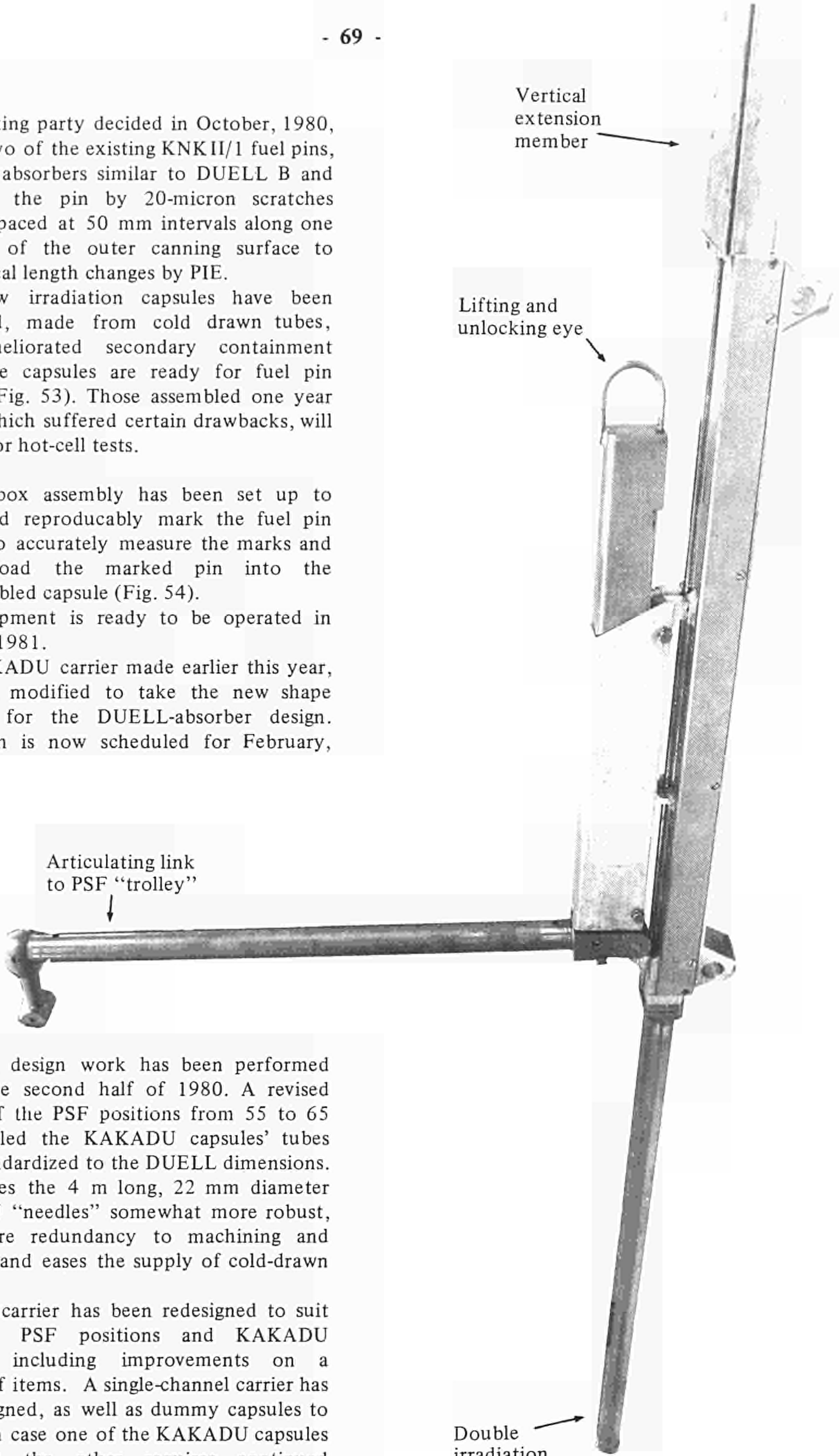


Fig. 53   KAKADU twin PSF capsule.

Extensive design work has been performed during the second half of 1980. A revised spacing of the PSF positions from 55 to 65 mm enabled the KAKADU capsules' tubes to be standardized to the DUELL dimensions. This makes the 4 m long, 22 mm diameter KAKADU "needles" somewhat more robust, gives more redundancy to machining and assembly and eases the supply of cold-drawn tubes.
Also the carrier has been redesigned to suit the new PSF positions and KAKADU capsules, including improvements on a number of items. A single-channel carrier has been designed, as well as dummy capsules to be used in case one of the KAKADU capsules fails and the other requires continued irradiation. It should be noted that
(a) those designs will not materialize for irradiation prior to 1982 and
(b) the KAKADU carrier accepts capsules with 6,0 and 7,6 mm O.D. fuel pins, but the capsules' internal parts are different, thus achieving a maximum of standardization.

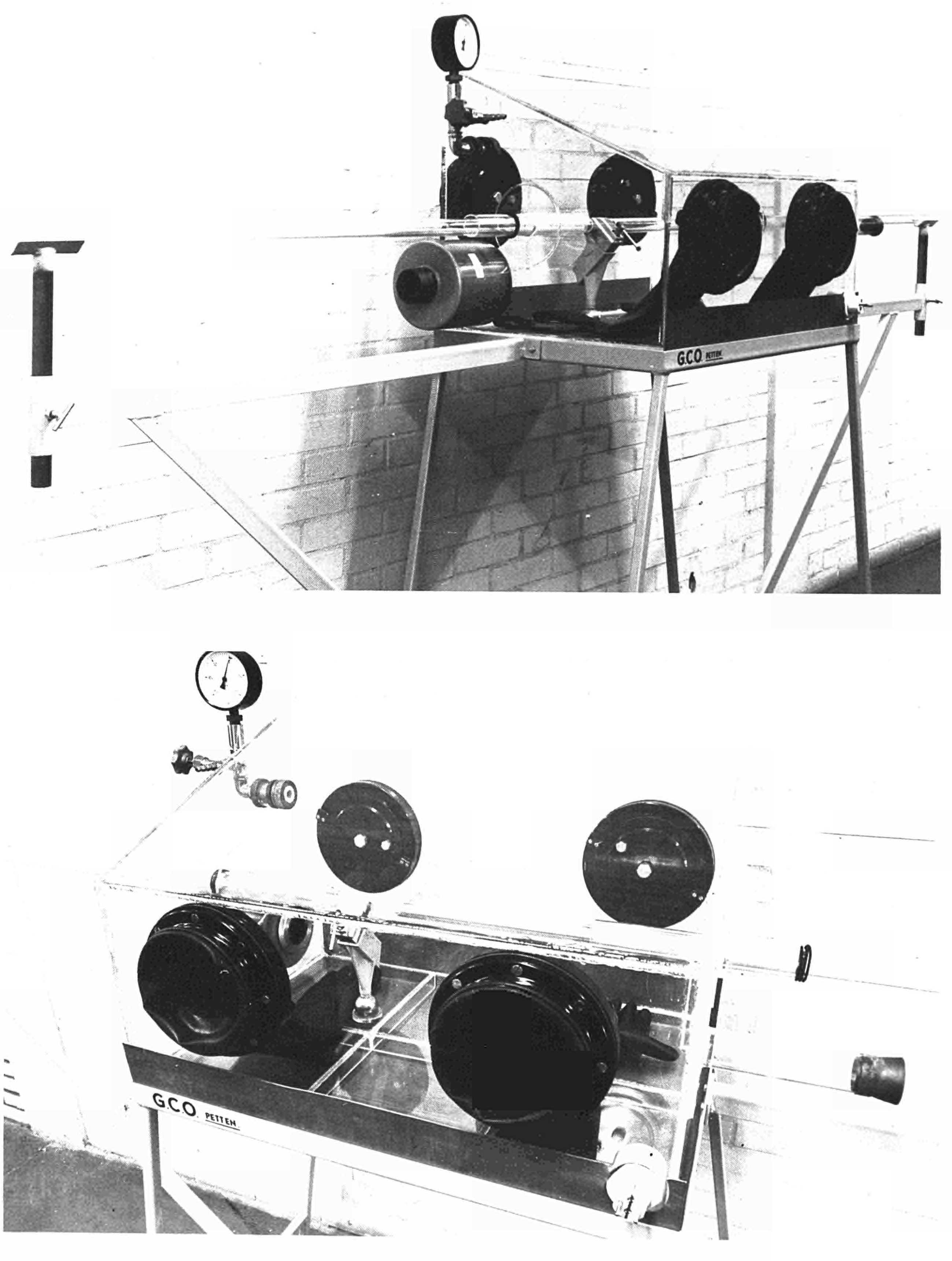


Fig. 54 Glove box facility for pre-irradiation handling and marking of LMFBR fuel pins.

Oxide fuel overpower experiments

These tests are designed to accurately determine the power level of incipient fuel melting.

D184 MONOX

During the reporting period, detailed drawings and a call for tender have been completed and thermal and nuclear analysis are ordered. This refers to the fast flux short time (FFST) facility and to the single irradiation capsule MONOX. The capsule tubes have been standardized to the DUELL design to ease the supply of cold-drawn precision tubes.

Preliminary nuclear analysis and discussions with the reactor supervisors have led to an alternative approach: it may be easier to perform the power-to-melt experiments in a special start of the reactor, instead of loading and unloading during reactor operation. In the case of a special start, three experiments could be irradiated simultaneously in the C5 position and the reactor power would be raised only to the level required for this experiment. The reactor would subsequently be shut down and the experiment replaced by those scheduled for the remaining cycle. In this case the TRIOX equipment would be used for better efficiency. The decision is pending awaiting detailed nuclear analysis.
The irradiation has been re-scheduled for early in 1982.

D192 TRIOX - Fast Flux Long Time (FFLT) facility.

Particular attention has been given to the $\Delta p, \Delta t$ power assessment in the FFLT design. It has been decided to maintain a rather large coolant flow cross-section to stretch a loss-of-coolant accident in time, but to introduce, as an improvement on earlier designs like CATRI, an adjustable restrictor to limit the coolant flow in favour of a better Δt signal and a mixing chamber for more accurate Δt measurement by type K thermocouples. A method is sought to safely apply artificial surface roughness to the cooled capsule surface, to improve the heat transfer capsule-coolant and to be able to further decrease the flow rate.
Detail drawings have been completed during the reporting period, and the manufacture phase is being prepared.
The irradiation has been re-scheduled for early in 1982.

D147 Carbide fuel testing in a PSF capsule (CAREL)

The second experiment of this series has been started in HFR cycle 80-07. The rise to full power is shown in Figs. 55 and 56.

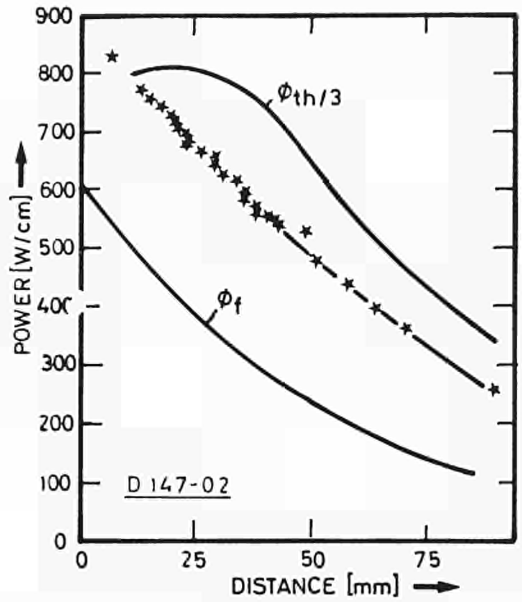


Fig. 56 D147-02. Maximum linear rod power vs. distance from core box wall, thermal and fast flux densities.

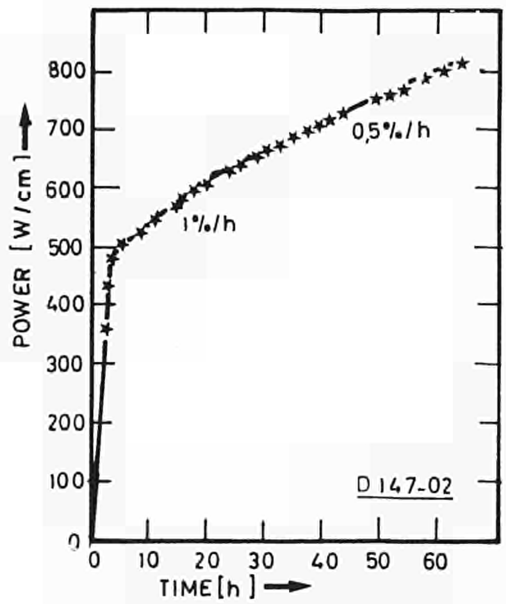


Fig. 56 D147-02. Maximum linear rod power vs. time during initial start-up.

The power of 800 W/cm has been attained as well as the required can temperature profile of 400 °C at the bottom and 600 °C at the top of the fuel stack. A burn-up slightly in excess of 1 °/o FIMA has been achieved after 74 days of steady-state full-power operation.
Subsequently, in November, 1980, power cycling started. The experiment is now operated during 12 hours at 800 W/cm and 12 hours at 400 W/cm for another 406 days to achieve 70.000 MWd/t total burn-up.

Neutron radiography using epithermal neutrons has been applied successfully for this experiment, showing even minor cracks in the high enriched UPuC fuel pellets, and an onset of crack-healing in the top part of the fuel stack after about 1,5 °/o FIMA.

D148 Carbide fuel testing in a TRIO capsule

The second experiment of this series has been started in cycle 80-11, despite late delivery of one of the fuel pins and severe manpower problems. Initial fuel conditioning is achieved (similar to D147) by slow rise to full power (about 60 h), which is illustrated in Fig. 57 by the fuel centre temperature versus time.

A fast neutron radiography shortly after start-up (Fig. 58) shows the crack pattern in the fuel pellets in comparison to the reference picture taken before irradiation. It should be noted (Fig. 59) that the Cd screens of the CATRI facility do not mask the fuel, thus unloading of the capsules from the facility for the purpose of intermittent neutron radiography is no longer necessary.
This is an additional advantage of the radiography technique using epithermal and fast neutrons.
The experiment is intended to be continued into 1982.

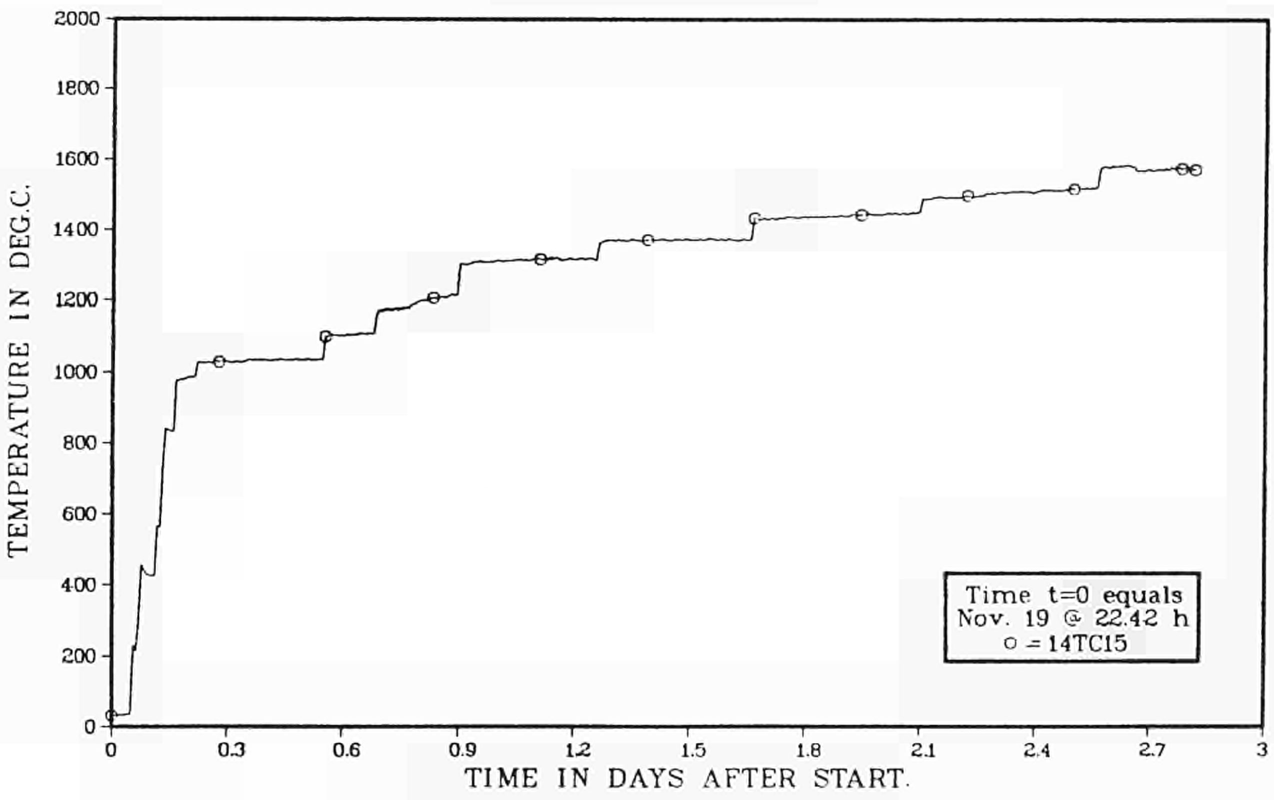
c) Other types of irradiations

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.
The drawings have been completed during the reporting period and all parts are under fabrication.
Begin of the irradiation is now scheduled in HFR cycle 81-08 (Sept. 1981).

Fig. 57 START-UP FUEL CENTRE TEMP. OF D148-02

Experiment CATRI.



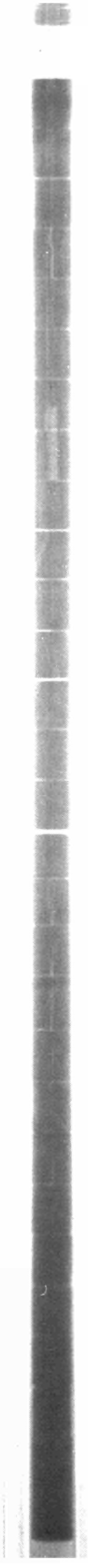


Fig. 58

D148-02 CATRI.
Single fuel pin
neutron radiograph
← before and after →
irradiation start-up.

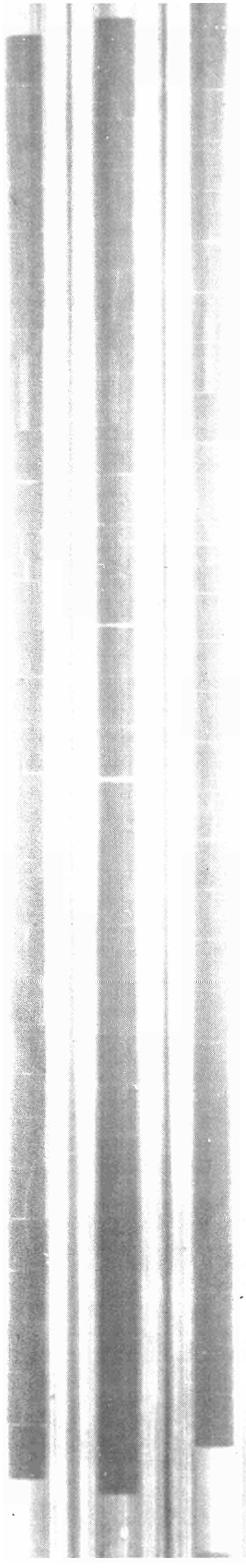
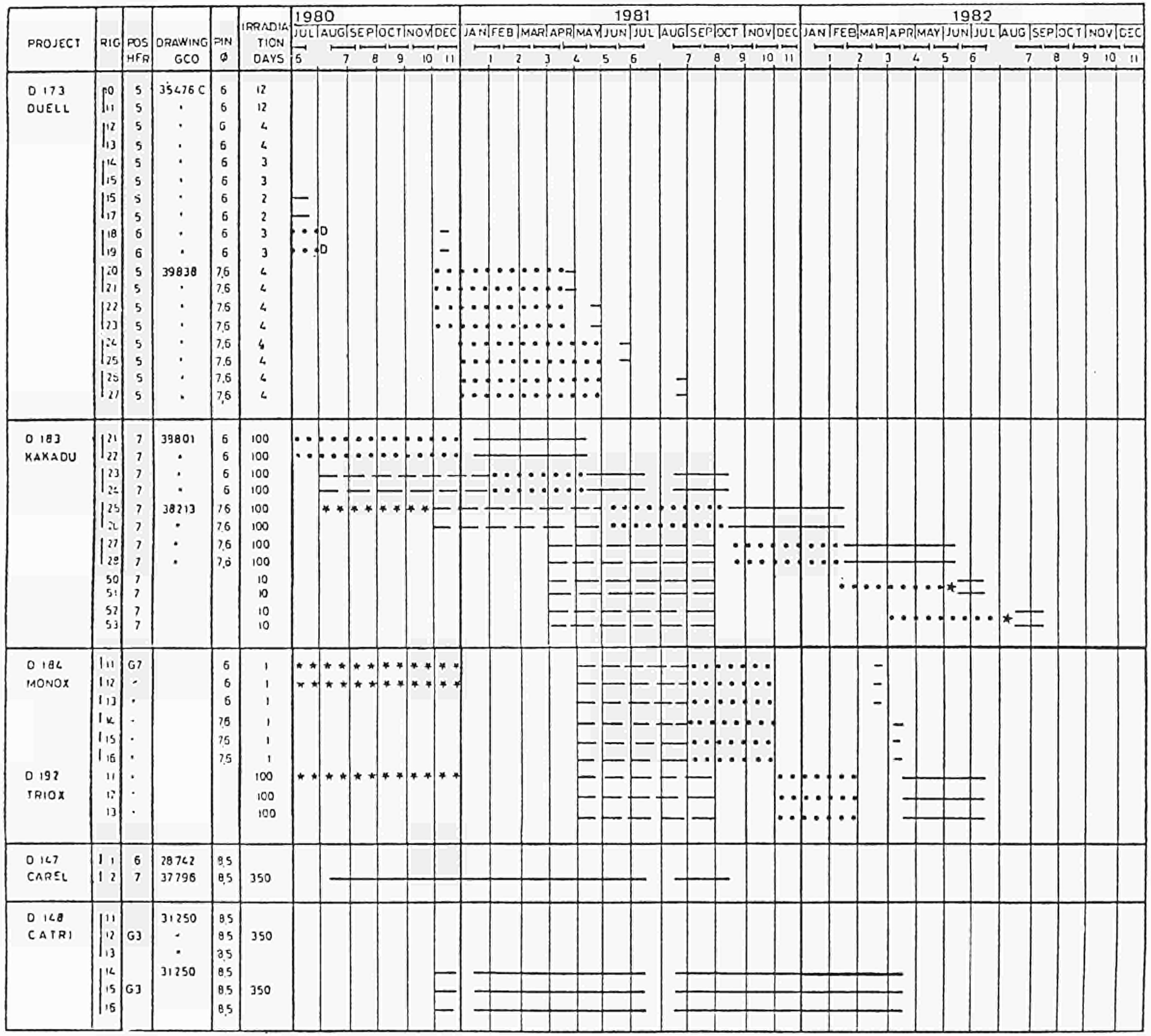


Fig. 59

D148-02 CATRI.
Neutron radiograph
of specimen carriers
inside their TRIO
facility.



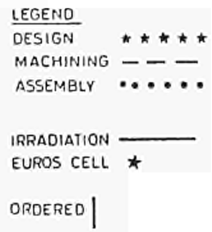


Fig. 60 LMFBR fuel projects. Time schedule updated : 7.11.1980.

E170-01 POCY, POwer CYcling experiment with fuel pin profile gauge

This experiment for JRC Karlsruhe, in which a mixed highly enriched carbide fuel pin was irradiated, has now been terminated, after successfully completing over 250 days of operation at linear powers of between 150 and 1050 W/cm, with inside clad temperatures up to 630 °C.
The fuel pin profile gauge performed effectively for the life of the experiment. Fuel pin recovery is scheduled to take place in early 1981.

D170-02

A similar experiment to POCY is scheduled to be irradiated for KfK late in 1981 or early in 1982.
The main differences are smaller pin diameter (7,6 mm), and lower power (540 W/cm).

D191 Fuel creep rig

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 KfK requirements have now been finalised and a modified proposal with the final design should be forwarded in January 1981.
Start of the irradiation is now scheduled for early in 1982.

References to paragr. 2.2.3.5

1. A.J. Nolten
Bestralingsverslag SHOT-capsule R54-F45. RA Memo 80-39.
2. F. Genet, F. Mason, P. Zeisser.
Irradiation of FBR fuel pins for start-up ramp tests. Projekt D173, DUELL A. Design and Safety Report.
Technical Note P/F1/80/35, Nov. 1980.
3. H. Steiner, H. Elbel
Die Berechnung von Temperaturasymmetrien in gekapselten Brennstäben mit dem Rechenprogramm TEXDIF-P.
KfK 2961 (June 1980).
4. W. Dienst et al.
Fuel-cladding mechanical interaction in fast breeder fuel pins; observations and analysis.
J. Nucl. Mat. 91 (1980), p. 73 - 84.

2.2.3.6 Miscellaneous

a) HFR vessel material. Surveillance irradiations (see paragr. 2.1.3.8).
Specimens have been irradiated in the existing RX092 facility, during all 5 cycles of the reporting period.
A new facility (RX189) is presently under development.

b) Gamma calorimeter (see paragr. 2.1.3.3)
The nuclear heating probe has been used during 4 cycles.

c) Radioisotope production, activation analysis.

1. Standard isotope facilities

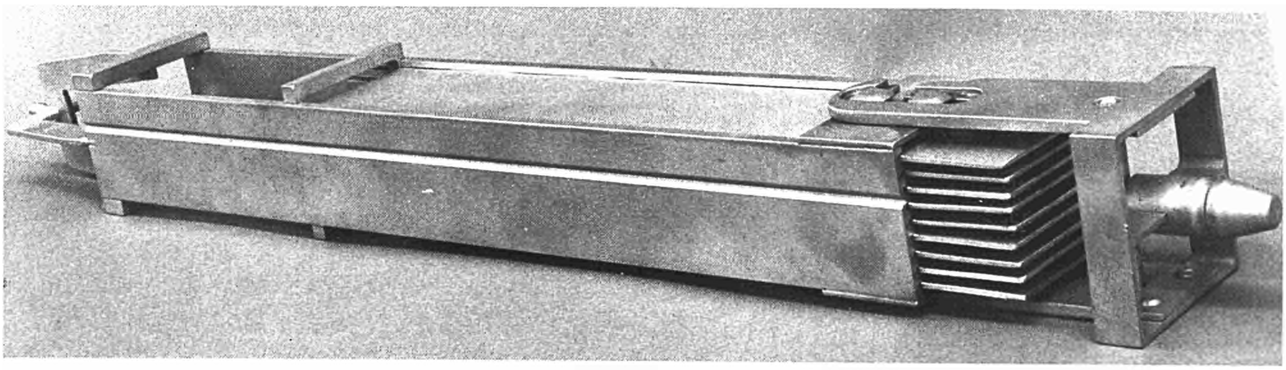
Their utilization during the reporting period is summarized in Table 11, and compared to previous years.

2. Special isotope facilities

ER136 ^{99}Mo production from fissile targets (FIT)
These facilities are also mentioned in Table 11.
During the reporting period, intense utilization of both PSF and in-core facilities continued. Increasing problems were experienced with the in-core-FIT due to wear and deformation of handling tool and locking mechanism. The manufacture of an improved in-core device has started, which will
1) eliminate the aforementioned problems,
2) accommodate 9 instead of 3 samples.
The new facility (ER136-05) is expected to be operational from about April, 1981.

ER144 High Flux Isotope Facility (HIFI)
The irradiation of iridium capsules in HIFI-02 continued on a routine basis.

ER179 ^{99}Mo production from 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 a maximum will be irradiated simultaneously during about 6 days and transported after 24 hours cooling to the reprocessing plant.
During the reporting period, the irradiation holder has been manufactured and tested (Fig. 61). The special transport cask has been received in Petten for testing of the different stages of manipulation and for the assembly of auxialiary handling tools and equipment.



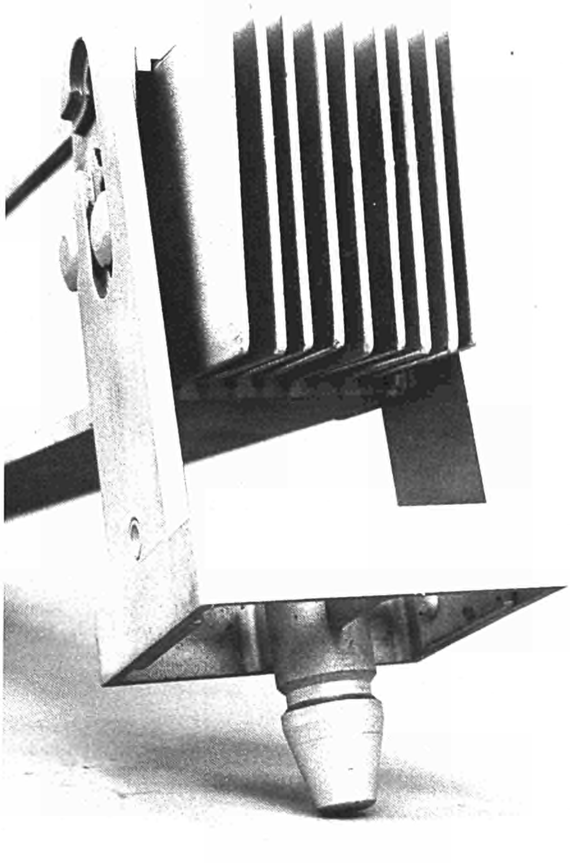


Fig. 61 ^{99}Mo production facility "MOLY" (ER179).

Overall view and detail of the target holder.

Table 11 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	second half 1980
PR 1	Pneumatic rabbit system	772	501	723	502	629	314
HR	Hydraulic rabbit system	6	11	36	34	26	14
ER6	Pool side isotope facility (PIF)	60	23	60	– *)	–	–
ER8	High flux pool side isotope facility (HFPIF)	15	9	30	58	49	48
ER70	Pool side rotating facility (PROF)	44	63	76	53	94	57
ER7-2	Reloadable isotope plug (RIP)	14	12	24	– *)	–	–
ER90	Reloadable isotope facility (RIF)	262	191	287	232	209	153
GIF	Gamma irradiation facility	*16*	*1*	*12*	*1*	*10*	*10*
ER136	Fissile irradiation facility (FIT) **)	12	--	27	12	10	38
ER144A	High flux facility for isotopes (HIFI)	–	10	89	83	31	17
	Total (except GIF)	1135	820	1352	974	1048	614

*) facility removed
**) each irradiation is composed of three targets

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.
In cycle 80-03 the experiment occupied position PSF 1, from 80-05 through 80-09 position PSF 9, using a specially designed capsule carrier.
The irradiation was terminated after cycle 80-09. Dismantling, specimen recovery and transport to Ispra are scheduled for end of January, 1981.

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 man-power/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

Fig. 62 Control console with new data logger.

2.3.3 Results

2.3.3.1 Experiment Control Installations (Proj. 310)

A number of standard control installations and other equipment for irradiation experiments are made available within the basic service package of HFR. They include

- microprocessor-controlled data loggers (Fig. 63),
- centralised data collection and processing,
- alarm collecting and processing units,
- gas supply and control circuits,
- cooling water supply and control circuits, etc.

During the reporting period, the last four of the new data loggers have been delivered bringing the total up to eight. This completes the change-over to the new generation of data-acquisition equipment together with an enlargement of the number of channels up to 980 (see Fig. 62 and paragr. 2.1.3.2).
A new capsule storage facility has been installed in pool nr. 3, doubling the number of safe storage positions for irradiated capsules (Fig. 64).
A re-designed filler element has been manufactured (Fig. 65) which is expected to resist to rough handling without damage.

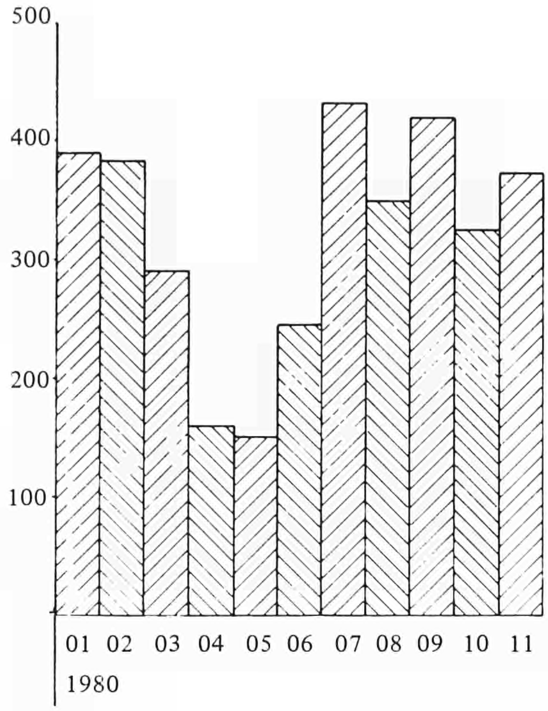


Fig. 63 Datalogger occupation 1980 (without special instrumentation like BWFC, D166, etc.).

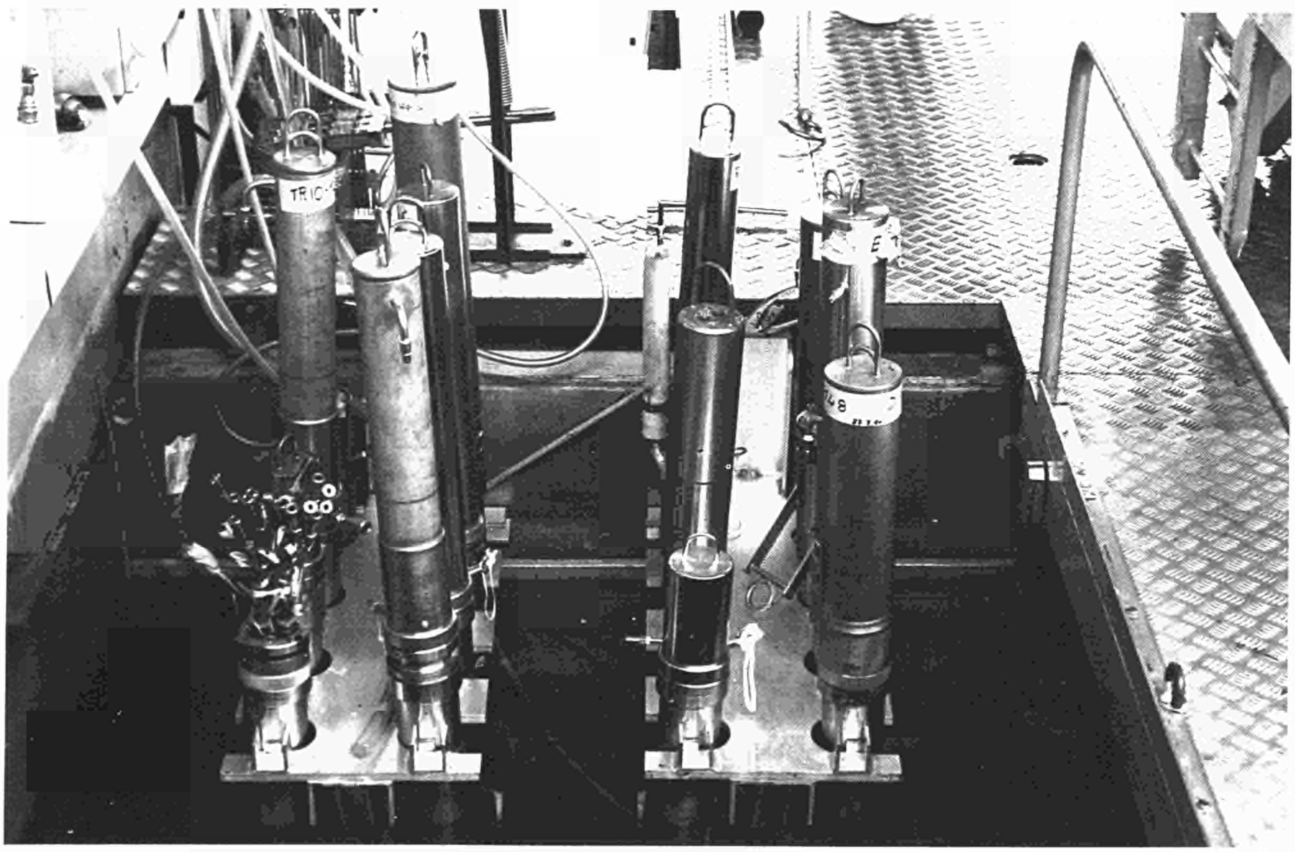


Fig. 64 Enlarged active storage facility.

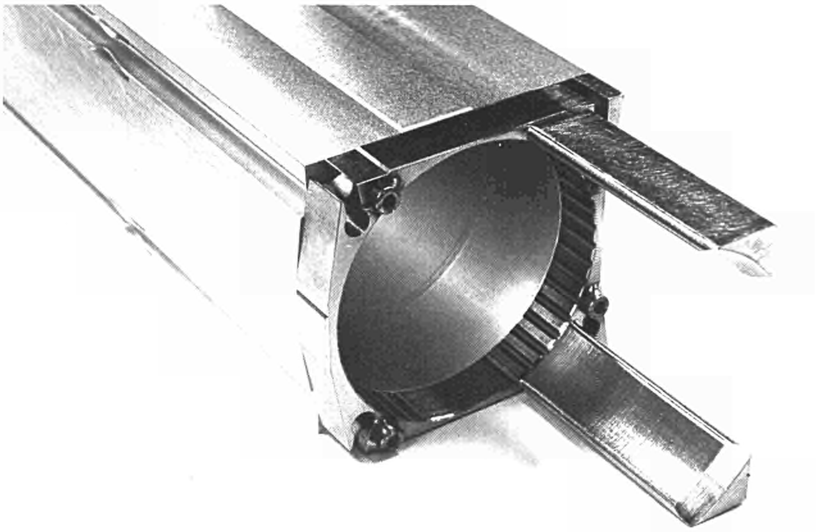


Fig. 65 Re-inforced design of HFR filler elements.

Fig. 66 New ("W3") shielded container for on-site transports.

2.3.3.2 Testing and commissioning (Proj. 300)

Acceptance tests of 31 sample holders and 4 complete irradiation devices have been performed during the reporting period.

2.3.3.3 Dismantling cell (Proj. 330)

The new shielded container for internal transports has been delivered and tested (see Fig. 66).

An adjustable carrier for a 15 t container next to the DM-cell has been ordered, as well as a special turn-over facility to be fixed onto the transport trailer. It will enable a 90° movement (horizontal $\rightarrow$ vertical) of the 17 t ILONCA container and a 60° turn of the EUROS container.
Modifications on the EUROS container were executed.
Development has been pursued of special hoisting facilities for containers.

Training of personnel and testing of several of the new containers have been performed. First handlings in the frame of the MOLY (ER179) project took place.
The cell team provided the following services during the reporting period :

- dismantling of 32 specimens carriers,
- preparation and surveillance of 29 on-site, 29 waste and 3 external transports,
- preparation for waste disposal (crushing) of 162 HFR fuel element end fittings and parts of control rods,
- disassembly of 11 obsolete capsules and supports.

2.3.3.4 Neutron Radiography (Proj. 340, ECN proj. 8.293)

a) Pool cameras

The existing camera showed a number of mechanical defects and leaks requiring two unloadings from the pool and ad-hoc repairs. Two actions have been initiated to improve the situation:
1) the front piece (diaphragm holder) has been redesigned and manufactured (Fig. 67).
2) Preparations for the specification, design and fabrication of a new in-pool neutron radiography facility have been started, by ECN.

During the second half year of 1980, 124 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.

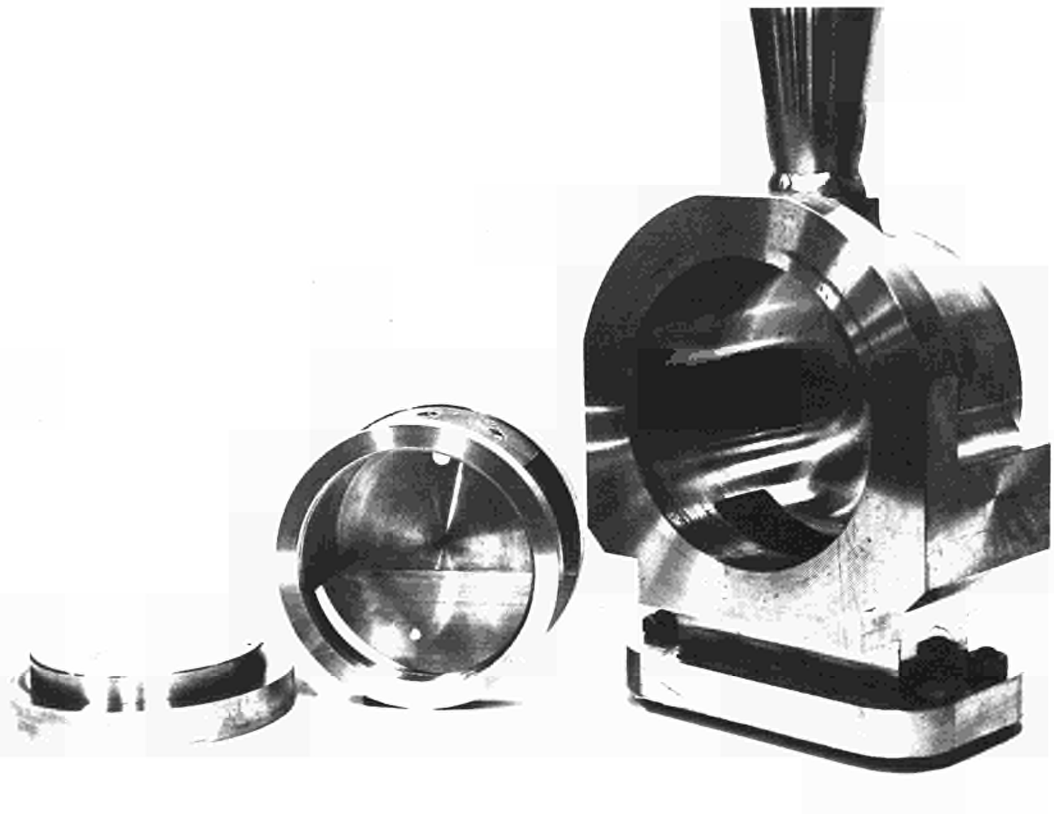


Fig. 67 New fixed-diaphragm "nose" of the pool neutron radiography camera.

b) "Dry" camera

As regards the beam tubes neutron radiography facility, the special drive mechanism for the operation on the central rotating plug as well as the drive and coupling mechanism for the displacement of the film camera have been installed.

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 reloading of active samples into TRIO-rigs have been carried out on the D156-12 and -20 series and on the D85-41, -42 and -43 series of experiments.
Transport between the hot cells and the HFR have successsfully been carried out with a new container (Fig. 68) which enables the direct loading of three radioactive sample holders into the hot cell; the transport thereof to the HFR and the direct loading into the three channels of a TRIO-rig .

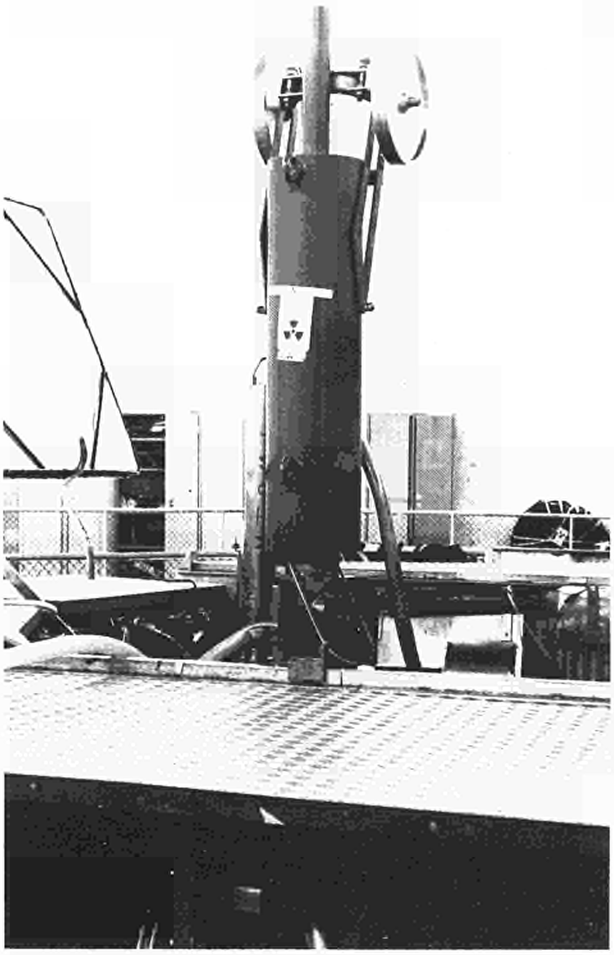


Fig. 68 Shielded storage, transfer, and loading container for specimen carriers with pre-irradiated specimens.

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.

The following work has been accomplished during the reporting period
- first handling, welding, and heating tests with a dummy capsule inside the cell box, and modifications of the "trolley" (Fig. 69 and 70),
- complete drawings and specifications for the electrical systems of the facility and their branching onto the Hot Laboratories (LSO) circuits,
- adaptation and testing of the special on-site transport container, manufacture of container handling and positioning equipment.

2.3.3.7 Standard Irradiation Devices (Proj. 407) Assembly Laboratories (Proj. 400)

a) In tank reloadable standard capsules

- The tendering procedures concerning HFR requirements until the end of 1984 are finally completed, and the first order will be placed early in 1981.
- An extensive study has been completed to standardise the in-pile capsules. The main changes which have been introduced concern standardisation of the thermocouple and gas line connectors and introduction of a standard position at which the sample carriers are connected to the vertical displacement units, where a rather complicated assembly is replaced by a commercially available mechanism. (A full report of this study is given in Technical Memorandum IT/80/1824).

  The final new design for TRIO capsules is shown on Fig. 71. An order for two prototypes is being placed. It is anticipated that after successful out-of-pile, and in-pile operation all in-core capsule heads will be converted to the new standard.

b) Assembly laboratories

The following devices have been assembled in the laboratories during the reporting period:
- 10 BWFC-RS/RD capsules,
- 8 specimen carriers,
- 2 complete devices, carrier and capsule,
- 12 PSF standard frames,
- 2 prototypes of TRIO heads with three independent vertical displacement units.

c) New PSF devices

The standard control units have been described in an Electronics report [2].
After introduction of the modifications and

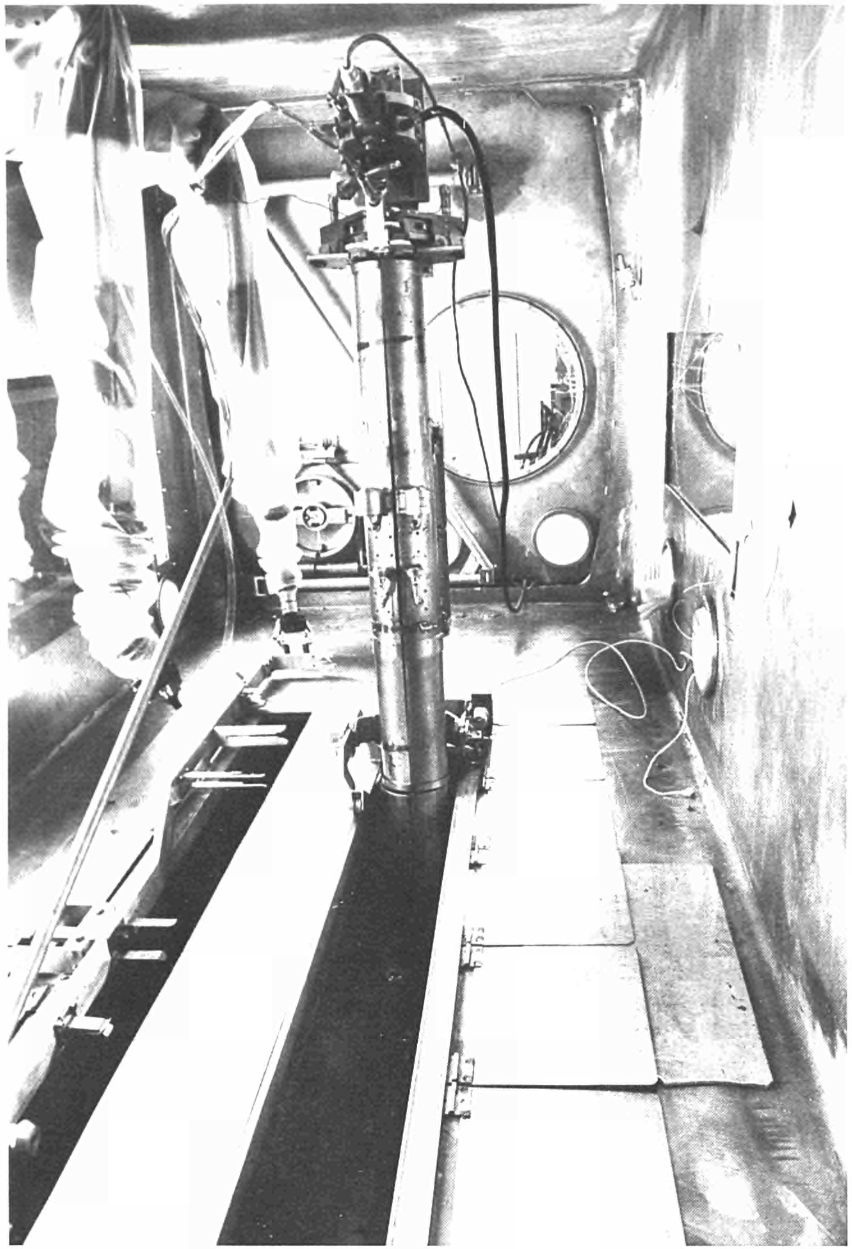


Fig. 69 EUROS Cell. Handling tests with the complete assembly furnace, welding head, and dummy capsule.

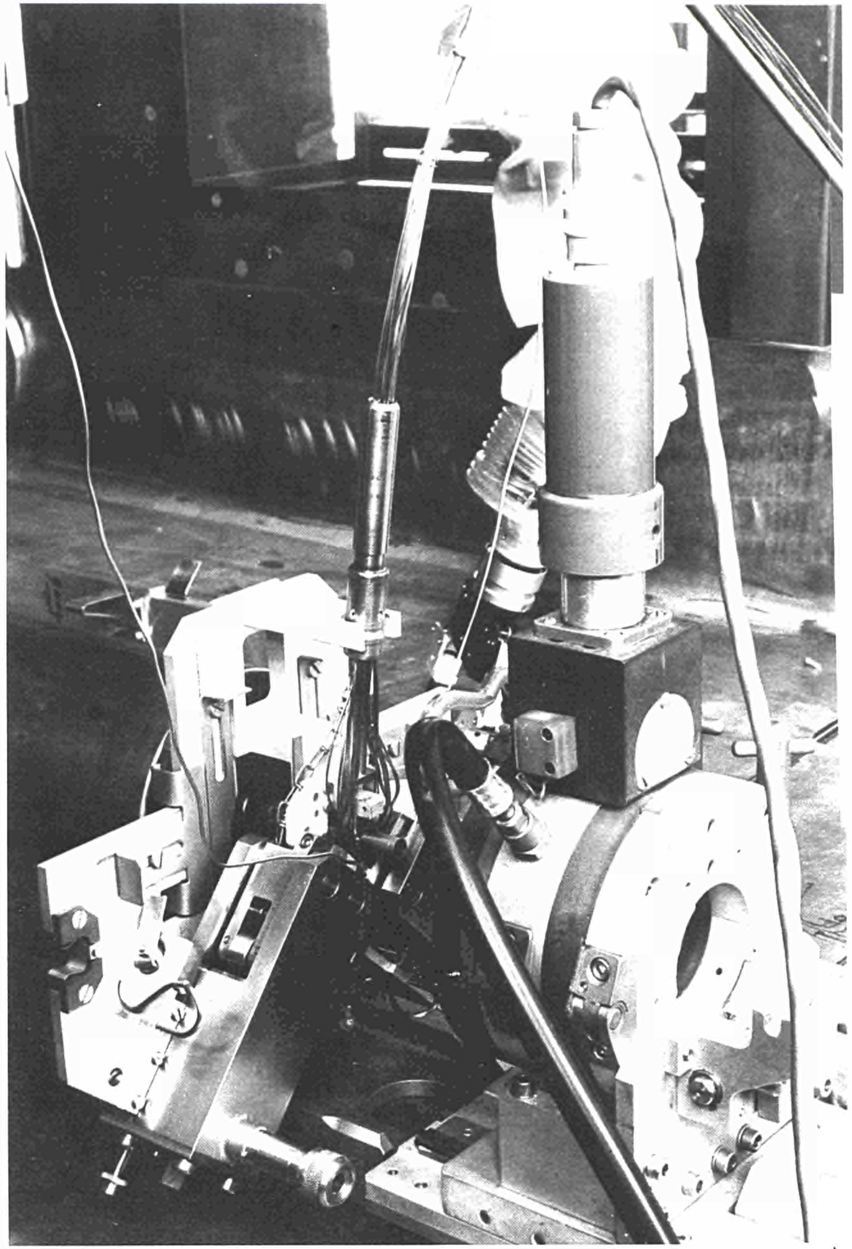


Fig. 70 EUROS Cell. Preparing a welding test with a dummy capsule.

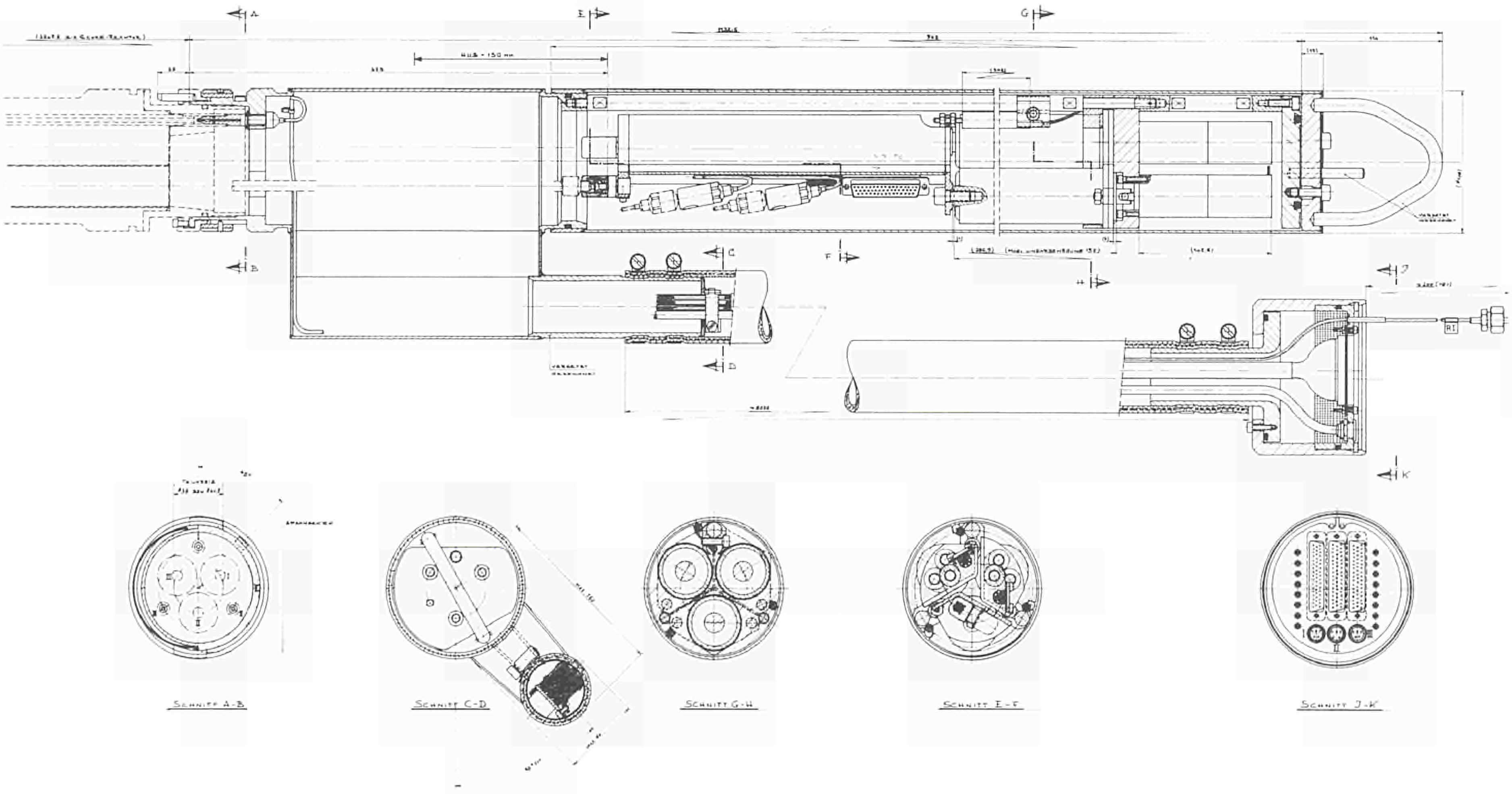


Fig. 71 New TRIO capsule head with three independent vertical displacement mechanisms using commercially available units.

improvements at the PSF table, trolleys and capsule carriers, reported in the previous HFR Programme Progress Report (January-June 1980), the PSF could fully be used for transient tests except for power change rates smaller than 10 W/cm.min.
Development of new drive mechanisms (trolleys) continued. A prototype trolley with a "push-pull" transmission was successfully operated an PSF 4. In a special test, feasibility of constant power change rates of 0,1 W/cm.min was demonstrated. A second type of drive system using a screw/nut drive was tested out-of-pile and will become operational at PSF 5 in cycle 81.01. Fig. 72a shows the trolley and Fig. 72b the drive unit for remote control of the trolley.

References to paragr. 2.3.3.7

1. W. Olthoff, A. Grosz
   VDU Remote Control Unit, Model 310.
   Technical Note P/F1/80/31, Sept. 1980.

2. W. Olthoff, A. Grosz, F. Roserot de Melin
   PSF Trolley Stuureenheid, Model 308.
   Technical Note P/F1/80/37, Nov. 1980.

2.3.3.8 Development of LWR fuel pins testing facilities (Proj. 405)

A central command unit for the eddy current and diameter measuring device for remote measurements on irradiated fuel pins in the HFR pools has been developed and installed.
The fast data logger "MINC-11" for digital recording and evaluation of eddy current and diameter measurement data was taken into operation. Software for data collection on MINC-11, and evaluation of data on HP 45-system was developed.
Investigations for installation of a non-destructive evaluation method for fission gas release determination were started. In a first feasibility check Kr-85 could be very well detected in the fuel pin gas plenum by special gamma scanning procedure.
Studies for a mini-loop for LWR fuel pin irradiations, a gamma scan system for the reactor pool and an extension of the non-destructive measuring systems for pool inspection of irradiated fuel pins were started.

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

Activities during the reported period were concentrated on :

- training courses for HFR experiment operators
- connection and testing of first experiment, E138-02 (see chapter 2.2.3.2)
- connection of data logger and preparation of experiment and process data evaluation programme
- assembly of Ge(Li) measuring station
- energy- and efficiency calibration of Ge(Li) detector for various measuring geometries
- final commissioning with connected experiment.

The final design and safety report was issued (ref. [1]). Design and operating approval were given by the EAC (Experiment Assessment Committee) and the RSC (Reactor Safety Committee).
A manual, giving instructions on basic operations and check-out procedure has been issued (ref. [2]).

References to paragr. 2.3.3.9

1. R. Conrad
   Sweep Loops. Systems for Operation, Control and Analysis of HTR Fuel Irradiation Experiments, Project 406. Design and Safety Report.
   Technical Note P/10/80/23 (July 1980)

2. F. Gasperini, R. Conrad
   Manual for Sweep Loops
   Technical Memorandum IT/80/1679,(Dec., 1980).

2.3.3.10 Computing Facilities

a) General

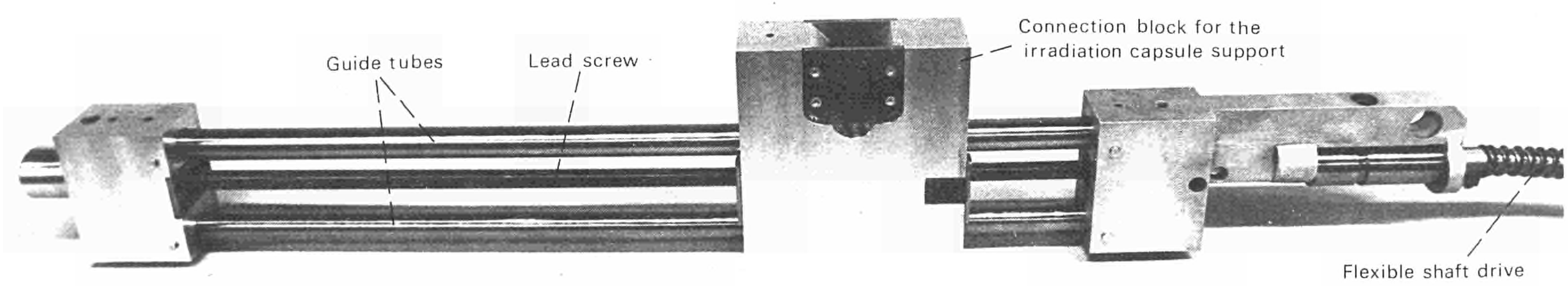
The ever increasing needs for design calculations and for data acquisition, processing, and retrieval have caused a considerable enlargement of facilities on one hand and the training of more and more staff for computer operation, on the other hand.

In 1980, the hardware used within the HFR programme comprised :

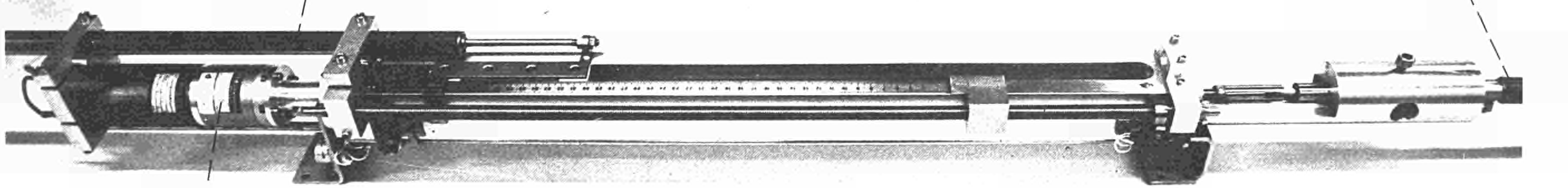
- the JRC PDP11/15 installation with multi-user access mainly working for BWFC data processing, special heat transfer problems and other tasks requiring specific software support.
  The old PDP 15 has gradually been taken out of service by transfer of its programs onto the enlarged PDP11/34.
- seven PDP11 mini computer based data acquisition and on-line treatment systems inside the reactor containment building.
- two HP9845 office computer installations (see below),
- the CDC CYBER175 batch terminal (see below).

b) HP 9845B desk top computers

One installation is used specifically for the BWFC Programme, 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 for calculations and data processing exist for general utilization.



a. horizontal "trolley".



b. vertical drive unit.

Fig. 72 New nut & screw PSF trolley.

During the reporting period, the following new developments have taken place :

Program: STRESS

Allows the stress analysis (thermal and pressure stress) in a cylindrical geometry (capsule and/or sample holder)

Subprograms for "GRAPHICS"

Consists in a set of 10 subprograms allowing plotting of linear and logarithmic as well as mixed coordinates These procedures save much time if a main program requires a plot on the thermal printer.

Improvement of program TEXPLO

In order to simplify the input instructions, this program was transformed into two different codes : TEXPLO allows the plotting of a text, and BUPLO allows the plotting of a set of blocks. The french and german characters were introduced.

A third HP9845 installation has been ordered by the middle of December 1980.

c) CYBER Batch terminal

The provided facilities include :
- line printer 300 lines/min.
- card reader 300 cards/min.
- paper tape reader 500 characters/sec.
- 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.

The ready access now obtainable to the main frame, has improved capacity to such an extent that waiting time for terminal usage is obviated.

A PERT type critical path program ("PLANIT") has been implemented, which regularised all the HFR experiments, from inception to dismantling and reporting on completion.
As an indication of main frame usage during the reported six month period, over 50.000 central processor seconds were used, and stored information required over 45.000 record block days.

2.3.3.11 Project Management

- Planning

The HFR planning meeting was held twice and two editions of the loading chart (nos. 42 and 43) were issued.
The working group of timing schedule and practical aspects for the post-irradiation phase of experiments met twice.

- Documentation

The 14th Newsletter of the "Euratom Working Group on Reactor Dosimetry" was assembled and issued.

A revised edition of the "Handbook of Materials Testing Reactors and Ancillary Hot Laboratories in theEuropean Communities" (EUR 5369, 1977) is under preparation.

The edition 1981/82 of the report "HFR Petten Characteristics of Facilities and Standard Irradiation Devices" (EUR 5700e) is being assembled.

- EWGRD

A first meeting of the EWGRD program committee for the preparation of the 4th ASTM-Euratom Symposium on Reactor Dosimetry, now scheduled for 22/26 March, 1982, at NBS (National Bureau of Standards), Washington, D.C., was held on 6/7 October 1980 at CEA-CEN Grenoble.
The subgroup "Irradiation Damage" met on 7 October 1980, also in Grenoble.

- IDWG

The 26th plenary meeting of the Group was held on 8/10 October 1980 at the GKSS Research Centre (F.R. Germany). The interest in the meeting was high and 37 papers were presented to 80 delegates. JRC Petten contributed six papers.

- NRWG

The organization of an international topical meeting on neutron radiography has been started, in collaboration with the societies for non-destructive testing of the USA, France, F.R. Germany and the United Kingdon. The conference will take place in San Diego, CA, USA, on December 7 - 10, 1981.

- ACPM

The Advisory Committee on Programme Management met in Petten on November 7, 1980. It reviewed status and progress of the HFR Programme on the basis of 10 documents prepared by JRC Petten.

2.3.3.12 Irradiation Technology Support and Development (ECN)

- In-pile crack-growth measurements

A technical feasibility study has been initiated for the development of an in-core irradiation facility for the on-line measurement of crack-growth during fatigue testing of metal samples. Discussions have been held with the ECN Materials Department in order to specify desired test condition characteristics and possible measuring and control methods and with manufacturers of fatigue-testing equipment on the possibility to adapt their equipment for under-water application within a very confined space. In the first stage of the project, a facility for in-core fatigue testing will be designed. Crack growth measurement feasibility will be taken up at a later stage.

- Heat pipes

A working plan for the development of a heat-pipe controlled irradiation facility has been submitted. Though preliminary technical assessment and thermodynamic calculations seem to support the contention that a heat-pipe facility will be superior in temperature distribution and control characteristics compared to a conventional mixgas-controlled facility, these advantages do not - at present - outweigh the technical and economical uncertainties of the further development of such a facility. Considering that the presently available facility reasonably satisfies experimental requirements it was decided not to pursue the heat-pipe project.

- Thermocouples

During the reporting period 350 mineral-insulated metal sheathed thermocouples have been welded and tested for a variety of in-core and other projects.
A new commercial fluidized-bath furnace was investigated for its applicability as an accurate calibration device. In spite of encouraging results, some improvement in stability turned out to be necessary. Therefore, a design has been drafted for a new and improved fluidized-bath furnace, in which thermocouple calibration can be performed with improved efficiency.

- Na melting in TOP facility

An investigation has been carried out with respect to melting behaviour of the sodium at the bottom of the sodium containment in the TOP capsule. This pocket contains the magnetic core of the fuel clad displacement transducer.
For a proper operation of the TOP capsule it is necessary that the sodium between the magnetic core and the pocket wall could be brought into a molten phase by means of the trace heating elements only.
In a test facility (Fig. 73) simulating the capsule part under consideration, several configurations have been investigated.
The experiments led to the conclusion that the original design had too little resistance between sodium pocket and heat sink to achieve sodium melting by trace heating only. An alternative design has been proposed.

- AUGIAS, Automatic gasmixing system for irradiation experiments

A proposal has been written for a gasmixing system which supplies desired gasmixtures to each irradiation experiment. Making the gasmixture and supply to the capsule and any following changing of the gasmixture are done automatically.
In this proposal it is also discussed where the mixing has to be done: outside the irradiation facility or in the capsule itself. If the mixing is done in the capsule itself, both gases are continuously flowing to the capsule. This method of temperature control is used in Harwell with good results.

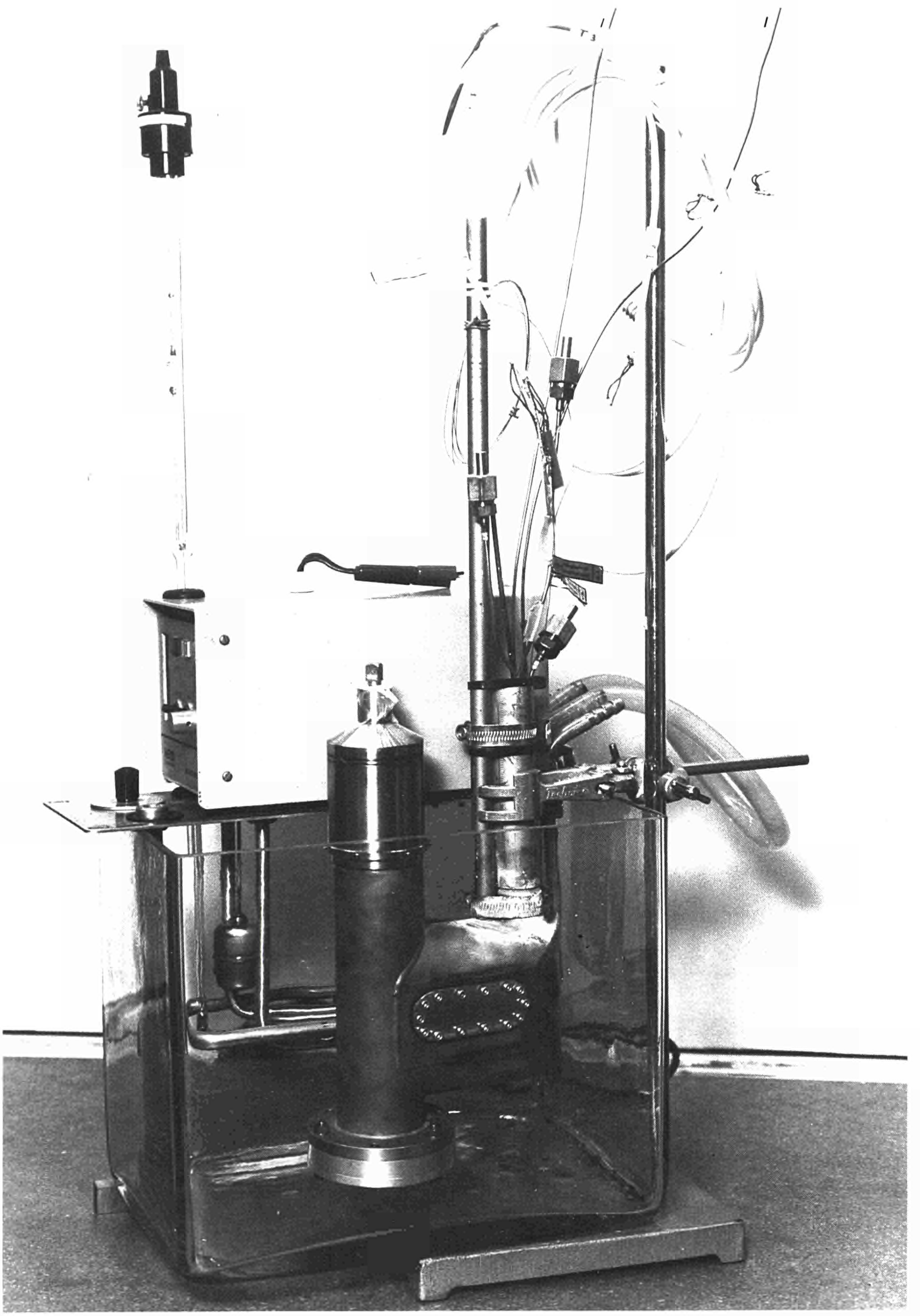


Fig. 73 Test device for the investigation of sodium melting in the TOP capsule.

Conclusions

3. CONCLUSIONS

3.1 HFR Operation and Maintenance

The operation of HFR Petten has been pursued during the second 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 (5) and unscheduled (13) 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 two maintenance periods, 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. Several new data loggers have been installed, and connected to a central processor.

3.2 Reactor Utilization

The average occupation could be brought to a satisfactory level during the reporting period.

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, development towards long-term irradiations,
- further developments for a new series of fast breeder reactor fuel transient irradiation tests,
- 18 ramp tests on pre-irradiated LWR fuel pins,
- permanent utilisation of nine horizontal beam tubes,
- intensified occupation of radioisotope production and activation analysis facilities.

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,
- manufacture and testing 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.

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

4. JRC PUBLICATIONS
HFR Programme, July - December 1980

- Programme Progress Report HFR, Jan. - June 1980.
  P. von der Hardt
  COM 3839

- Newsletter on Reactor Radiation Metrology, Nr. 14, November 1980.
  H. Röttger (Ed.)

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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