EIR-Bericht Nr. 388

EIR-Bericht Nr. 388

Eidg. Institut für Reaktorforschung Würenlingen
Schweiz

Gas Cooled Fast Reactor Research and Development Program

Progress Report 1979

G. Markóczy, M. Hudina, R. Richmond, P. Wydler,
R.W. Stratton, P. Burgsmüller

Würenlingen, März 1980

EIR-Bericht Nr. 388

GAS COOLED FAST REACTOR RESEARCH AND DEVELOPMENT PROGRAM

Progress Report 1979

G.Markòczy, M.Hudina, R.Richmond, P.Wydler, R.W.Stratton
P.Burgsmüller*

March 1980

*Sulzer Brothers Ltd.

S U M M A R Y

The research and development work in the field of core thermal-hydraulics, steam generator research and development, experimental and analytical physics and carbide fuel development carried out 1979 for the Gas Cooled Fast Breeder Reactor at the Swiss Federal Institute for Reactor Research is described.

Z U S A M M E N F A S S U N G

Die für den gasgekühlten schnellen Brutreaktor auf den Gebieten der Wärmeübertragung und -strömung, Dampferzeuger-Entwicklung, experimentellen und analytischen Physik und der Entwicklung von karbidischem Brennstoff am Eidgenössischen Institut für Reaktorforschung 1979 durchgeführten Arbeiten werden beschrieben.

GAS COOLED FAST REACTOR RESEARCH AND DEVELOPMENT PROGRAM

EIR Progress Report 197[illegible]

1. Introduction

2. GCFR Core Thermal-Hydraulics
   - 2.1. Overall Status of the R&D Program
   - 2.2. Investigation of Fundamental Thermal-Hydraulic Effects
   - 2.3. Computer Code Development
   - 2.4. AGATHE HEX Code Verification Experiment
   - 2.5. Benchmark Calculations
   - 2.6. International Meetings
   - 2.7. References for Chapter 2

3. GCFR Steam Generator Research and Development Studies
   - 3.1. Introduction
   - 3.2. Helical Tube Bundle Performance
   - 3.3. Steam Collector Design and Analysis
   - 3.4. Steam Generator Conceptual Design
   - 3.5. Finned Tube Technology
   - 3.6. Metallurgical Aspects
   - 3.7. Reference for Chapter 3

4. GCFR Experimental Reactor Physics Program
   - 4.1. Introduction
   - 4.2. Experimental Program
   - 4.3. Methods and Data
   - 4.4. Results and Discussion
   - 4.5. Conclusions
   - 4.6. References to Chapter 4

5. Nuclear Performance and Safety Studies
   - 5.1. Steam Entry Reactivity Effects
   - 5.2. Fuel Cycle Studies
   - 5.3. References to Chapter 5.

6. Development of Mixed Carbide Fuel
   - 6.1. Fuel Development
   - 6.2. Irradiation Studies

1. INTRODUCTION

Based on a Memorandum of Understanding concerning the collaboration in the field of the Gas-Cooled Fast Breeder Reactor (GCFR) between the Swiss Federal Institute for Reacotr Research (EIR) and General Atomic (GA) which was executed on 1967 and updated every year since, EIR and GA conduct a cooperative R&D program for the GCFR.

Whereas GA has been carrying out work in the areas of plant design, physics, safety and licensing, fuel and materials development and advanced system studies the EIR program included the following areas in 1979:

I. Research and Development in GCFR Core Thermal-Hydraulics
- Subchannel analysis code development
- Full scale thermal-hydraulics code verification experiments
- Investigation of special problems (flow distribution, mixing, spacer pressure drop etc.)

II. Reactor Physics and Safety
- Experiments with Th-metal blankets in PROTEUS, analysis and reports.
- Nuclear performance and safety studies for alternate fuel cycles and symbiotic reactor systems. Steam ingress and streaming effect calculations.
- Review of GCFR safety

III. Steam Generator Studies for GCFR

IV. (Pu,U) Mixed Carbide Fuel Program
- Review of basic process chemistry to give fundation for improved fuel quality leading to increased output.

By 1978 most of these activities have been incorporated into the framework of the Umbrella Agreement in the field of Gas-Cooled Reactor Concepts and Technology which was signed by Germany and the United States on February 11, and joined by France and Switzerland on September 30, 1977. The corresponding Project Work Statements were submitted.

The present progress report summarizes the Swiss activities for the GCFR conducted in 1978.

2. GCFR Core Thermal-Hydraulics

2.1. Overall Status of the R&D Program

The aim of the program is to develop analytical models and correlations for the prediction of the temperature and pressure distribution in gas cooled rod bundles in connection with the thermalhydraulic design of the GCFR fuel elements.

A five-step procedure was chosen at EIR and is being followed in achieving this aim:

1. Fundamental thermalhydraulic effects which influence the thermal behaviour of the GCFR fuel rod bundle are being investigated by literature survey, analytical studies and specific experiments. Based on the results of these investigations, models and correlations are developed.

2. Synthesis of the models in a comprehensive thermohydraulic design code that enables a sufficiently accurate thermal-hydraulic layout of the fuel elements.

3. Experimental verification of the computer codes by tests with electrically heated instrumented multiple rod bundles. Establishment of the accuracy and reliability of the codes.

4. Benchmark calculations with different available multichannel thermalhydraulic computer codes.

5. Parametric and sensitivity studies. Design calculations and recommandations for the design.

By the end of 1978 the first step was done. A few specific experiments are still going on in order to increase the accuracy of the correlations developed.

The second step is almost finished. Two GCFR design codes CLUHET and SCRIMP are working satisfactorily. The first version of CLUHET code, which is a more sophisticated code, was finished in 1978. Based on the extensive verification studies during 1979, this first version of the code was improved and described in a report.

The third step was initiated in 1976, starting with the measurements within a 37 rod bundle. This first bundle geometry was constructed in two variantes.The full measuring program of bundle 1 geometry was only carried out with the second variant (start 1977). In 1979 the second bundle geometry was tested. After some auditional tests in this geometry the change to the bundle 3 will follow in 1980.

Benchmark calculations with the different computer codes which can handle the GCFR conditions were started 1977. The first two calculation tasks dealing with 19- and 12-rod bundles were finished in 1977. The third to fifth tasks, based on the results of the 37 rod hexagonal bundle were calculated in 1978 on 1979. Further calculations will follow. The goal and schedule of these activities are to establish generally acceptable and experimentally verified GCFR analysis methods for turbulent flow (1978), laminar flow (1979), and transition flow conditions.

The expenditures of EIR for this program amount to 8350 k$ between 1967 and 1978. 1100 k$ were spent in [illegible] and 1020 k$ are planned for 1980.

2.2. Investigation of the fundamental thermalhydraulic effects

2.2.1. Rough surface thermalhydraulics

The measurements on the thermalhydraulic performance of artificially roughened rod surfaces were continued with a typical GCFR geometry of the ribs (rough rod of the AGATHE HEX experiment (1)) in a single pin test section. These tests were started in 1978 as a joint KfK-EIR experiment (2) and were continued in 1978 and 1979 with the different rods of bundle 2. The roughness form and the characteristic roughness dimensions are practically the same as for the bundle 1. The evaluated results show only a small change in friction factors and Stanton numbers for the entire anulus (Fig. 1 and 2). The results of the roughness functions for momentum (R) and heat (G) transports show the usual scatter of values but there is no systematic dependency upon roughness height respectively e/d_{h1} ratio. The final evaluation and the preparation of the report were initiated.

The influence of the rib profile on the friction characteritics of the rough surfaces was carried out with large scale roughnesses in an annular channel. An important effect of the rounded leading edge was found which confirms the theoretical assumptions. The results of the quantitative analysis will be summarised in a special report.

The new EIR method for the evaluation of thermohydraulic characteristcs of rough surfaces from measurements in annular channels, which was developed in 1978 (3) was further improved. The method uses a slightly modified eddy diffusivity concept applicable to the asymmetric turbulent flow. The evaluation of heat transfer results was extended to the calculation of the friction characteristics using the eddy diffusivity of momentum. The method was satisfactorily tested during 1979 by recalculating the results of the ROHAN experiment (4).

2.2.2. Investigation of the spacer pressure drop

Since 1972 the pressure drop on grid-spacers was extensively experimentally investigated in an air rig with actual bundle geometry (PROSPECT experiment) (5). Different spacer geometries were tested in a rather limited Reynolds number range. Additional experimental information obtained in the AGATHE HEX bundles confirms very well the results of the PROSPECT experiment. In order to confine with the verification of the analytical calculation model developed at EIR (6) (7) large scale spacer, to be measured in a 7 rod hexagonal bundle, was designed and constructed.

2.3. Computer Code Development

2.3.1. CLUHET code

The first version of the subchannel analysis computer code for gas cooled rod bundle calculations CLUHET was finished in 1978. To test this first version of the code the typical conditions of the AGATHE HEX bundle experiment were used. Based on the comparison between experimental results and analytical predictions some further improvement of the code were needed. Besides the new heat conduction subroutines (8), a new model for crossflow resistance was developed and incorporated. The mathematical solution for the differential equations was also improved, so that a considerable reduction of computing costs was achieved. To describe this new improved first version of the code, a report was prepared and will be issued early 1980.

2.3.2. SCRIMP code

In order to enable bundle calculations including radiation heat transport for the power tilt cases, the radiation routines, were reprogrammed. The calculation capacity could be increased, thanks to an improved program structure. The limiting number of subchannels and rod surfaces was extended to 32 and 94 respectively.

An extensive comparison of the measured results of the AGATHE HEX bundle 1 experiment with the analytical predictions was carried out. The results of this study were presented in a SCRIMP code verification report (9). Based on these comparisons the computer code SCRIMP can be considered as verified for the fully turbulent Reynolds number range ($Re > 2.10^4$).

2.4. AGATHE HEX Code Verification Experiment

To verify the design codes under realistic operational conditions the AGATHE HEX experiment (1) with an extensive measurements program was started. This test program covers different bundle geometries and a wide range of Reynolds numbers ($700 \leq Re \leq 4.10^5$). The cross sections of the different bundles are given in Fig. 3 and the main characteristics of the bundles are summarised in Table 1. The length of the test section, the spacers, the surface roughness and the roughened length are the same as for the GCFR core. The measurements were started in 1976 with the first test section of the bundle 1 geometry (37 heated rod bundle). A complete measurements program including power tilt tests was carried out and finished in early 1978.

The analytical predictions of the SCRIMP code in the whole Reynolds number range were compared with measured results (9). The summary of the deviations between predicted and measured values is given in Table 2. The quality of the predictions(maximal deviations), together with its definition, over the total Reynolds number range is presented in Fig. 4 for bundle pressure drop and in Fig. 5 for surface temperatures.

The calculations with the first version of CLUHET code were also started in 1978 (see fig. 6 to 8). Considering the important improvements of the CLUHET code in 1979, some of these calculations will be repeated. The first predictions were not satisfactory for the power tilt cases (10). Better results can be expected with the improved version.

The second bundle with 31 heated rods was fabricated during 1978 and the test section installed at the end of the year. The measurements with this bundle, characterised by the flat shroud walls were started at the beginning of 1979. After the measurements with uniform heating of high and transition Reynolds numbers, some heater failures stopped the measuring program. The investigation of heater rods showed that the failure was located inside the rod at the contact between nickel and copper leads. At the end of 1979 the measurements were continued using the Bundle 1 rods. The measurements program together with some additional tests will be finished in 1980.

In the Fig. 9 and 10, one of the available measuring cases of Bundle 2 is presented together with the analytical prediction of pressure and temperature distributions.

Extensive thermal-hydraulic calculations with SCRIMP and CLUHET computer codes were carried out in order to check the GA assumption that introducing the large solidity of spacer will reduce diametral temperature profiles to 25°C (11). The EIR results could not confirm these optimistic values. Based on the design experiences with bundles 1 and 2, the calculation results of different codes and the suggested EIR choice criterion, the basic design of the bundle 3 was determined (12), (see also Fig. 11 and 12) The construction and production of different parts of bundle 3 was started in 1979.

2.5. Benchmark Calculations

In order to compare the analytical predictions of different available rod bundle computer codes, benchmark calculations were started in 1977. The first two cases (19 rod (13) and 12 rod (14) bundle) show that the results of different codes agree very well at high Reynolds numbers, if the basic calculation conditions are identical.

The third, fourth and fifth cases were based on the AGATHE HEX experimental conditions. The needed data were always summarised by EIR, (15) (16) (17). In addition to the original participants GAC, KfK and EIR, these cases were also calculated by CEGB Berkeley and UKAEA Windscale. The short review of the Benchmark meetings was given in the Minutes (18) (19) (20). By comparison with the previous calculations further important reduction in the scatter of the results of different codes for uniform heating and particularly at high Reynolds numbers was observed. Starting from the untransformed single rod experimental data, different correlations for the roughness functions R and G were obtained but thanks to consistent transformation methods applied, the bundle results were close together. The aim of the fourth benchmark calculations was to prove the effect of conduction heat transfer. This was clearly demonstrated, only by the codes SCANDAL and SCRIMP having the possibility of comparison of results with and without conduction. It was agreed that the conduction is important and should be generally included in the calculations. The task of the fifth benchmark calculations was to demonstrate the effect of the heat transfer by radiation. Unfortunately the calculation of some codes was not finished in time or not complete. Because of the leak of simple comparison possibility no important conclusion were made in the short time available for the benchmark meeting. An additional internal meeting was made at EIR in order to assure the full benefit from the analysis of these calculations. Experiences obtained since start of this program clearly show, that in addition to the experimental investigations, the benchmark calculations are the most important procedure for the verification of the computer codes.

2.6. International Meetings

The 5th Heat Transfer Specialists Meeting of the OECD-NEA Co-ordinating Group on Gas-Cooled Fast Reactor Development was held in Würenlingen, Switzerland, 14 to 16 May 1979. 26 participants from 8 differen countries (B, CH, D, I, NL, S, UK, USA) attended this meeting. 33 papers were presented by 18 participants in 6 sessions. The contributions of EIR to GCFR research activities was presented in 7 papers, one of them was commonly writen by KfK and EIR. The summary of the sessions together with some other important informations was edited by R. Ekholm (21).

2.7. References for Chapter 2

(1) M. Hudina and G. Markòczy: The hexagonal bundle heat transfer and fluid flow experiment AGATHE HEX. Nuclear Engng. Design, Special Issue, 40, 121-131 (1977).

(2) M. Dalle Donne, M. Hudina, M. Huggenberger, L. Meyer, K. Rehme: EIR, KfK Joint Heat Transfer Experiment on a Single Rod, Roughened with Trapezoidal Rounded Ribs and Cooled by Various Gases.
EIR-Bericht Nr. 349, Würenlingen 1978 also KfK 2674, EUR 5755e.

(3) M. Hudina: Evaluation of heat transfer performances of rough surfaces from experimental investigation in annular channels.
Int. J. Heat Mass Transfer, Vol. 22. pp, 1381-1392, 1979.

(4) M. Hudina: ROHAN experiment (final report) Part 1.
EIR-TM-IN-694, Würenlingen 1977.

(5) P. Barroyer: PROSPECT II: Experimental facility -- program of measurements and principles for the data reduction.
EIR report TM-IN-689/I. January 25, 1977.
Experimental results and analytical prediction of spacer coefficients. EIR-report TM-IN-689/II December 15, 1977
PROSPECT III: Rhombic Spacer Pressure Drop Coefficient in Smooth and Rough bundles.
EIR-TM-IN-734, Würenlingen 1978.

(6) P. Barroyer: Analytical Model for the Prediction of Spacer Pressure Drop Coefficients.
EIR-Bericht Nr. 350, Würenlingen 1978.

(7) P. Barroyer: Analytical Model for the Prediction of Spacer Pressure Drop Coefficients. Transactions of the ANS 1978 Winter Meeting, Washington, 1978, Vol. 30, pp. 525-526.

(8) P. Barroyer: Subchannel analysis computer code CLUHET. Status Report SL-R/28, Paper 2 of session IV. NEA GCFR heat transfer specialists meeting, Würenlingen, May 1979.

(9) M. Huggenberger: AGATHE HEX EXPERIMENT. Verification of the thermal-hydraulic computer code SCRIMP by the 37-rod bundle. EIR-TM-IN-801. Würenlingen, Oct. 1979.

(10) G. Markòczy and M. Hudina: AGATHE HEX bundle experiment. Review and status report. SL-R/25, Paper 5 of session III., NEA GCFR heat transfer specialists meeting, Würenlingen May 1979.

(11) C.B. Baxi: Re-analysis of AGATHE HEX Bundle. GA Internal Correspondence 790430240, 30.4.1979.

(12) P. Barroyer, M. Huggenberger: AGATHE HEX EXPERIMENT. Design and Instrumentation of Bundle 3 EIR-TM-IN-800, 11.10.1979, Würenlingen.

(13) K. Rehme: Experimentelle thermo- und fluiddynamische Untersuchungen an einem 19-Stabbündel mit künstlichen Oberflächenrauhigkeiten, KfK 2313, Karlsruhe 1976.

(14) M. Dalle Donne et al: BR-2 bundle mockup heat transfer experiments. Nuclear Engng. Design, Special Issue, 40, 143-156, (1977).

(15) M. Hudina, P. Barroyer, M. Huggenberger: Benchmark Calculation III. Task Definition. EIR-AN-IN-684, Würenlingen, 1977.

(16) M. Hudina, M. Huggenberger: Benchmark calculation IV.
Task definition.
EIR-AN-IN-796, Würenlingen 1978.

(17) M. Hudina and M. Huggenberger: Benchmark calculation V.
Task definition.
EIR-AN-IN-852, Würenlingen, 1978.

(18) M. Hudina: Minutes of the III. Heat Transfer Benchmark
Calculation Meeting. Würenlingen, 13-14 March, 1978.
EIR-AN-IN-723, Würenlingen, 1978.

(19) M. Hudina: Minutes of the IV. Heat Transfer Benchmark
Calculation Meeting, Würenlingen, 17-18.5.1979.
EIR-AN-IN-857.

(20) K. Rehme: Minutes of the 5th Heat Transfer Benchmark
Meeting. Karlsruhe, Dez. 13 and 14, 1979.

(21) OECD-NEA GCFR PROGRAMME: 5th GCFR Heat Transfer Specialists
Meeting, May 14 to 16, 1979, Würenlingen.
Edited by R. Ekholm, Studsvik Energietechnik AB. CO-15,
1979-09.

TABLE 1 CHARACTERISTICS OF THE AGATHE HEX BUNDLES

	Bundle 1	Bundle 2	Bundle 3
Number of rods in a hexagonal array	37	31	34
Heated length (mm)	1150	950	950
Roughened length (mm)	850	600	950
Roughness type, trapezoidal profile			
Nominal roughness height (mm)	0.1	0.1	0.15
Nominal rib pitch-to-height ratio	12	12	12
Nominal rib width-to-height ratio	3.5	3.5	3.5
Distance between spacers (mm)	200	250	250
Rod outer diameter (mm)	8.4	8.4	8.4
Rod pitch to diameter ratio (mm)	1.3	1.3	1.5
Maximum gas temperature (°C)	500	500	500
Maximum cladding surface temperature (°C)	750	750	750
Maximum heat flux (W/cm^2)	80	100	80
Thermocouples per rod			
Standard	4	4	4
Maximum	12	12	12
Rod heating	indirect dc	indirect dc	indirect dc
Maximum radial power tilt	2/1	1.8/1	1.5/1
Axial power distribution	constant	constant	constant

37-Rod Bundle			MAXIMAL DEVIATIONS OF ANALYTICAL PREDICTI		
HEATING POWER DISTRIBUTION	FLOW CONDITIONS	NO OF TEST RUNS CONSIDERED	TOTAL PRESSURE DROP	SURFACE TEMPE	
				SMOOTH PART	RO
UNIFORM	TURBULENT	23	+ 4 - 10	+ 8 0	
	TRANSITION	7	- 10 - 25	+ 18 + 5	
	LAMINAR	6	- 8 - 17	+ 8 0	
POWER TILT	TURBULENT	2	- 12 - 16	+ 7 0	
	LAMINAR	1	- 14	± 22	

TABLE 2. SUMMARY OF THE DEVIATIONS BETWEEN PREDICTED AND MEASURED VALUES.

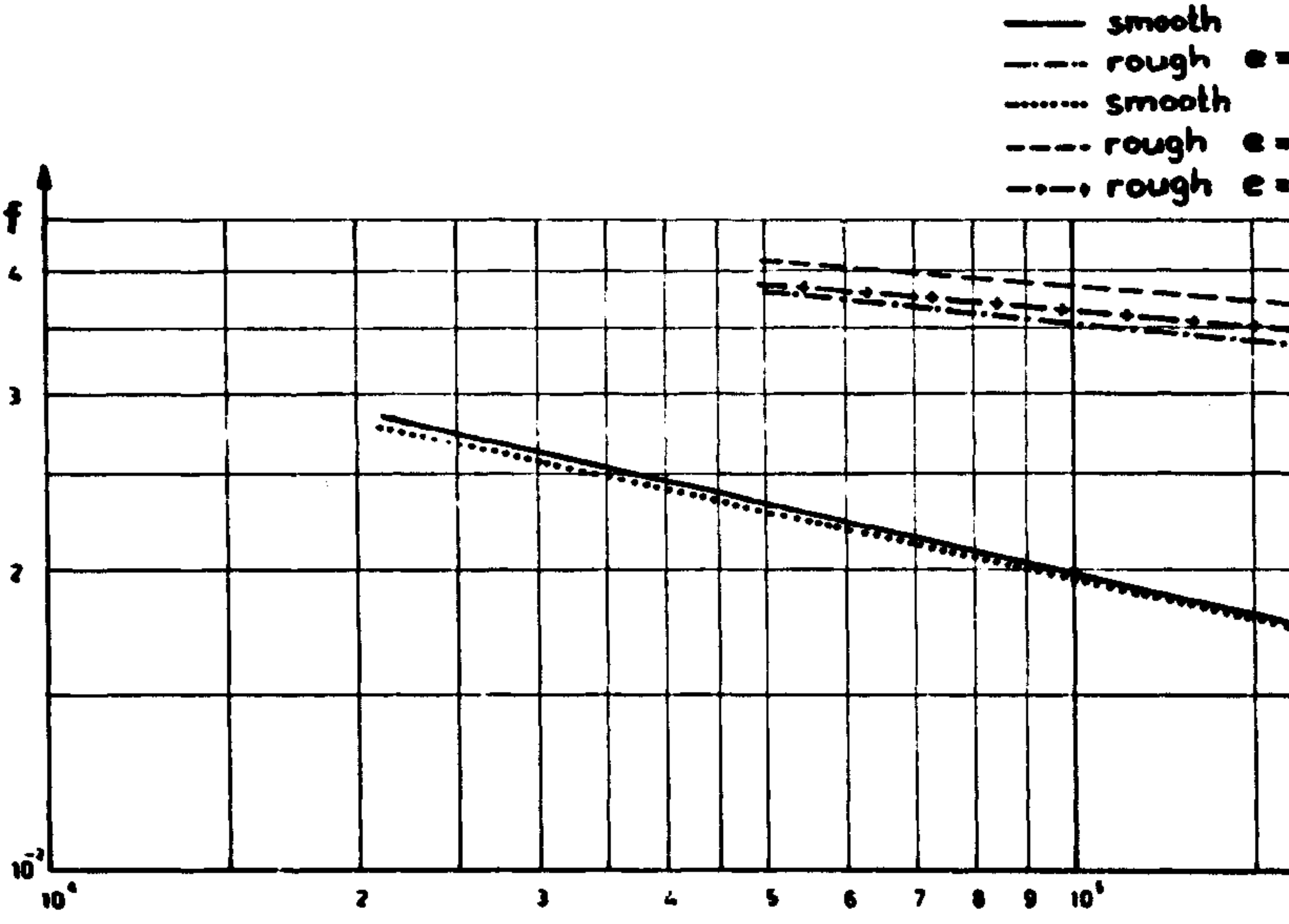


FIG. I. MEASURED FRICTION FACTORES (f_{ROUGH} corrected for T_{Wall}/T_{Bul}

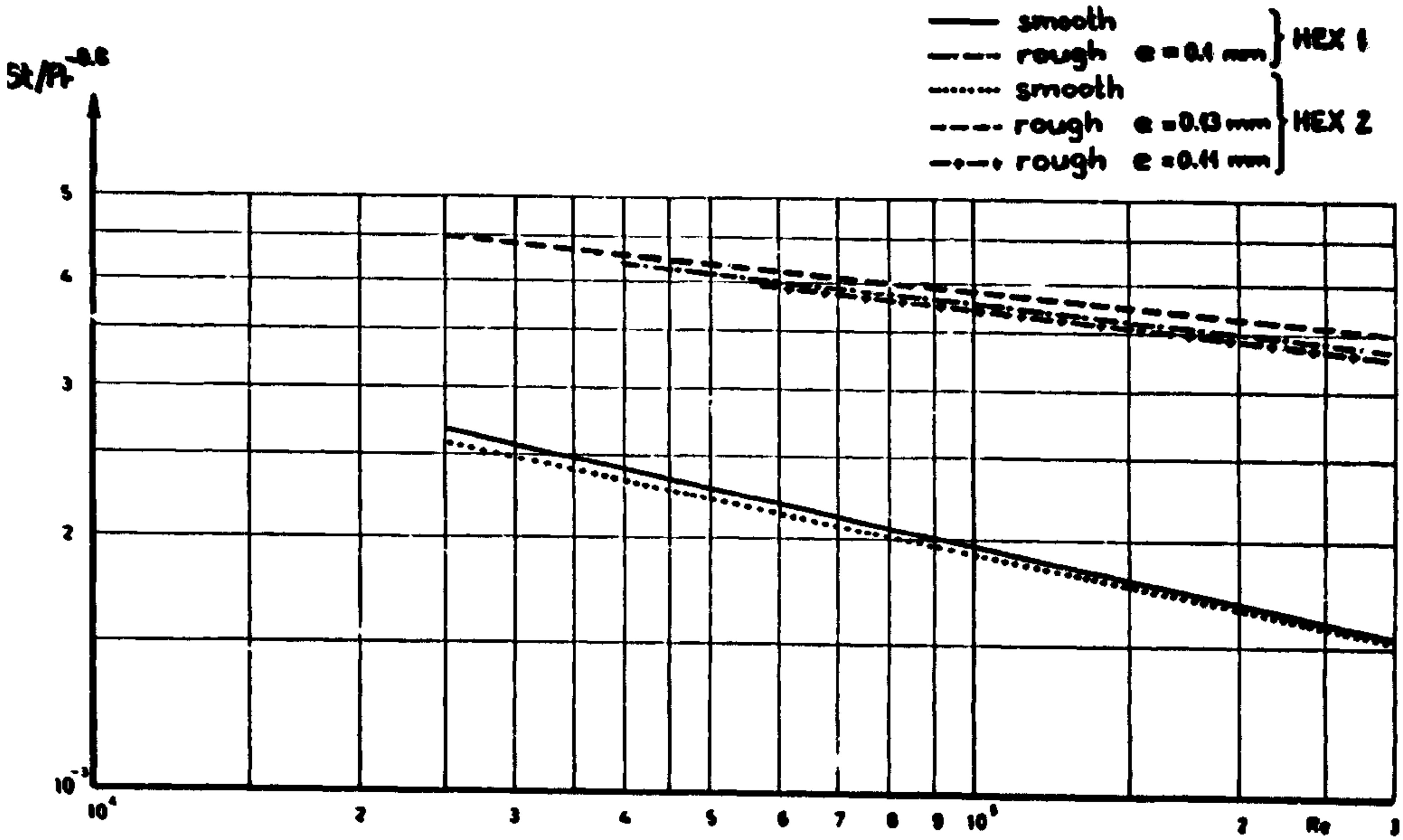


FIG. 2. MEASURED STANTON NUMBERS (St_{ROUGH} corrected for $T_{Wall} / T_{Bulk} = 1$)

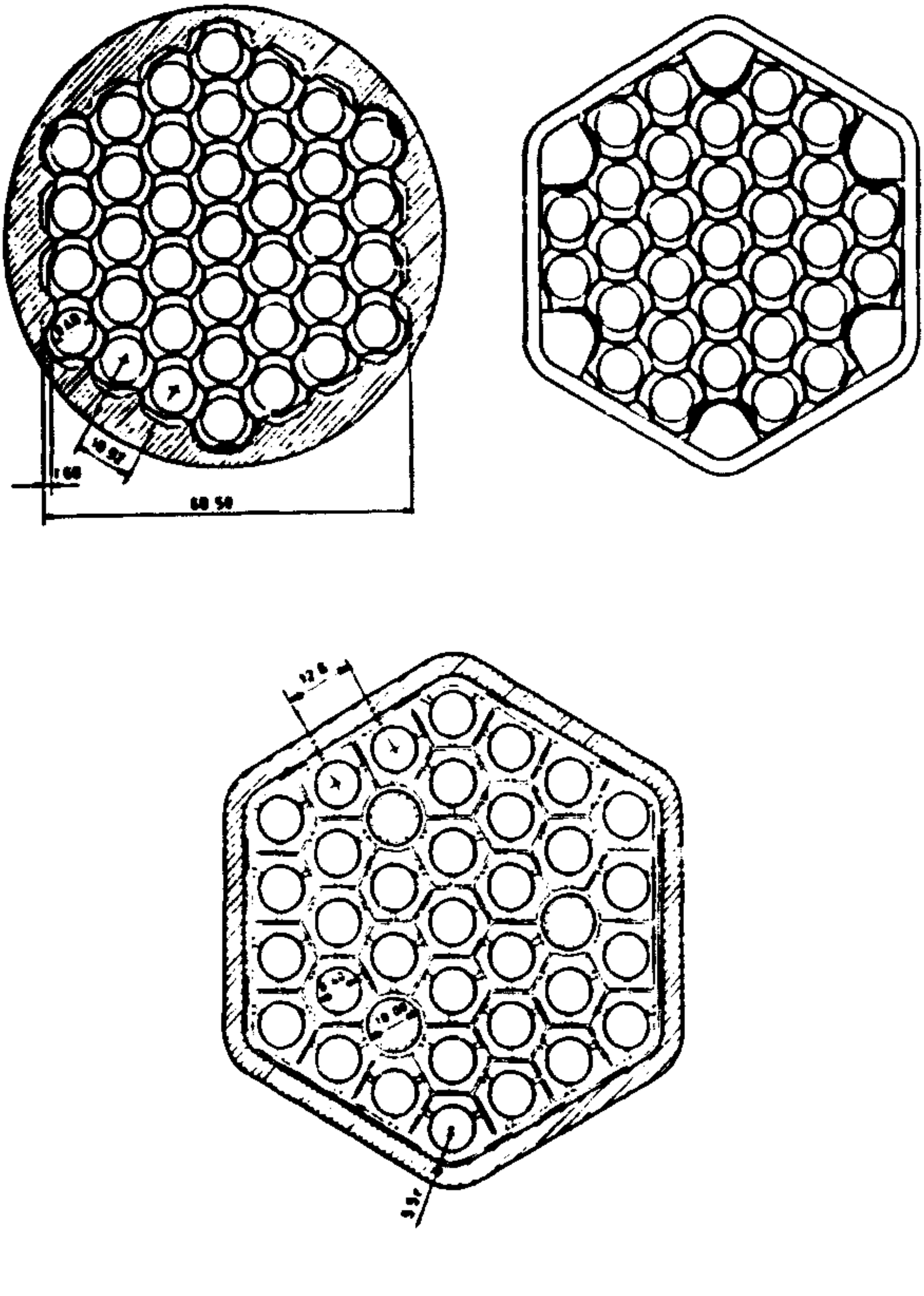


Fig.3. CROSS SECTIONS OF THE AGATHE HEX BUNDLES

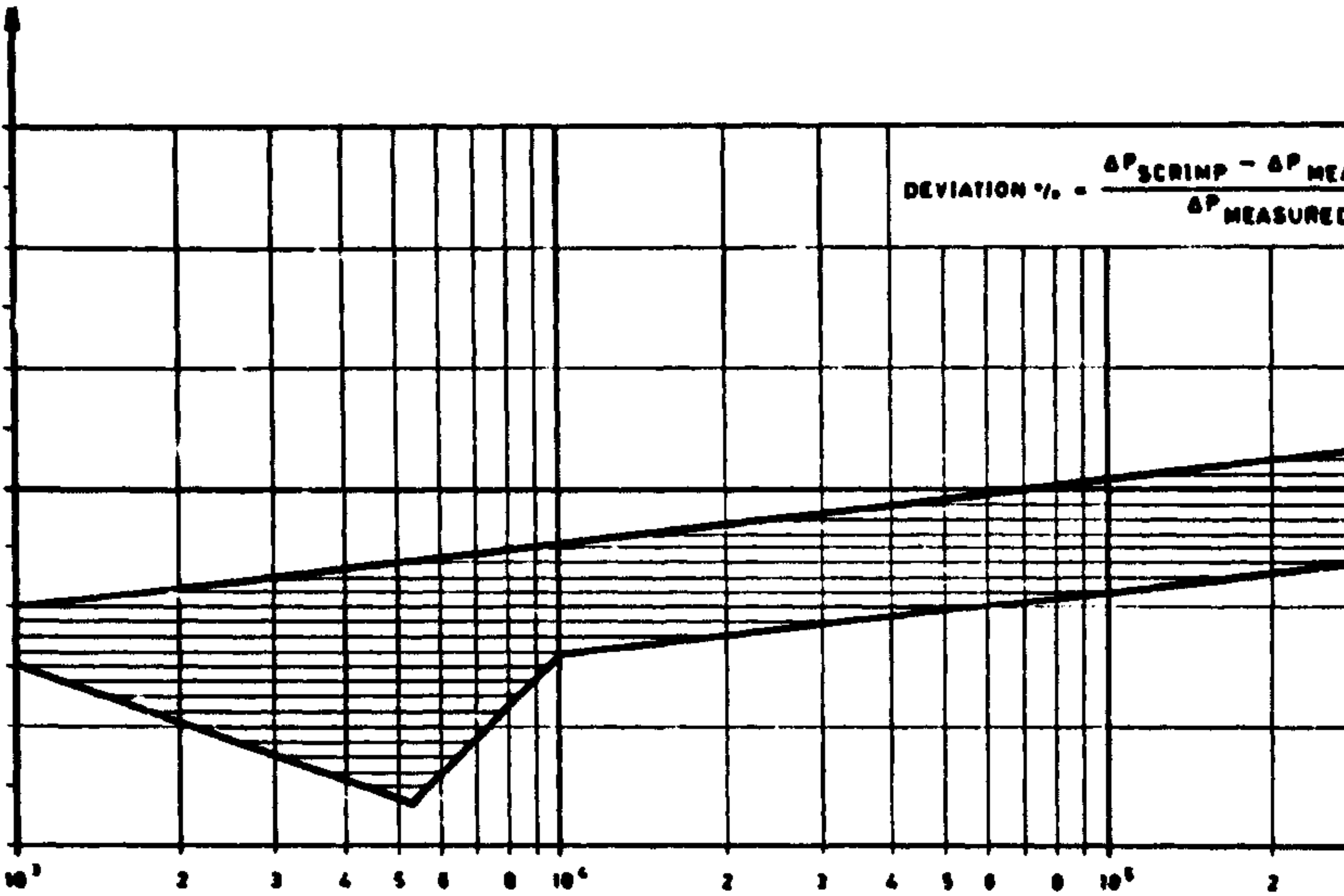


FIG. 4. QUALITY OF PREDICTIONS. RELATIVE DEVIATIONS BETWEE CALCULATED AND MEASURED BUNDLE PRESSURE DROP (UNIFORM POWER DISTRIBUTION)

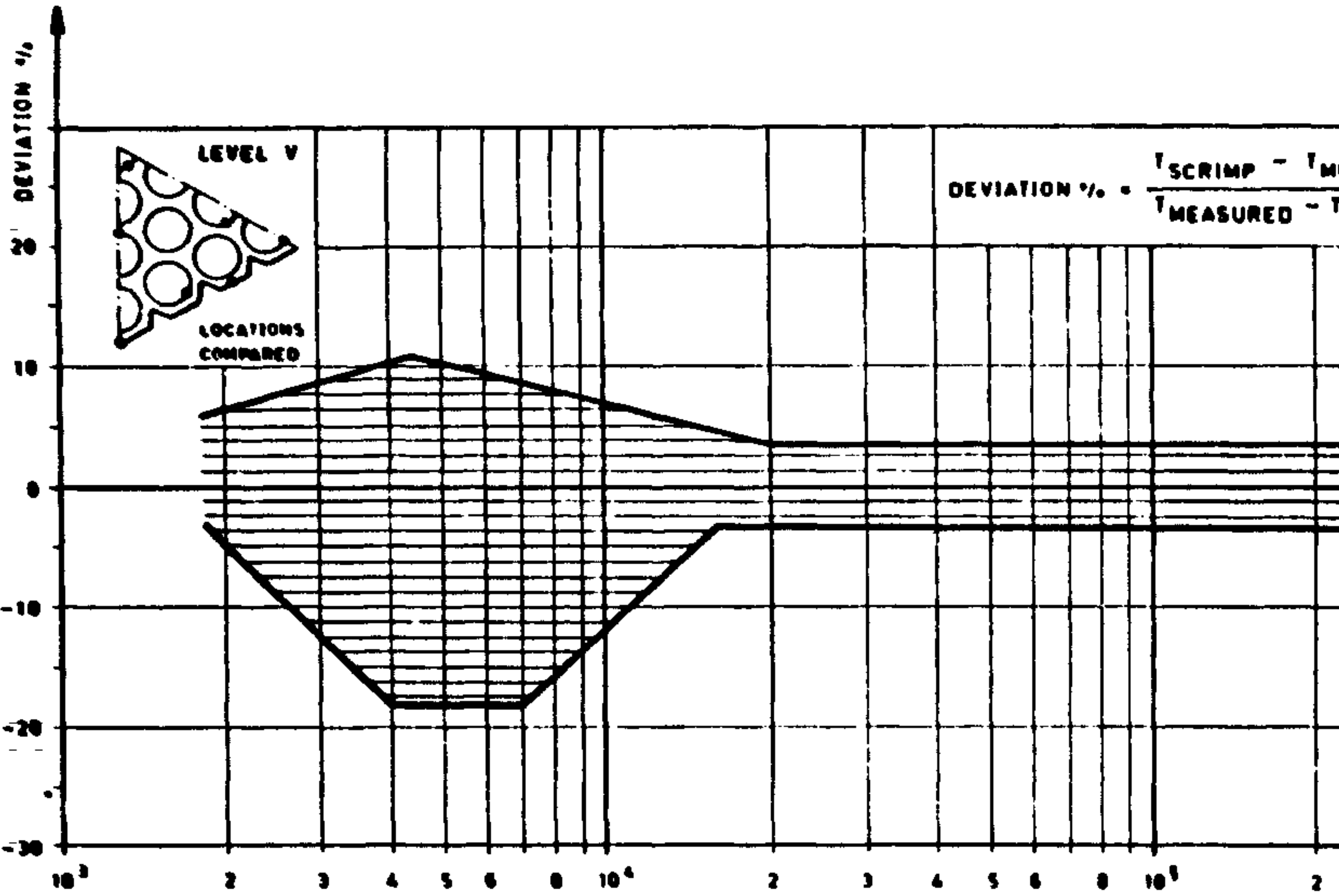
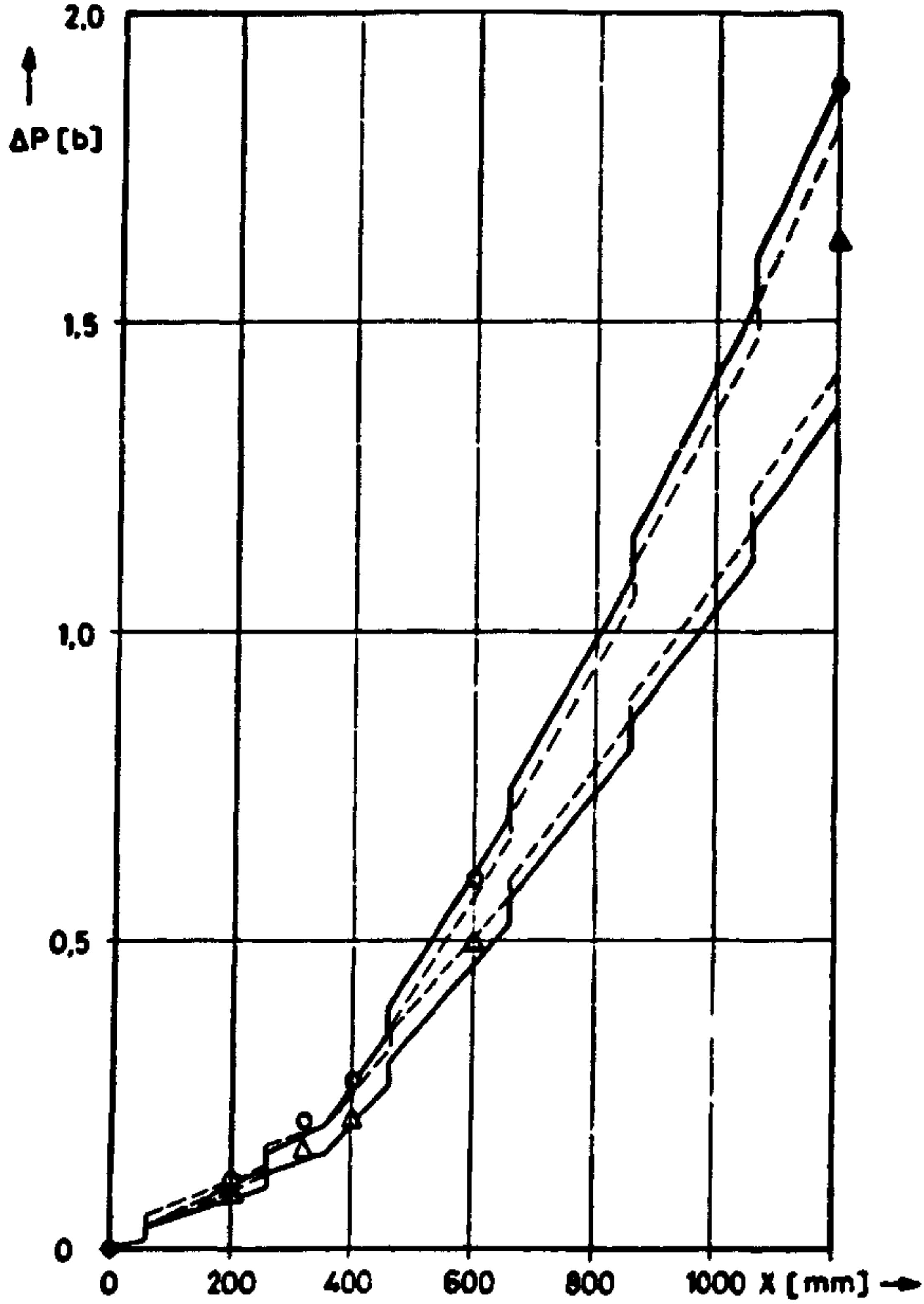


FIG. 5. QUALITY OF PREDICTIONS. RELATIVE DEVIATIONS BETWE
AND MEASURED SURFACE TEMPERATURES NEAR BUNDL
(UNIFORM POWER DISTRIBUTION)



MEASURED $Re = 3,35.10^5$

O UNIFORM HEATING

Δ POWER TILT 1,9:1

CALCULATION

—— SCRIMP

--- CLUHET

FIG. 6. COMPARISON OF PRESSURE DROP DISTRIBUTION IN 37 ROD BUNDLE

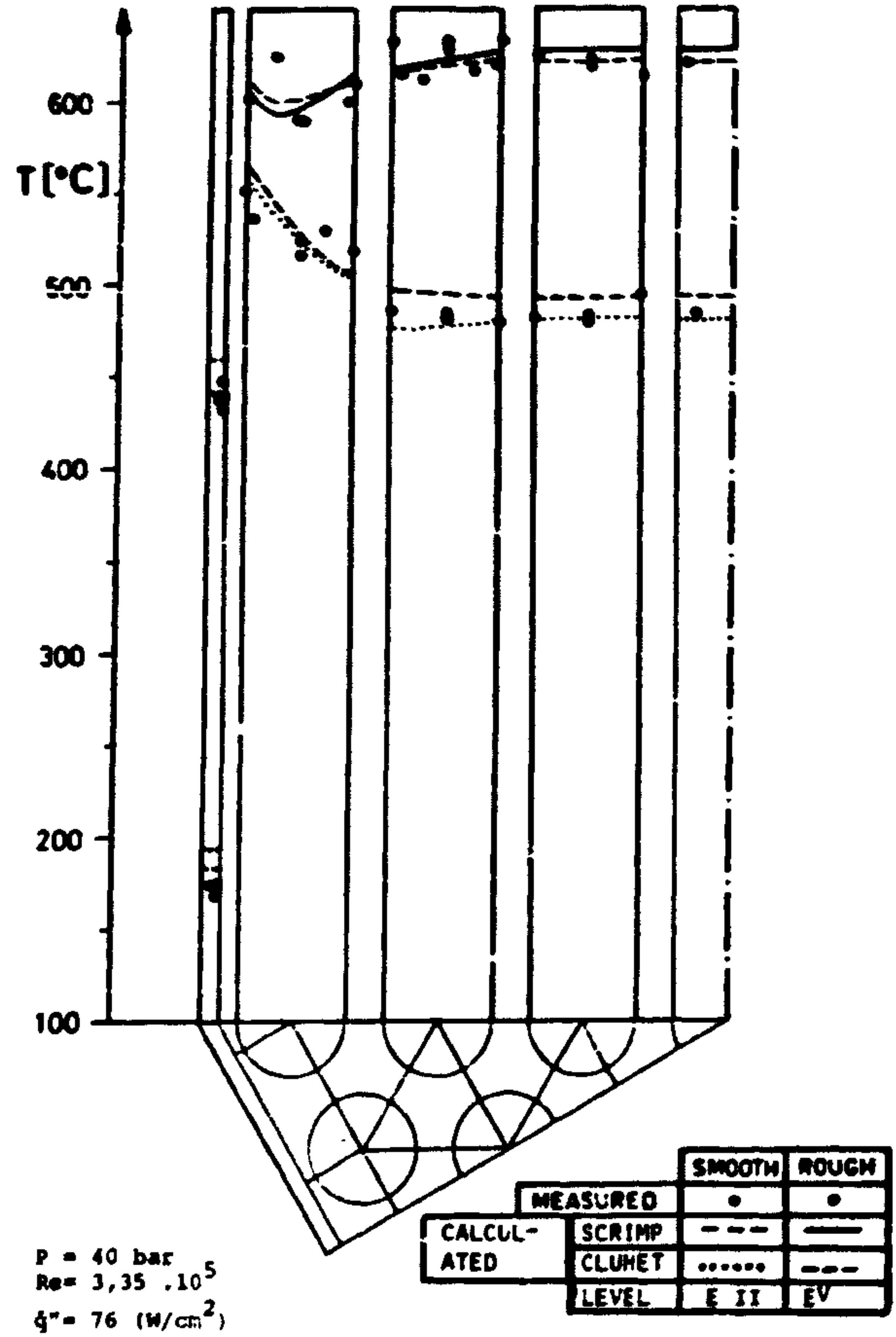
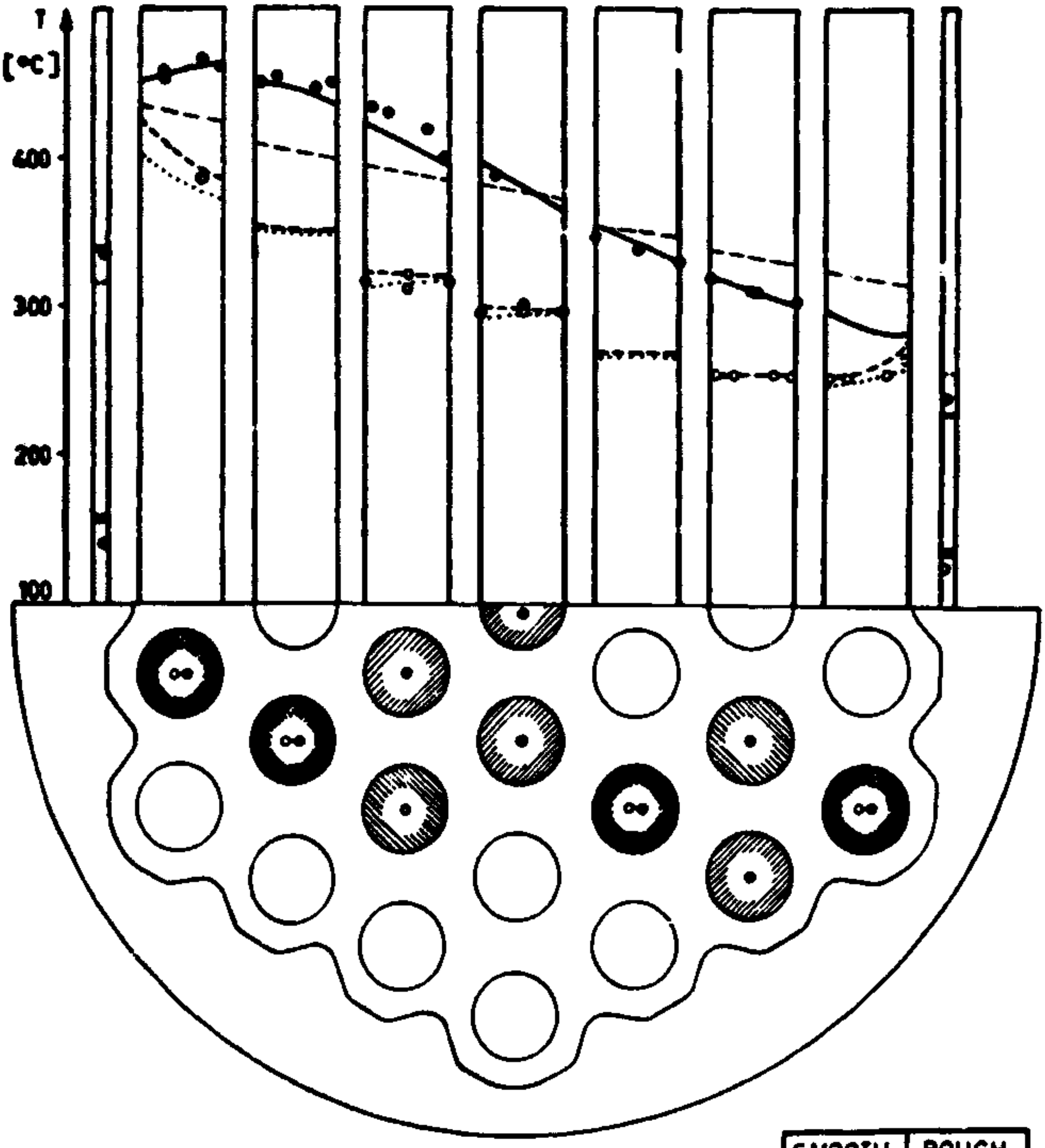


FIG. 7. COMPARISON OF TEMPERATURE DISTRIBUTION IN 37 ROD BUNDLES



P = 40 bar
Re = 3,33.10^5

		SMOOTH	ROUGH
MEASURED		•	•
CALCULATED	SCRIMP	– – –	——
	CLUHET	•••••	– – –
	LEVEL	E II	E V

FIG. 8. COMPARISON OF SURFACE TEMPERATURES FOR
37 ROD BUNDLE WITH POWER TILT OF 1.88:1.

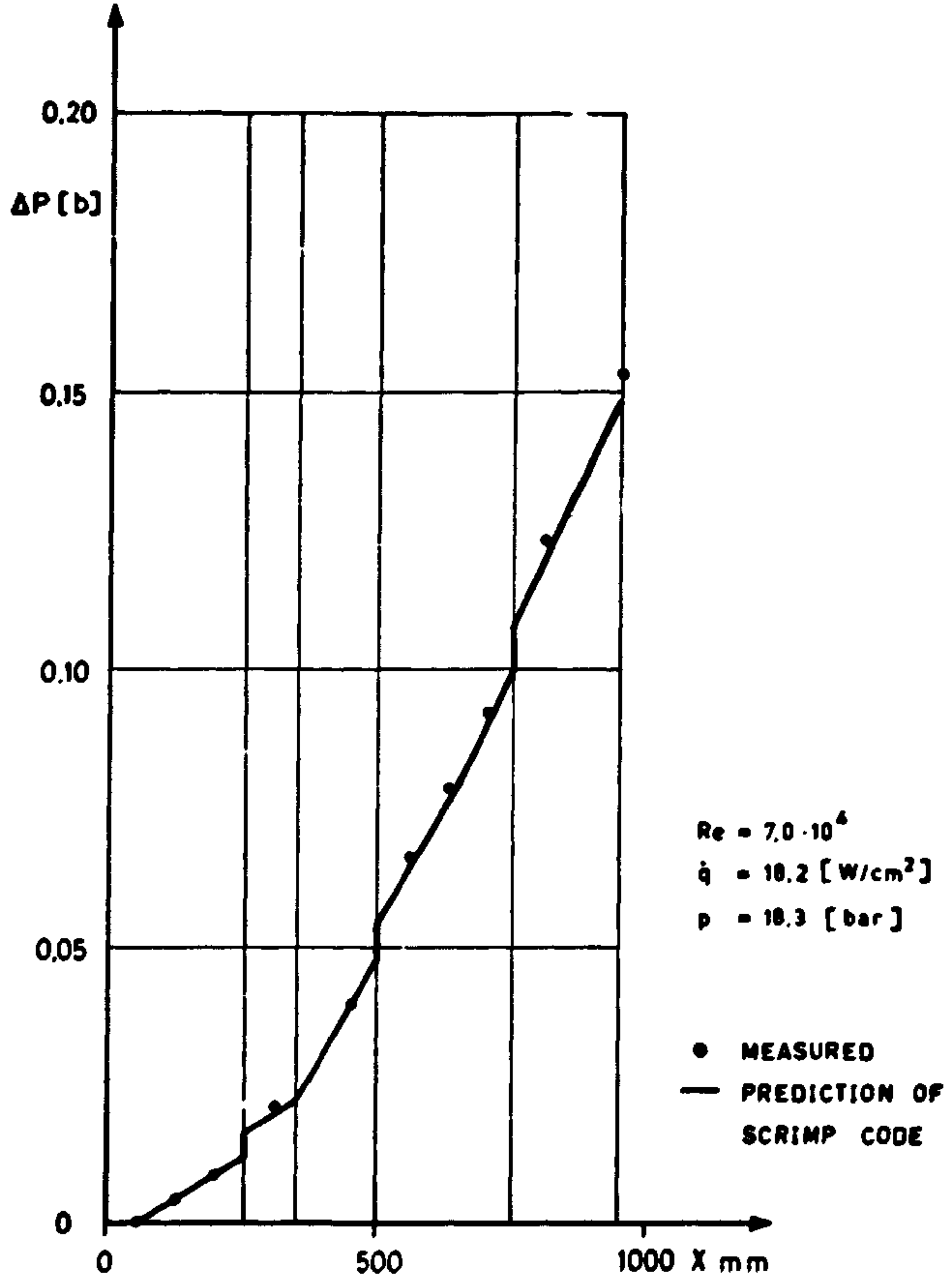


FIG. 9, PRESSURE DROP DISTRIBUTION IN 31 ROD BUNDLE

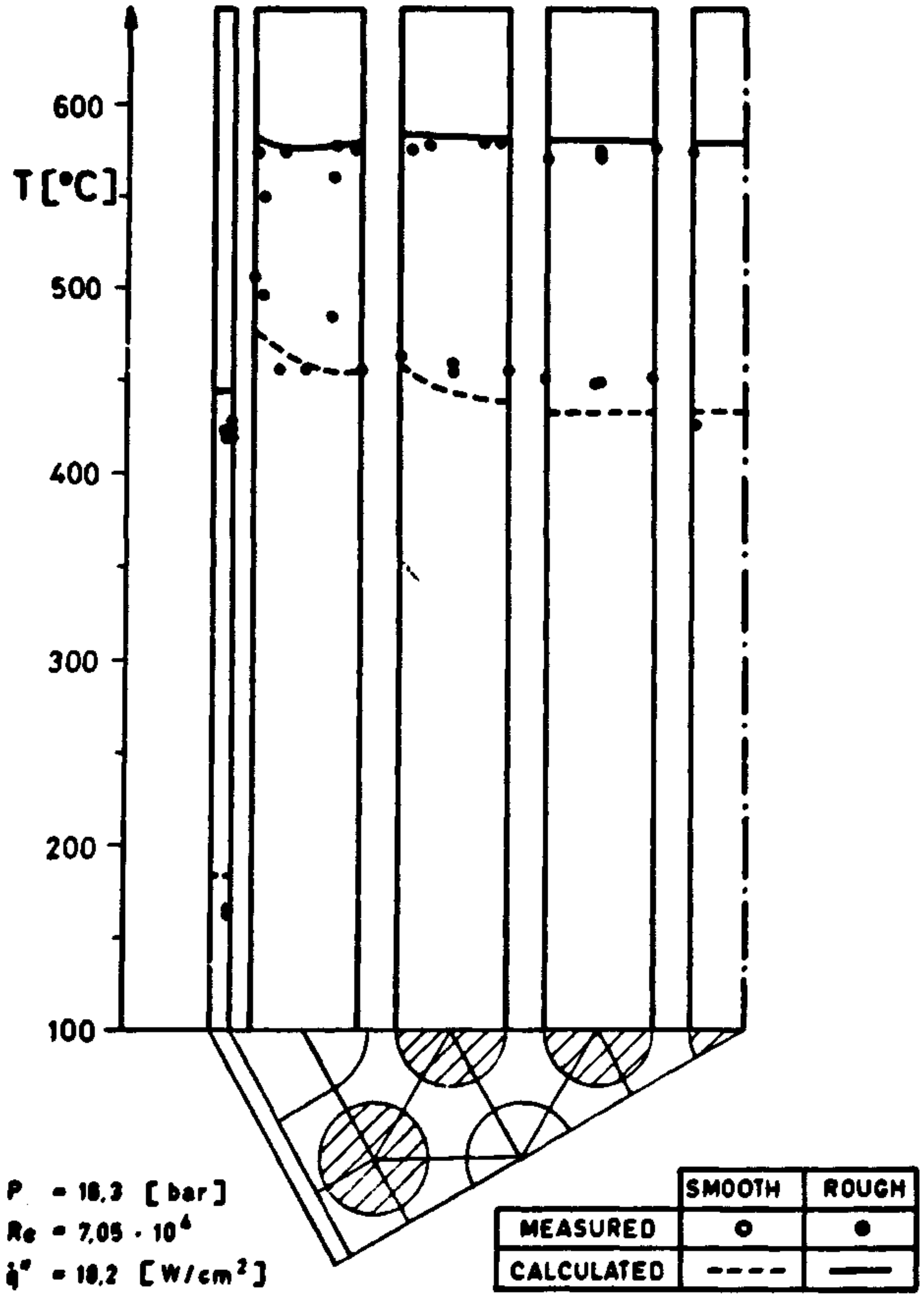


FIG. 10. COMPARISON OF TEMPERATURE DISTRIBUTION IN 31 ROD BUNDLE

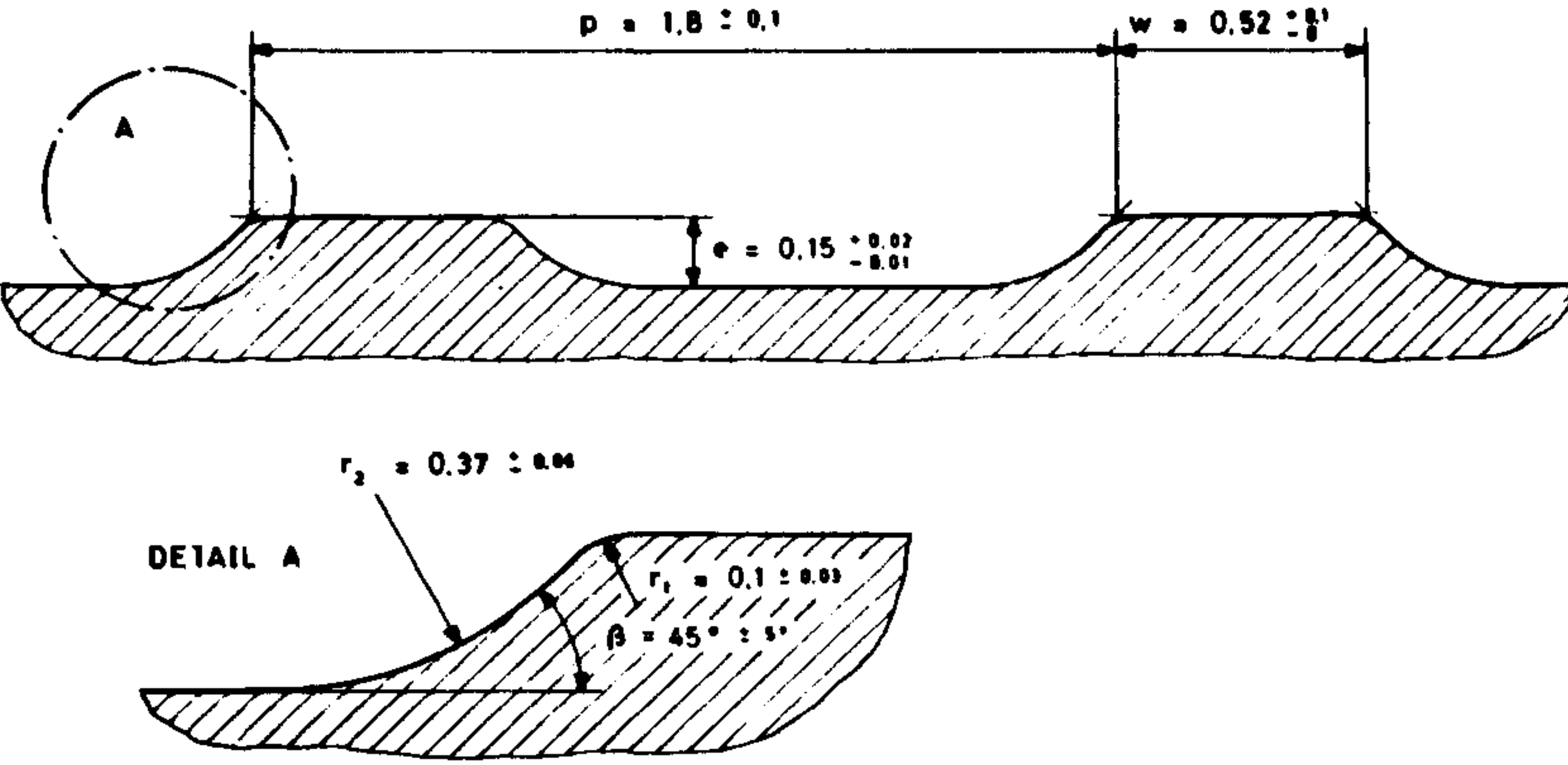


FIG. 11. ROUGHNESS FORM OF THE BUNDLE 3 RODS

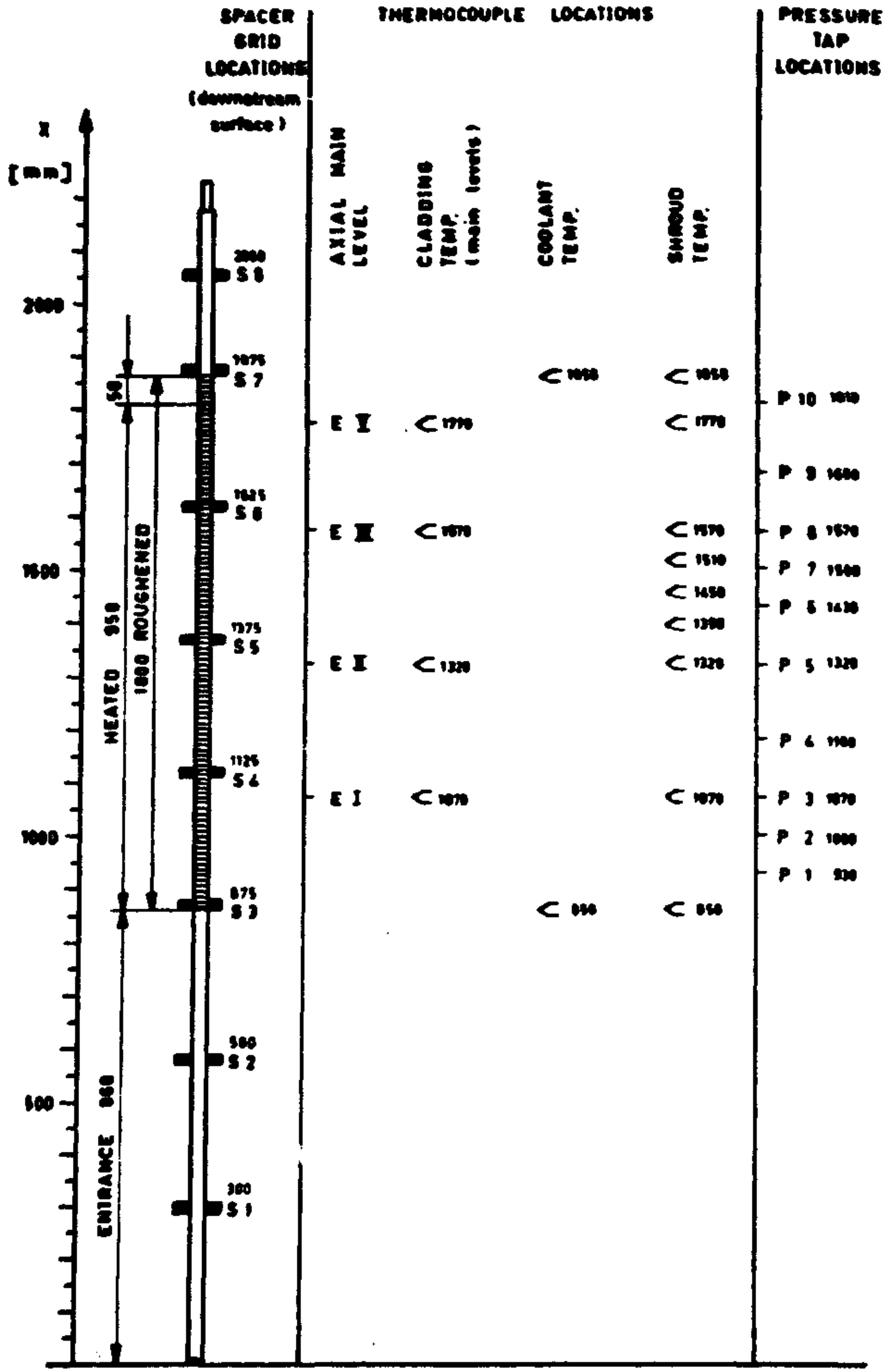


FIG. 12. AXIAL LOCATIONS OF SPACERS, THERMOCOUPLES AND PRESSURE TAPS (Bundle 3)

3. GCFR Steam Generator Research and Development Studies

3.1. Introduction

These activities were carried out at Sulzer Brothers Ltd, Winterthur within the framework of a contract between this Company and EIR. The steam generator design work is held consistent with GAC's basic concept of a GCFR plant. Principles in design and development, however, are supposed to reflect SULZER's own experiences and ideas. In this sense SULZER's activities to date under the EIR contract are to be considered a generic rather than a specific support of the GCFR steam generator development work at GAC.

While the work in CY 1978 covered a series of generic items such as heat transfer in curved tubes, surface sizing principles, vibrations in helical bundles, alternate steam generator materials, etc. the program of CY 1979 was directed towards the generation of a conceptual steam generator design and concentrated on the following subjects:

3.2. Helical Tube Bundle Performance

The steam generator bundle was investigated under three main aspects:

- finned tubes vs plain tubes: by partial or total finning of the heating surface considerable savings in size and weight can be achieved.

- different tube materials: with the temperatures prevailing in a GCFR-SG the use of I800 is not mandatory. By selecting a high chrome alloy stresses from differential thermal expansions can be reduced.

- reduced cold end temperatures and live steam pressure: by reducing core inlet temperature the circulator power and sound levels can be considerably reduced. The lower steam pressure increases pinch point temperature difference and thus reduces heating surface. It also helps to keep turbine end moisture low.

3.3. Steam Collector Design and Analysis

Alternative options of a steam collector were investigated under design and maintenance aspects and in particular with regard to their loadings under the severe GCFR transient conditions.

- "conventional" tubesheet: this is known to have high stress concentrations at the tubesheet/barrel junction. Access for inspection and plugging is good.

- subdivided tubesheet: with three small tubesheets stresses are not significantly reduced but access is difficult.

- hemispherical collector: stresses are more equally distributed. Access is possible. A restrainer is not reasonably applicable.

- subcollectors penetrating the primary boundary: this solution allows reduction of critical stresses inside the primary containment but renders tube ISI from the hot end impractical. Also tube plugging allowance is affected.

An analytical comparison between the single tubesheet and the hemispherical collector was performed. The latter was recommended for integration in the conceptual design.

3.4. Steam Generator Conceptual Design

A series of alternative designs were drafted. These concepts present different combinations of the location of main subcomponents such as:

- steam exit (top or bottom)
- feed water inlet (top or bottom)
- main flange position (top or bottom)

A further variable is the circulator position, which is either horizontal sideways the PCRV or vertical below to PCRV.

A critical review of the alternatives was made and an evaluation of their characteristics performed on the basis of a set of qualification criteria. The most promising concepts are recommended for selection of a reference design to be further elaborated. Special design areas have been worked out in more detail.

3.5. Finned Tube Technology

The proposed application of integral fin tubes requires the associated technology to be investigated. Emphasis was placed on:

- the finning method: tests show that tubes from high chrome alloy are difficult to fin unless fin geometry is modified.

- protection devices: several concepts were proposed to protect the tube against wear. Two systems were fabricated and tested. The results are satisfactory and suggest conical systems to be preferred.

- fabrication: welding, coiling, threading and inspection are reviewed with regard to finned tube particularities.

- economics: savings to be achieved by the partial employment of finned tubes are assessed.

Inspection methods were also addressed. The basic difficulties are similar to those existing for plain tubes. In general, the results from these investigations have indicated the integral fin tube to be an attractive and viable design feature.

3.6. Metallurgical Aspects

With regard to the proposed application of a 12% Chromium steel (X20CrMoV12 1) for the superheater of the GCFR steam generator some characteristics of this material were reviewed. Emphasis was placed on the decarburization within the 2V4 Cr/12Cr weld and on 12Cr manufacturing experience regarding mechanical processing, weldability and heat treatment.

The analytical, experimental and design work carried out in support of the GCFR steam generator development during CY 79 led to the proposal of several performance and design features which are considered to be worth integration into further systems layout and analysis.

A technical report on the above work has been issued in January for review by EIR and GAC. Subsequent discussions between SULZER and GAC's GCFR systems and components representatives are supposed to result in more strictly defined boundary conditions for the steam generator. Within these boundaries a more specific and more detailed design and development work shall be carried out in continuation of this program.

3.7. Reference for Chapter 3

Steam Generator for Gas-Cooled Fast Reactor. A Concertual Design Study. Technical Report 1979. Sulzer Brother Ltd.

4. Experimental Reactor Physics Program

4.1. Introduction

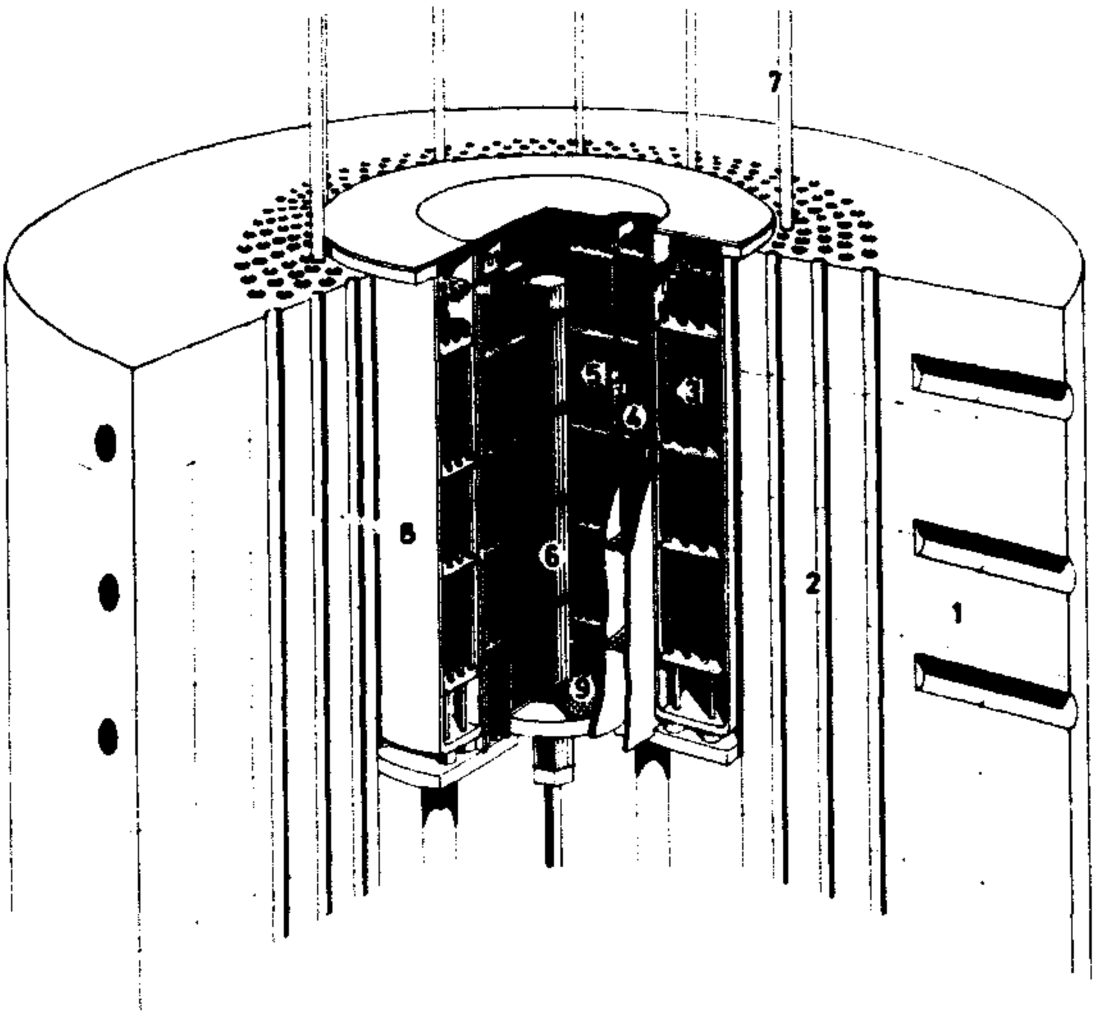
In most countries with an interest in fast reactors, extensive studies have been carried out in recent years on the comparison of classical reactor designs with heterogeneous designs containing discrete islands of fertile material. Physics advantages which have been established are the reduced sodium void effect in LMFBRs and the improved breeding capability. Of late, interest in alternate fuel cycles - prompted largely by non-proliferation considerations - has increased the importance of breeding gain improvement since such cycles normally give poorer breeding than the standard Pu-U238 cycles. The current status of non-conventional fast reactor core design has recently been described at length (1). It is clear that proper quantitative comparison of differing fuel cycles is possible only when the calculational methods and the nuclear data for each case have been equally well validated. This validation is best made against an extensive base of experimental data.

In the PROTEUS critical facility at Würenlingen, we have just completed a series of reactor physics experiments aimed explicitly at generating reliable measured data for fast systems with various blankets, and at checking the ability of our standard analysis routes to calculate the results of these experiments. The blanket geometries included in our GCFR lattice are simpler than those used in recent LMFBR experiments at Argonne (2), whereas our blanket compositions are more varied.

4.2. Experimental Program

In each experimental configuration a plutonium-fuelled core feeds neutrons into either a central radial blanket or an upper axial blanket. Three blanket compositions have been investigated. Firstly we employed a conventional depleted UO_2 blanket. Next ThO_2 fertile blankets were employed, thus modelling a plutonium - transmuter system of the type which could be conceivably used as a producer of U233 for alternate fuel cycle reactors. Finally fertile blankets of densely packed thorium metal were employed. These blankets, which may have practical applications because of their suitable neutronic and materials behaviour, provided a more stringent test of the methods and data to be validated.

The PROTEUS reactor (3) in which the experiments were carried out is a mixed fast-thermal assembly with a pin fuelled central fast lattice, as illustrated in Figure 1. The locations and sizes of both central and axial blankets are indicated schematically in Figure 2 (they were in practice successively, rather than simultaneously installed). The core zone consists of PuO_2/UO_2 pellets in 7 mm diameter fuel rods on a 10 mm hexagonal pitch. The depleted UO_2 and ThO_2 blankets had the same geometry; the former consisted of suitable sintered pellets and the latter of loosely filled sintered particles. The homogeneous density in the ThO_2 zones was thus low ($\sim$ 6 g/m^3). The metal blankets, on the other hand consisted of 19 mm rods on a close 21 mm pitch so that the thorium density was increased to 11.3 g/cm^3. The boron plastic layers indicated in Figure 2 served to reduce the penetration of low energy neutrons from the thermal driver zones into blanket zones.



1 GRAPHITE REFLECTOR
2 GRAPHITE DRIVER
3 D_2O - DRIVER
4 BUFFER (U-METAL)
5 FAST ZONE (PuO_2/UO_2)
6 TEST COLUMN
7 CONTROL ROD

Fig. 1: The PROTEUS fast-thermal critical assembly.

In all cores measurements were made of important integral reaction rate ratios at the centre of the reactor and of reaction rate profiles through core and blanket zones. In some cores dif-

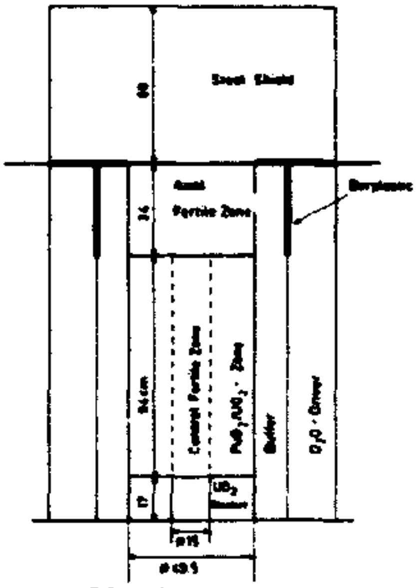


Fig. 2: Blanket zone configurations

ferential neutron spectrum measurements were also performed using gasfilled proton recoil counters. The reaction rates monitored were capture in U238 and Th232 (the principal breeding reactions), fission in Pu239 (the main neutron source) and threshold fission in U238 and Th232 (important for blanket power production and for sodium voiding effects). These reactions are denoted respectively as C8, C2, F9, F8, F2. In addition the (n,2n) reaction in Th232 was measured as this is an important source for build up of U232 which can complicate fuel handiing in Th232-U233 cycles. The experimental programm in PROTEUS is now completed; analysis of the later stages is still in progress.

4.3. Methods and Data

Most of our analyses to date have been based upon use of the UK collision probability code MURAL8, which employs an adjusted data library, FGL5, in 2240 fine groups. It is, however, important to note that the Th232 and U233 data included in our version of FGL5 are in simpler form (37 broad groups plus shielding factors) and were, in fact, derived from the ENDF/B-4 based broad group library LIB-IV (4). Calculations have also been made with the GGC4 code using nuclear data from ENDF/B-4. Currently in progress are investigations using the updated thorium data proposed for ENDF/B-5. Cell calculations of varying complexity were employed to produce broad group cross-sections for use in full reactor calculations. The central blanket configurations were normally modelled in a 1-D cylindrical transport theory code. The axial blanket configurations were calculated using 2-D diffusion or transport theories. Normally the 1-D calculations had 26 energy groups and the 2-D runs used P1-corrected 10 group cross-sections.

4.4. Results and Discussion

In Table 1, we give a summary of the important measured and calculated reaction rate ratios at the centres of the three central blanket cores. Of most interest are the breeding reactions C8 and C2. The ratio of C8/F9 is well predicted in all cases by the FGL data used; ENDF/B data is known to overpredict this ratio (5). The ratio of C2/F9 is underpredicted slightly in the infinitely dilute case of the normal PROTEUS lattice, but is overpredicted in thorium-containing zones. This is believed to be partly due to inadequate calculation of the self shielding for Th232 capture (5) although Argonne workers (2) using different methods also obtained calculated values a few percent higher than experiment. Calculations with new ENDF/B-5 thorium data are in progress and it will be of great interest to see the impact of these on the important calculated value of C2/F9.

Table 1: Comparison of calculated (C) and Experimental (E) Reaction Rate Ratios in the Centre of the PROTEUS reactor

Reaction Rate Ratios	Normal PuO_2/UO_2-Lattice		Heterogeneous Lattice with Central UO_2-Zone		Heterogeneous Lattice with Central ThO_2-Zone		Heterogeneous Lattice Central Th-Metal-Z
	Experimental values	C/E	Experimental values	C/E	Experimental values	C/E	Experimental values
$\frac{\sigma_c\ (U238)}{\sigma_f\ (Pu239)}$	0.1334 ± 1.1%	0.978	0.1356 ± 1.8%	1.005	0.1773 ± 2%	0.994	0.1574 ± 2%
$\frac{\sigma_f\ (U238)}{\sigma_f\ (Pu239)}$	$3.042 \cdot 10^{-2}$ ± 1.3%	1.069	$2.075 \cdot 10^{-2}$ ± 1.2%	1.041	$2.235 \cdot 10^{-2}$ ± 2%	1.040	$1.825 \cdot 10^{-2}$ ± 2%
$\frac{\sigma_f\ (U235)}{\sigma_f\ (Pu239)}$	1.017 ± 1.5%	0.986	1.043 ± 1.2%	0.9[illegible]7	–	–	–
$\frac{\sigma_c\ (Th232)}{\sigma_f\ (Pu239)}$	0.2000 ± 1.3%	0.972	–	–	0.1663 ± 1.7%	1.057	0.1492 ± 1.5%
$\frac{\sigma_f\ (Th232)}{\sigma_f\ (Pu239)}$	$8.061 \cdot 10^{-3}$ ± 2%	0.888	–	–	$5.645 \cdot 10^{-3}$ ± 2%	0.902	$4.700 \cdot 10^{-3}$ ± 2%
$\frac{\sigma_f\ (U233)}{\sigma_f\ (Pu239)}$	1.519 ± 1.3%	0.990	–	–	1.577 ± 1.5%	1.001	1.521 ± 1.5%
$\frac{\sigma_{2n}\ (Th232)}{\sigma_c\ (Th232)}$	$6.84 \cdot 10^{-3}$ ± 2.5%	1.019	–	–	$4.98 \cdot 10^{-3}$ ± 5.2%	1.003	$5.65 \cdot 10^{-3}$ ± 2.5%

Of the threshold fission reactions F8/F9 is significantly overpredicted, as is normally the case with FGL data, and F2/F9 is consistently underpredicted. The marked increase in C/E for both these threshold reactions in the metal blanket relative to values in the other blankets suggests that the fluxes around 1 MeV are overpredicted in the metal case. The poorer prediction of F2 than of F8 may be acceptable since the absolute magnitude of the former cross-section is much smaller and the contribution to power production correspondingly lower. The ratios F5/F9 were used mainly as spectral indices as they do not occur to any significant degree in the lattices. The C/E values do not change much from core to core, reflecting the similar energy dependence of numerator and denominator in the reaction rate ratio, and the values are close to 1·0 indicating that the infinitely dilute cross-sections in the library are adequate.

The reaction rate ratio comparisons described thus far are supplemented in Figure 3 by a display of the measured and predicted spatial dependence of reaction rates in the same cores. Only the important source (F9) and sink (C2, C8) reactions are illustrated. In all central blanket cores the radial distribution of Pu239 fission is well predicted, but the resonance capture reactions are harder to calculate accurately. The standard method of obtaining blanket cross-sections by a cell calculation which takes no account of the adjacent core zone does not suffice for resonance absorbers which appear on only one side of the interface. The infinitely dilute cross-sections obtained for the zone with no resonance absorber will be higher than the true effective cross-section. Accordingly, in the thorium cores, significant improvements were achieved by deriving broad group cross-sections for zones near the interface from multi-region collision probability cell calculations employing models which extended through the core/blanket interface. The comparison of these calculations

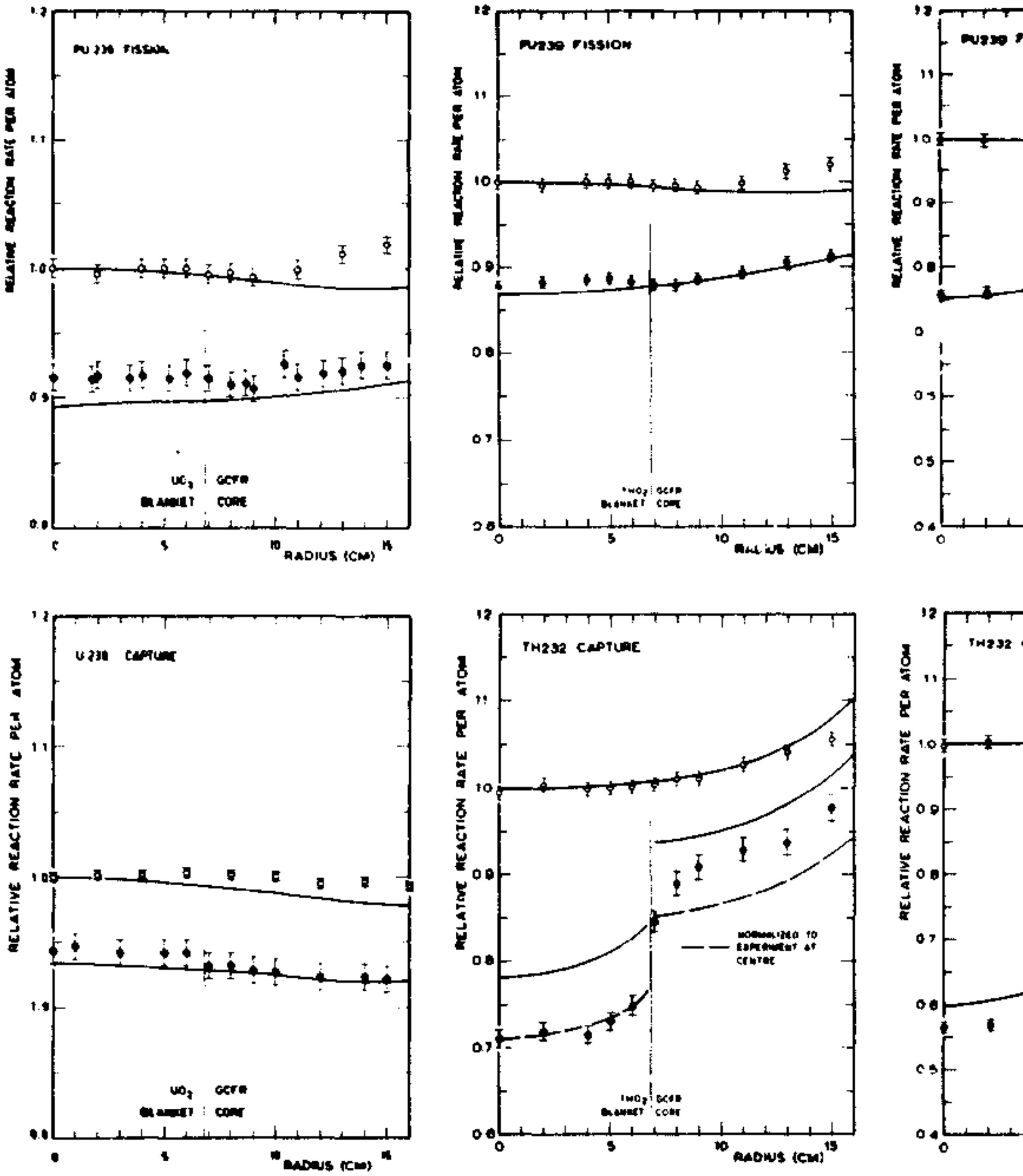


Fig. 3:

Radial variation of reaction rates per atom in lattices with central blanket zones. The central values for the standard lattice without column were normalized to unity. In both experiment and calculation, the thermal flux in the reactor driver zone was held constant on introduction of the central blanket.

with the simpler isolated zone cases is exemplified in Figure 4; here SIGØ and PHIØ are the cross-sections and resulting fluxes from reactor calculations starting with isolated cell models and

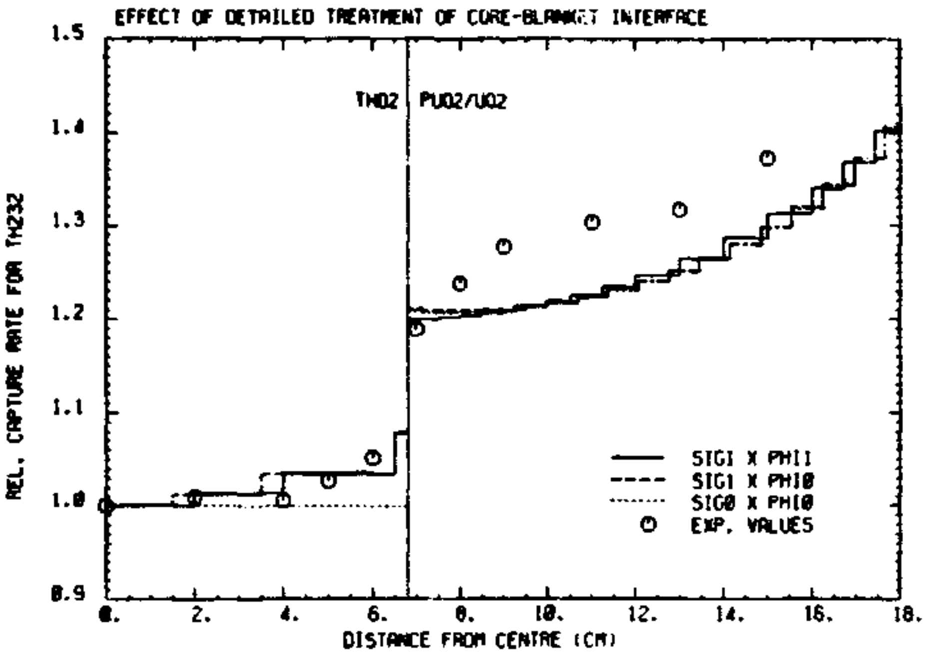


Fig. 4: Influence of cell modelling on calculation of thorium capture reaction rate profile in core with central ThO_2 blanket.

SIG1, PHI1, the corresponding values from the more detailed modelling.

The configurations so far described allow some assessment of the adequacy of data and methods for designing cores with relatively small fertile islands inside fuel zones. The axial blanket assemblies in PROTEUS correspond more to conventional Pu-U238 breeders, or to Pu-driven, U233-producing transmuter cores. Figure 5 illustrates the axial profiles of reaction rates through core and blanket zones of the assemblies.

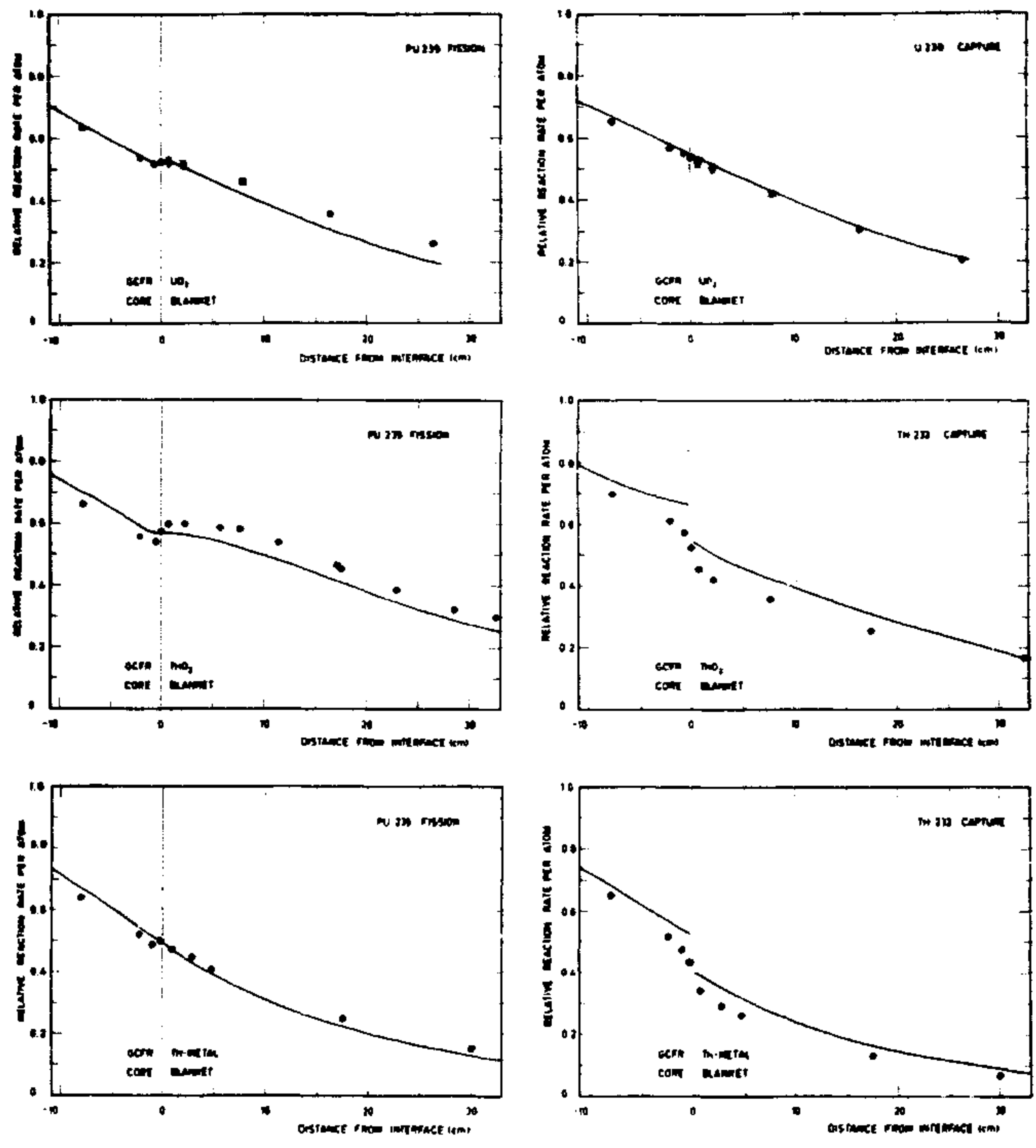


Fig. 5: Spatial variation of reaction rates per atom in lattices with axial blanket zones. The calculated values are normalized to experiment at the centre of the fast zone.

In each case the Pu239 fission reaction is well predicted in the core but underpredicted by varying amounts in the blankets. Transport calculations, in place of the diffusion runs which produced the displayed curves, do not greatly improve the underprediction. The U238 capture profile is well predicted in the UO_2 blanket core, whereas the Th232 capture rate is overpredicted in thorium zones relative to adjacent fuel zones. This again points to inadequate shielding data for thorium in our FGL library. Transport theory calculations showed little change for these reasonance reactions; their effect on threshold reaction profiles was clearer, as is illustrated for the case of Th232 fission in Figure 6.

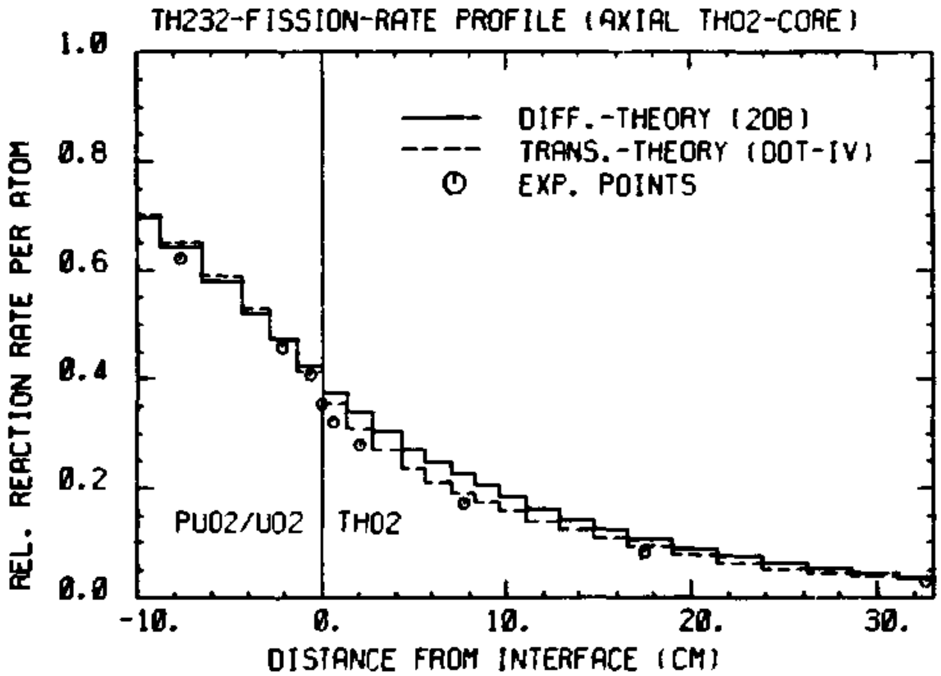


Figure 6: Comparison of transport and diffusion theory calculations of spatial variation of Th232 fission rate in the axial ThO_2 blanket core.

4.5. Conclusions

The broad conclusion of the experiments and analysis summarized above is that uranium blankets are significantly better calculated by our data and methods than are thorium blankets. The dense, extremely non-reactive thorium blankets present problems for conventional core calculational methods. The thorium predictions are, however, better than one might expect from the generous uncertainties normally assigned to ENDF/B-4 thorium data (6). The apparent deficiences in resonance shielding data in the FGL-based approach may be improved in the latest version of the library in which thorium is represented in fine group form. The possible deficiences in data and methods which are mentioned above do not affect the calculated values to an extent which can cast major doubts on current calculations of reactor systems, with heterogeneous cores.

4.6. References to Chapter 4

(1) Nuclear Technology, Vol. 44, (June 1979) I-82.

(2) LINEBERRY, M.J., et al., Physics Studies of a Heterogeneous Liquid-Metal Fast Breeder Reactor, Nucl. Tech. 44 (1979) 21.

(3) Seth S., et al., "GCFR Benchmarks: Experiments and Analysis", Nuclear Cross Sections and Technology (Proc. Int. Conf. Washington, D.C., 1975)

(4) KIDMAN, R.B., MACFARLANE, R.E., "LIB-IV, A Library of Group Constant for Nuclear Reactor Calculations", LA-6260-MS (March 1976)

(5) SCHMOCKER, U., "Benchmark Experiments with Thorium in Fast Reactor Lattices", Neutron Physics and Nuclear Data (Proc. Int. Conf. Harwell, UK, 1978) 999.

(6) TURSKI R.B., MCKNIGHT R.D., "Nuclear Data Sensitivity Coefficients for a (U233-Th232) Fuelled LMFBR", Advances in Reactor Physics (Proc. Topical Meeting Gatlinburg, USA, 1978) 455.

5. Nuclear Performance and Safety Studies

5.1. Steam Entry Reactivity Effect

To obtain additional insight into the physical processes caused by the ingress of steam into the core and blankets of a GCFR the earlier analytical work in this field was continued including perturbation theory regional break-downs and a study of the sensitivity of the steam entry reactivity effect to the composition of the fuel.

The calculations were performed for a model of the GCFR with 4 core zones and radial and axial blanket zones which confirm exactly to the General Atomic 300 MWe design (1). The effect of steam ingress was investigated both for the fresh reactor and for conditions corresponding to the middle of an equilibrium cycle. The nuclide concentrations for the latter were obtained from a two-dimensional burnup calculation in 10 energy groups using group constants derived from the British data set FGL5. The group condensation scheme allowed for interactions between zones by averaging the fine group cross sections with zone integrated fluxes from a one-dimensional radial reactor calculation. The plutonium enrichment of the fuel was adjusted to give $K_{eff}=1$ at the end of an equilibrium cycle.

Using the nuclide concentrations obtained from the burnup calculations two sets of 37-group constants were generated for the dry reactor and a steam density of 0.03 g/cm³ in the coolant channels. Group constants for intermediate steam densities were obtained by appropiately interpolating between the two sets. These group constants were then fed into 37-group whole reactor calculations in R-Z geometry which provided K_{eff} values and normal and adjoint fluxes for subsequent exact perturbation theory calculations. The whole reactor calculations included control rod homogenizations in Core Zones 1 and 3, the control absorbers being adjusted to obtain K_{eff} = 1 for the dry reactor, i.e.

the reactor was assumed to be critical when the steam ingress occured.

Results of the study are shown in Fig. 1 and Tables 1 and 2. As can be seen in Fig. 1 the total reactivity effect of the steam was computed to be negative in the whole range of interest (The maximum steam density of 0.03 g/cm³ corresponds to the H_2O inventory of three steam generators). A similar behaviour was found for reactor conditions corresponding to the middle of an equilibrium cycle (MOEC) and the beginning of life (BOL), provided that the plutonium inventory of the fresh core was reduced to obtain criticality at the end of the first reactor cycle. Fig. 1 also illustrates the importance of correctly compensating the excess reactivity of the core. A reactivity adjustment by means of control rods, for instance, may not be simulated by a buckling imposed overall leakage. The figure holds for a fuel temperature of 300°K, which is known to be a conservative case (At normal operating temperature the steam entry reactivity effect would be more negative). Fuel pin heterogeneity is included but streaming effects have been neglected since they are small and not relevant in comparison with the other effects addressed in this study.

Table 1 shows the effect of steam entry into different reactor zones and gives a regional perturbation theory breakdown of the total effect (contribution of each zone to the total effect). Although these results are not of great practical importance (The helium loops of the reactor are layed out such that differential steam ingress is an unlikely event), they contribute to the physical understanding of the effects. For instance, the following observations can be made:

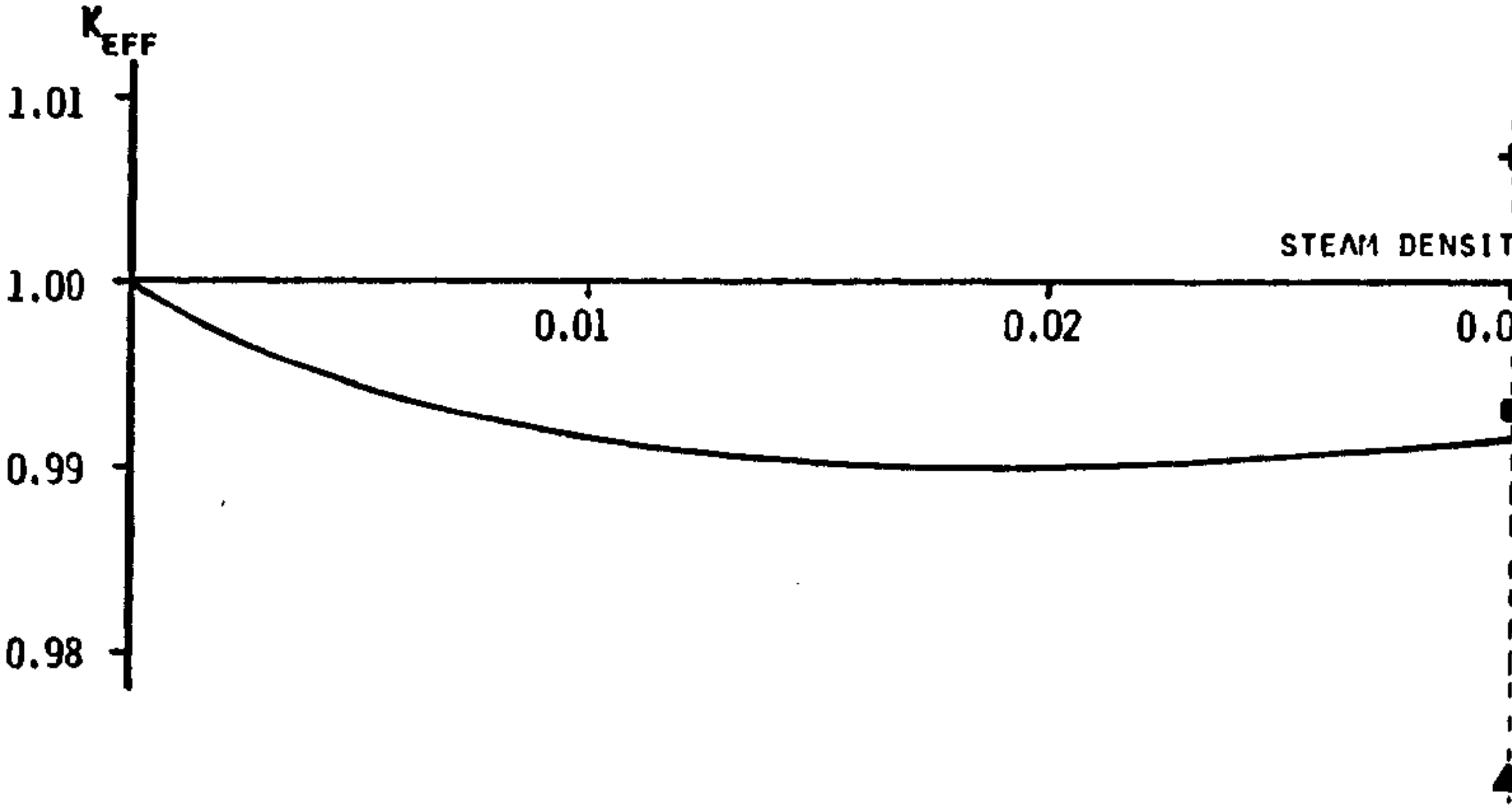
1. The blankets contribute considerably (72%) to the total (negative) effect.

2. Core zones with a high plutonium enrichment and no control absorbers give rise to positive contributions.

3. In the core positive and negative zonal contributions are compensating each other to a considerable extent.

4. The spectral interaction between wet and dry zones is such that the effect of steam ingress into a zone is slightly more negative than the contribution of this zone to the total effect.

The sensitivity of the steam entry reactivity effect to the composition of the fuel can be studied using one-group cross-sections. The representation in Table 2 is convenient for estimating the influence of small changes in the nuclide concentrations on ΔK_{eff} ("wet"-"dry"). It can be seen that changes in the isotopic composition of the plutonium can give rise to important effects. The replacement of pure ^{239}Pu by an equivalent amount of typical LWR grade plutonium, for instance, can be expected to give a positive net effect, the negative ^{240}Pu contribution being overbalanced by the positive ^{241}Pu contribution. The beneficial influence of ^{237}Np is also apparent. Results of an investigation of the influence of this nuclide on the steam entry reactivity effect were reported last year. It may be added that in the hard neutron spectrum of a GCFR an appreciable fraction of the ^{237}Np is fissioned. Therefore this nuclide, unlike normal absorbers, can be used to adjust the steam entry reactivity effect without noticeably deteriorating the breeding ratio.



— MOEC, EXCESS REACTIVITY (1.9%) COMPENSATED BY B-10 CONTROL RODS

\+ MOEC, EXCESS REACTIVITY (1.9%) COMPENSATED BY ADDITIONAL LEAKAGE

■ BOL, EXCESS REACTIVITY COMPENSATED BY B-10 CONTROL RODS AND REDUCED PU ENRI

▲ BOL, EXCESS REACTIVITY COMPENSATED BY B-10

FIG. 1: REACTIVITY EFFECT OF STEAM ENTRY IN A 300 MWE GCFR

TABLE 1: REGIONAL DEPENDENCE OF STEAM ENTRY EFFECT

REACTOR ZONE	Δρ (%)	
	EFFECT OF STEAM ENTRY INTO THE ZONE	CONTRIBUTION OF ZONE TO TOTAL EFFECT
CORE ZONE 1	- 0.27	- 0.19
CORE ZONE 2	- 0.07	- 0.06
CORE ZONE 3	- 0.46	- 0.30
CORE ZONE 4	+ 0.16	+ 0.20
RAD. BLANKET	- 0.25	- 0.17
AX. BLANKET	- 0.56	- 0.42
TOTAL REACTOR	———	- 0.82

MOEC, STEAM DENSITY = 0.03 G/CM³

TABLE 2: SENSITIVITY OF STEAM ENTRY EFFECT TO DIFFERENT NUCLIDES

NUCLIDE	$(\nu\sigma^F - \sigma^A)_{WET} - (\nu\sigma^F - \sigma^A)_{DRY}$ [b]					
	CORE ZONE 1	CORE ZONE 2	CORE ZONE 3	CORE ZONE 4	RADIAL BLANKET	AXIAL BLANKET
U238	- 0.05	- 0.05	- 0.04	- 0.04	- 0.06	- 0.11
Np237	- 0.92	- 0.86	- 0.66	- 0.69	- 1.33	- 3.08
Pu238	- 0.10	- 0.09	- 0.03	- 0.05	- 0.36	- 0.63
Pu239	+ 0.51	+ 0.46	+ 0.35	+ 0.37	+ 1.19	+ 2.16
Pu240	- 0.24	- 0.22	0.15	- 0.15	- 3.91	- 3.76
Pu241	+ 1.69	+ 1.56	+ 1.18	+ 1.26	+ 3.69	+ 6.23
B10	- 1.71	—	- 1.26	—	—	—

MOEC. STEAM DENSITY = 0.03 G/CM³

5.2. Fuel Cycle Studies

In the past year the studies concentrated on the following ^{237}Np recyling strategies:

1. An LWR strategy with reprocessing, in which ^{237}Np is separated and burnt in a GCFR with the aim of reducing the hazard potential of the actinide waste.

2. A "proliferation resistant" LWR strategy with uranium, plutonium and neptunium recycling as illustrated in Fig. 2, which has the property that all the plutonium in the fuel cycle is "denatured", i.e. has a relatively high ^{238}Pu content.

In both strategies a GCFR with a modified fuel cycle plays an important role. This fuel cycle is described in more detail in Ref. 2.

The strategy illustrated in Fig. 2 makes use of the build-up of ^{236}U in recycled uranium. Neutron capture in this nuclide followed by beta decay leads to the production of ^{237}Np. Direct recycling of ^{237}Np in the LWRs would also allow to build up fairly high concentrations of ^{238}Pu by the same process. In the concept illustrated in Fig. 2 a GCFR is employed to convert ^{237}Np into ^{238}Pu. The desirability of a separate ^{237}Np to ^{238}Pu conversion arises from the fact that ^{237}Np is a sensitive material from the proliferation point of view and should therefore not be recycled in the reactors whose fuel cycle is to be protected. The calculations of the mass flows for the LWR strategy with uranium, plutonium and indirect neptunium recyling took fuel losses and in-pile decay of ^{238}Pu into account. The results of the analysis of this strategy can be summarized as follows:

1. By means of the standard self-sustaining uranium and plutonium recyling strategy ^{238}Pu concentrations of 3.1% in the fresh LWR fuel can be achieved asymptotically. By adding indirect ^{237}Np recycling the ^{238}Pu content of the fresh LWR fuel can be raised to 6.5%, i.e. by more than a factor of two.

2. Nowhere in the fuel cycle the ^{238}Pu concentration drops below 5%.

3. 97.5% of the power is produced in the LWRs, i.e. a single "supplier" GCFR can support 40 dispersed "client" LWRs which operate with a relatively proliferation resistant fuel cycle.

The LWRs can be of an existing design. In the proposed mode of operation with an average burnup of 75000 MWd/t (heavy metal) the GCFR achieves a ^{237}Np to ^{238}Pu conversion efficiency of 63%. Other reactor types can be expected to give slightly better conversion efficiencies but in contrast to the GCFR could not be adopted to this fuel cycle without major design modifications.

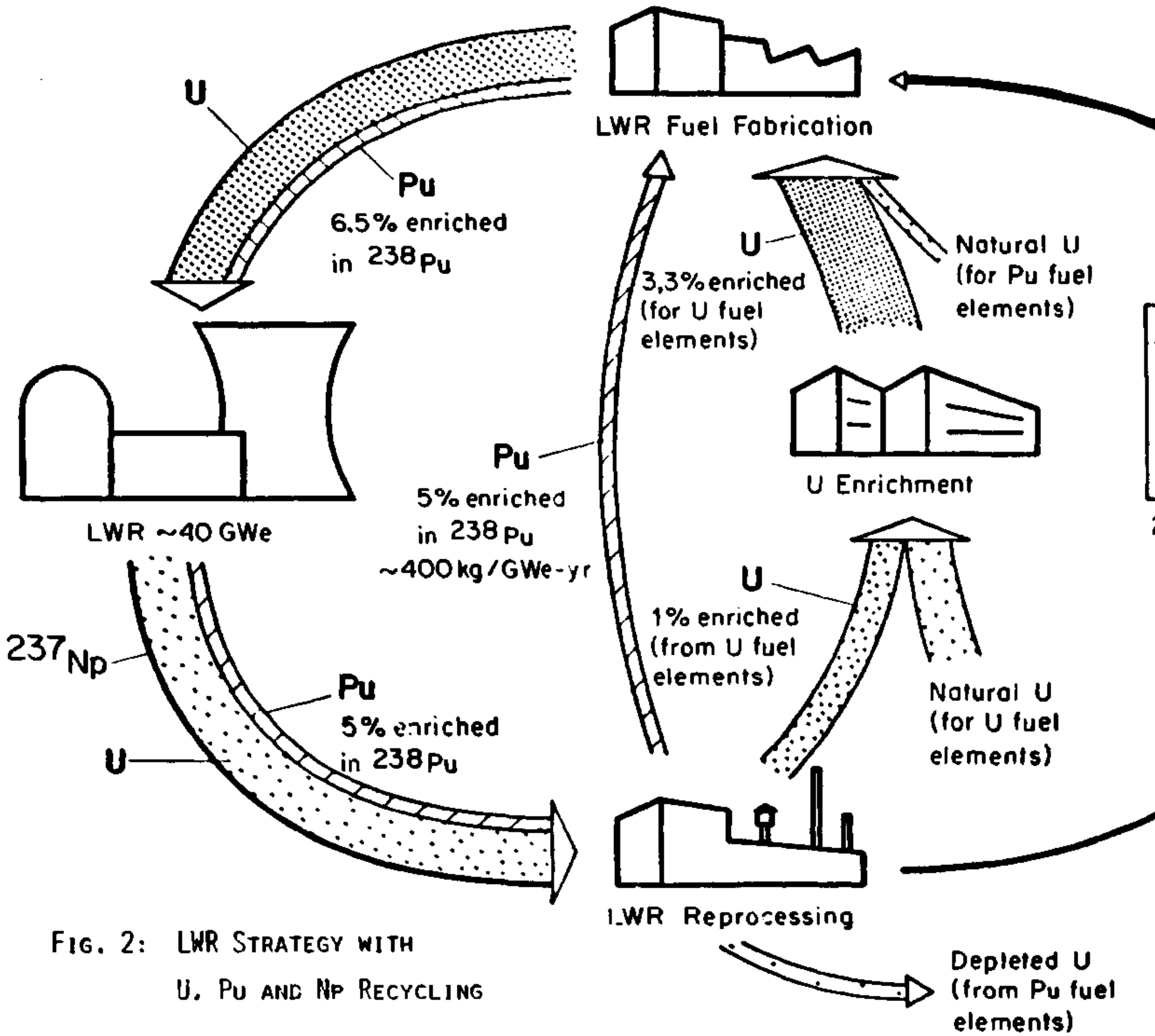


FIG. 2: LWR STRATEGY WITH
U, PU AND NP RECYCLING

5.3. References to Chapter 5

1. R.A. Moore
A Critical Experiment Program for the
300 MWe Gas-Cooled Fast Breeder Reactor
Gulf-GA-A12780, pp. 3-18 (1973)

2. P. Wydler, W. Heer, P. Stiller, H.U. Wenger
A U-Pu-Np Fuel Cycle to Produce Isotopically
Denatured Plutonium
Trans. European Nucl. Conf. 31, pp. 302-304 (1979)

6. Development of mixed carbide fuel

6.1. Fuel Development

The developments begun last year for a new gelation line for production of uranium-plutonium carbide microspheres were almost completed by the end of 1979. The new box unit with continuous sphere forming, washing and drying stages began its commissioning trials. A new unit for distilling the waste washing fluids and reconditioning for recycling was delivered and also began proving trials. With these new developments it is anticipated that a daily production of ∿ 400 g of (UPu) C should be possible.

The process development work concentrated mainly on the drying stages where significant improvements could be made. Studies where made to optimize the choice of carbon feed material (carbon black). On the analytical side an X-ray fluorescence analyser was purchased for U/Pu determination as well as analysis of trace elements. Support in deepening the understanding of the thermal treatment stages was provided by a recently installed thermogravimetric balance.

As a result of the changes described above demonstration runs producing UC particles showed a much improved and consistent quality within the desired specification.

6.2. Irradiation Studies

6.2.1. FILOS

Post Irradiation examination of the seventh FILOS experiment was completed. A report is in preparation. The examination rerealed a pattern of clad carburization at points on the clad in contact with the (hyperstoichiometric) fuel spheres where the local temperature gradients were high. This result is a strong pointer in favour of the theory that carbon transfer to the cladding occurs by solid/solid contact. Design studies were made, but not completed on an improved FILOS test section.

6.2.2. DIDO

After completion of the third DIDO experiment in 1978 the transfer of the pins back to EIR for PIE was heavily delayed. PIE finally began in October. At the time of writing the sodium capsules have not yet been opened. γ-scanning is however almost complete. There is no indication as yet of pin failures.

6.2.3. MFBS-7 DFR

A report on the MFBS-7 experiment in the BR-2 reactor at Mol was published in 1979 (EIR Report 376) and will be no longer included in these reports. The DFR PIE which was completed also showed the good behaviour of sphere-pac fuel compared to the pellets irradiated under similar conditions. Here again a report is in preparation.

6.2.4. Mol 11/K 5

No start could be made with the irradiation of this pin (now at Mol) due to the extended shutdown of BR-2.

6.2.5. Preparation for other experiments

A large amount of activity occured in developing the fabrication line for filling and fabricating "full size" fuel pins. Vibro-filling trials showed that fuel stacks of 1 m length could be filled. Considerable attention was given to methods of locating the clad tube into the filling box maintaining an effective seal without extensive contamination of the outer pin surface.

6.2.6. Fuel performance modelling

The first version of a sphere-pac code SPECKLE-1 was presented by Oregon State University, Dept of Nuclear Engineering working under contract to EIR. A preliminary model now exists of the thermal behaviour of the fuel using simple assumptions. Work also proceeded on modelling the fission/heat source distribution in the pin with increasing burn-ups. At EIR a review of the possible mechanical interactions between fuel and cladding was started. Measurements of sphere pac thermal conductivity were continued.

6.2.7. General

A review of priorities at EIR has meant that more attention is to be given to problems of current concern for Switzerland (such as waste disposal) at the expense of Projects such as this on carbide fuel development. As a result a reduction in staffing levels of $\sim$ 20% has taken place. At the same time a major part of the remaining effort will concentrate on fuel manufacture and preparation for large scale bundle tests where a research agreement is about to be signed. This realization of effort will mean that for the time being detailed parameter studies using the experimental loop FILGS will have to be postponed.