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## Analysis of SEFOR Experiments

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Abstract

The report describes methods and results of the analysis of
several SEFOR zero power and power experiments.

With few exceptions, the analysis of zero power experiments
confirms experiences which had been obtained by more simple
zero power assemblies for the more recent GfK cross section
sets and calculational methods.

The Doppler temperature coefficients as evaluated from several
SEFOR Doppler experiments are in good agreement with GE results,
although evaluation methods differ in several details.

The evaluated Doppler temperature coefficients are slightly
overestimated by nuclear calculations. However, the overesti-
mation is within the estimated uncertainties of the evaluations
from experiments.

6.12.1974

# Analyse von SEFOR Experimenten

## Zusammenfassung

Der Bericht beschreibt Methoden und Ergebnisse der Analyse verschiedener Nullenergie- und Leistungsexperimente in SEFOR.

Die Analyse der Nullenergieexperimente bestätigt mit wenigen Ausnahmen die Erfahrungen, die für die neueren Querschnittssätze und Rechenmethoden der GfK an einfacheren Nullenergieanordnungen gemacht wurden.

Die aus verschiedenen Dopplerexperimenten in SEFOR ermittelten Doppler-Temperaturkoeffizienten sind in guter Übereinstimmung mit GE-Resultaten, obwohl sich die Auswertemethoden in verschiedenen Einzelheiten unterscheiden.

Ergebnisse nuklearer Berechnungen der Dopplerkonstanten liegen etwas höher als die aus den Experimenten abgeleiteten Werte. Die Überschätzung liegt jedoch innerhalb der Analyseunsicherheiten.

Table of Contents

## 1. Preface

This report describes the main procedures and results of GfK analysis of SEFOR experiments.

It is not the purpose of this report to describe in detail experimental conditions, data collection and processing, etc., which are well documented in several GE reports, for instance in /1/ through /5/.

Emphasis of our analysis work was put on those experiments, which were specifically planned for, and the experimental conditions of which were suited for evaluation of Doppler reactivity effects. The demonstration of the shut down capability of the Doppler reactivity effects during transient power situations caused by reactivity ramps and the demonstration of an expectable degree of agreement with quantitative theoretical results were the main, and achieved, goals of the SEFOR program /6/.

The more one is interested in a quantitative transfer of Doppler reactivity effect experience from SEFOR to other fast breeders with similar neutron spectra, the more it is important to be consistent in the cross section sets used, in the models for the physical effects involved and in the calculational procedures. The difference between "measured" and calculated Doppler constants sensitively depends on several details of evaluation and calculation. To a certain degree, such details are specific for

specific groups of investigators or even countries because of
specific availability of cross section sets and codes and be-
cause of certain traditions in theoretical approximations.
Details of this report might therefore be useful for those
hypothetical future users of quantitative physical SEFOR in-
formation, who have access to GfK cross section sets and codes.

Besides of Doppler effect information (sections 3,4,5 and 6)
the report discusses SEFOR experiences concerning criticality,
material reactivity worth, fission rate distributions (sec-
tion 2), effective decay constant of delayed neutrons, prompt
neutron generation time (section 4) and fuel time constants
for changes of the temperature profile (section 5).

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## 2. Analysis of Zero Power Experiments

### 2.1 Summary

Methods and results of the analysis of some important criticality measurements, material reactivity worth measurements and fission rate distribution measurements are described in this section.

Calculated reactivity results are in reasonable agreement with experiments. Results obtained with the more recent GfK cross section sets are shown in the table below for 3 assemblies:

| Cross section set | $k_{eff}$ calc. ("experimental" is 1) | | |
|---|---|---|---|
| | 1 - C | 1 - D | 2 - A |
| MOXTOT 001 | .991 | .990 | |
| KFKINR | | .995 | .998 |

The results are consistent with experience obtained for the two cross section sets by analysis of more simple Plutonium fuelled

configurations in zero power assemblies and indicate that there
were no major errors in material distribution models and methodi-
cal procedures.

Fuel rod reactivity worths in assemblies 1-D and 2-A are over-
estimated by the calculations by amounts which plausibly can be
explained by errors in the delayed neutron data and - in the case
of assembly 1-D - by a slightly asymmetrical boron rod loading
in the core. The overestimation of boron rod reactivity worths
($\sim$ 30%) in assembly 1-D indicates an overestimation of the mo-
derating power of BeO in this assembly.

Fission rate distributions were underestimated near the axial and
radial core boundaries by typically 20% when the calculations were
normalized to experimental points near peak fission rate positions.
Some of the reasons are identified, other possible reasons are
discussed.

## 2.2 Criticality

Criticality calculations were done for three assemblies:

- Ass. 1-C  (minimum critical core after replacing the very low
  Plutonium enrichment rods with normal fuel rods)

- Ass. 1-D  (full size core with 14 $B_4C$ rods)

- Ass. 2-A  (full size core with 7 $B_4C$ rods, BeO rods replaced with
  steel rods).

The loading patterns of these assemblies are shown in Figures 2.1
through 2.3. Appendix A contains the geometrical models and regional
homogenized compositions that were used in the calculations.

## Calculational Methods

Criticality calculations were based on two dimensional diffusion cal-
culations with the code DIXY /1/. 26 neutron energy groups were used
in all cases. The pronounced cell heterogeneity effects (the fuel rod
diameter is 2.3 cm) were taken into account by using heterogeneity cor-
rected cross sections which were computed by the cell code ZERA /2/.
The fuel rod cell models used are shown in Figs. 5 and 6 of App. A:
Corresponding material densities are listed in Table II of App. A. For
purposes of comparison, some diffusion calculations were done for the
homogenized core regions.

Comparative one dimensional (axial and radial) 26 group diffusion and
$S_8$-calculations were performed in order to evaluate reactivity correc-
tions due to transport effects.

The reactivity calculations included the so called REMO correction /3/
which essentially is a correction of the elastic downscattering cross
sections. It takes into account that the collision density spectrum in
the specific reactor differs from the standard spectrum that was used to
compute the microscopic cross sections of the 26 group set.
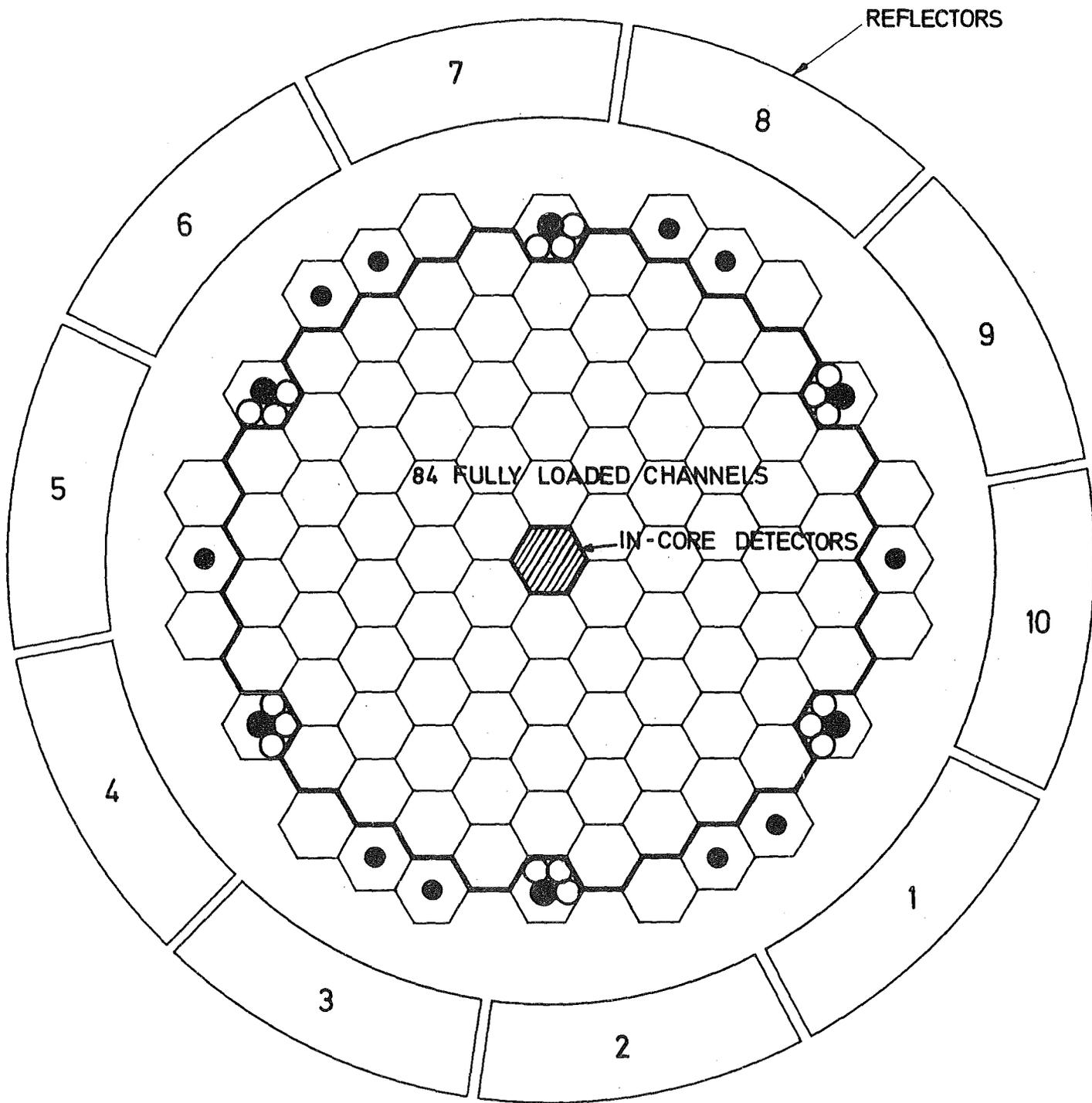
FIG. 2.1    SEFOR  MINIMUM  CRITICAL  LOADING  (ASSEMBLY  I-C )

FIG. 2.2   CORE LOADING FOR ASSEMBLY I-D

Central channel
(no fuel)

Reflectors
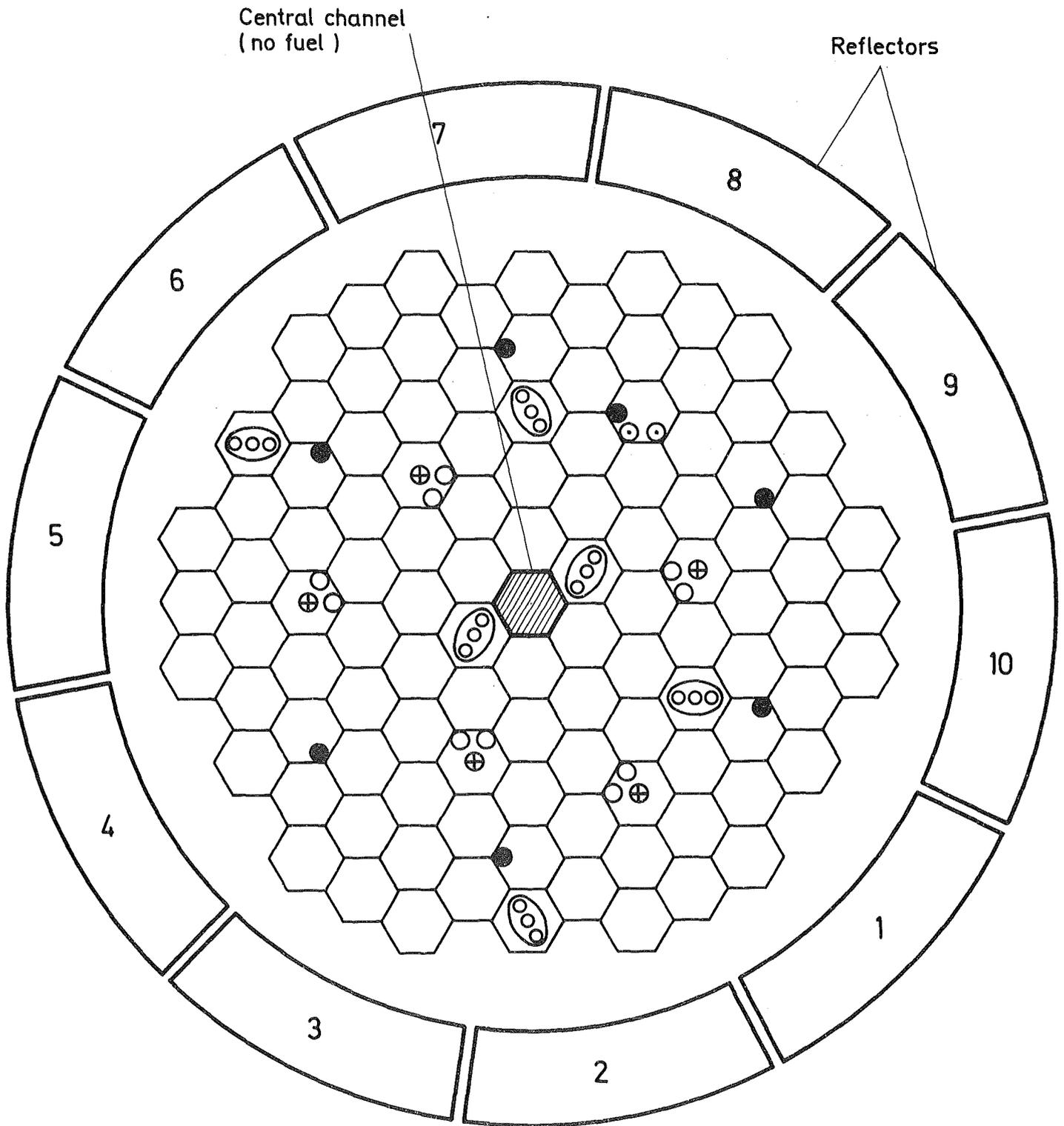
617 Standard fuel rods
97 Standard stainless steel tightener rods
6 IFA BeO tightener rods
7 B$_4$C rods
2 GUINEA pig rods

⬭OOO⬮ IFA's
● B$_4$C rods
O Vacant fuel locations
⊕ Vacant tightener rod locations
⊙ GUINEA pig rods

Fig. 2.3    Core II Loading Assembly II - A

The SEFOR criticality measurements were done at a core temperature of
$450^\circ$ K. The atom densities listed in Appendix A correspond to this
temperature. However, the diffusion and transport calculations were
done with microscopic cross sections for $300^\circ$ K, and a Doppler effect
correction was necessary to extrapolate the results to $450^\circ$ K.

The geometrical reactor models correspond to the situation that all
reflector segments are in their upper position. During the measurements,
criticality was achieved by partially lowering one or two control
reflector segments. Thus, the calculations refer to a hypothetical more
or less supercritical situation ($k_{eff}$ = 1 + excess reactivity con-
troled by lowered reflectors). The excess reactivity was negligable
only in Ass. 1-A and 1-C (minimum critical loadings).

## Results of Criticality Calculations and Discussion

Most of calculations for Ass. 1-C and 1-D were performed with the
MOXTOT cross section set. The main results are shown in Table I:

Table I:

### Results of Criticality Calculations for Ass. 1-C and 1-D
### Obtained with MOXTOT Cross Sections

| Method or Correction | $k_{eff}$ | |
|---|---|---|
| | 1-C | 1-D |
| 2 dim. diffusion with heterogeneous cross sections | .9773 | .9825 |
| Transport correction | +.011 | +.0104 |
| REMO correction | +.004 | +.004 |
| Doppler correction $300^{\circ}K \rightarrow 450^{\circ}K$ | -.003 | -.003 |
| Excess tightener rods | +.0013 | - |
| Reflector position | - | -.004 |
| Total | .991 | .990 |

The correction for excess tightener rods (BeO) was added because of
the irregular BeO/fuel ratio at the core boundary in Ass. 1-C (compare
Fig. 2.1). It was based on material worth measurements for BeO rods.
The reflector position correction was based on control reflector reac-
tivity calibration measurements.

The heterogeneity effects on reactivity were $\Delta k_{eff}$ = +.0069 for Ass. 1-C and $\Delta k_{eff}$ = +.0110 for Ass. 1-D.

The results indicate a consistent underestimation of about one percent of criticality in both assemblies. This is quite satisfying in view of cross section uncertainties and of methodical difficulties in dealing with the rather complicated geometry.

The underestimation of criticality is consistent with similar under-estimations of zero power Plutonium assemblies reported by Kiefhaber /4/ for MOXTOT calculations.

For purposes of comparison, some one dimensional (radial) results of 26 group calculations for Ass. 1-C are shown in Table II for some other cross section sets.

Table II   One Dimensional Reactivity Results for SEFOR 1-C and
           1-D for Some Cross Section Sets

| Set | 1 - C | | 1 - D | |
|---|---|---|---|---|
| | $k_{eff}$ | $k_{eff} - k_{eff}$(MOXTOT) | $k_{eff}$ | $k_{eff} - k_{eff}$(MOXTOT) |
| MOXTOT 001 | .9662 | - | .9663 | - |
| KFKINR | - | - | .9755 | +.0092 |
| NAPPMB 001 | .9594 | -.0068 | .9592 | -.0071 |
| SNEAK 001 | .9529 | -.0133 | - | - |
| KFKSET 040 | .9873 | +.0211 | - | - |

Criticality calculations for Ass. 2 A were performed with the KFKINR set and based on two dimensional homogeneous diffusion calculations. The results are shown in the table below:

| | $k_{eff}$ |
|---|---|
| 2 dim. diffusion, hom. [a] | .997 |
| | |
| Heterogeneity correction | +.008 |
| Transport correction [b] | +.010 |
| REMO correction [c] | - |
| Doppler correction $300^{\circ}K \to 450^{\circ}K$ | -.002 |
| Irregular core loadings [d] | -.008 |
| Reflector position | -.007 |
| | |
| Total | .998 |

Comments:

a) For reasons of consistency with the heterogeneity correction and of comparison with Core 1 results, the number quotet is that for ZERA cross sections for lattice cell dimensions reduced by a factor $10^{-3}$.

b) The correction was assumed to be the same as for Ass. 1-D.

c) The REMO correction was not calculated. The collision density spectrum of Ass. 2 A is similar to the spectrum used for the evaluation of KFKINR cross section set. Based on experience with REMO corrections for assemblies with similar spectra it is estimated that the REMO correction is smaller than .001.

d) The main contribution is due to 10 vacant fuel rod locations. Addidional contributions are for the presence of 5 vacant tightener rod locations and for the presence of 2 Guinca Pig rods (see Fig. 2.3). The corrections were based on material worth measurements.

An exact comparison with results for Ass. 1-D is not possible. However, Table II indicates a $k_{eff}$ difference of .009 between KFKINR and MOXTOT calculations, neglecting corrections. Minor differences in the corrections for the two cross section sets can be expected from the REMO correction, which is .004 for MOXTOT assumed to be negligable for KFKINR. The estimate for the KFKINR result for Ass. 1-D is therefore: .009 + .009 - .004 = .995

The difference between this number and .998 for Ass. 2 A is not as small as the MOXTOT-difference between Ass. 1-C and 1-D, but it is within the expected uncertainties. Some possible reasons for the difference are:

- an overestimation of the $B_4C$ rod worth in Ass. 1-D (see below)

- errors in the geometrical modelling of $B_4C$ containing regions in the core

- errors in the moderating power of BeO and SS and in the neutron absorption of steel,

- differences between the (not calculated) KFKINR REMO correction for Ass. 1-D and 2-A

- differences in the impact of various possible cross sectional errors of the KFKINR set for the softer 1-D spectrum and the harder 2-A spectrum on $k_{eff}$.

Despite of the .3% inconsistency of the KFKINR results for Ass. 1-D and 2-A, both reactivity results are rather close to 1, which again is consistent with other satisfying experiences with criticality calculations for Plutonium fuelled critical assemblies with this cross section set.

In summary, the criticality calculations for the three SEFOR assemblies analyzed indicate, that there are no major errors in geometrical models and methodical procedures.

## 2.3 Analysis of Material Worth Measurements

Material reactivity worth experiments in SEFOR were performed by removal or exchange of rods of certain materials at various radial positions in the core. Experimental details are described in /5/ and /6/.

Our general calculational procedure for material reactivity worths was the application of perturbation theory to two dimensional 26 group neutron flux and adjoint distributions.

Fig. 2.4 and Fig. 2.5 show measured and calculated (with the KFKINR set) fuel rod reactivity worths as functions of the radial posion in the core. The fuel rod worth is overestimated by about 10% for all positions in Core 1 (Ass. 1-D). The overestimation is essentially reduced for Core 2 (Ass. 2-A).

Boron rod reactivity worths in Core I are overestimated by typically 30% with the KFKINR set and by typically 10% in Core II (compare Figs. 2.6 and 2.7) Fig. 2.6 demonstrates that the situation in Core 1 is even worse for the MOXTOT set.

Comparative cell calculations for evaluation of flux depressions in and around $B_4C$ rods in Core 1 led to a 6 percent decrease of the neutron capture rate in $B_4C$ compared to the homogeneous case (compare Appendix B). A similar correction was not calculated for Core 2. However, from the difference of the neutron spectra in Ass. 1-D and Ass. 2-A, it can be estimated that the Core 2 correction would not be essentially smaller.

Cell calculations also showed that the heterogeneity correction for the fuel rod reactivity worth is less than 2%.

A possible contribution to the overestimation of reacitivity worths in Ass. 1-D is the slightly asymmetrical boron rod loading in this assembly, resulting in a certain flux and adjoint tilting. Most of the rod worth measurements were done along a radius extending from the center towards

Fig. 2.4   SEFOR 1-D
Fuel Rod Reactivity Worth Results

- 16 -
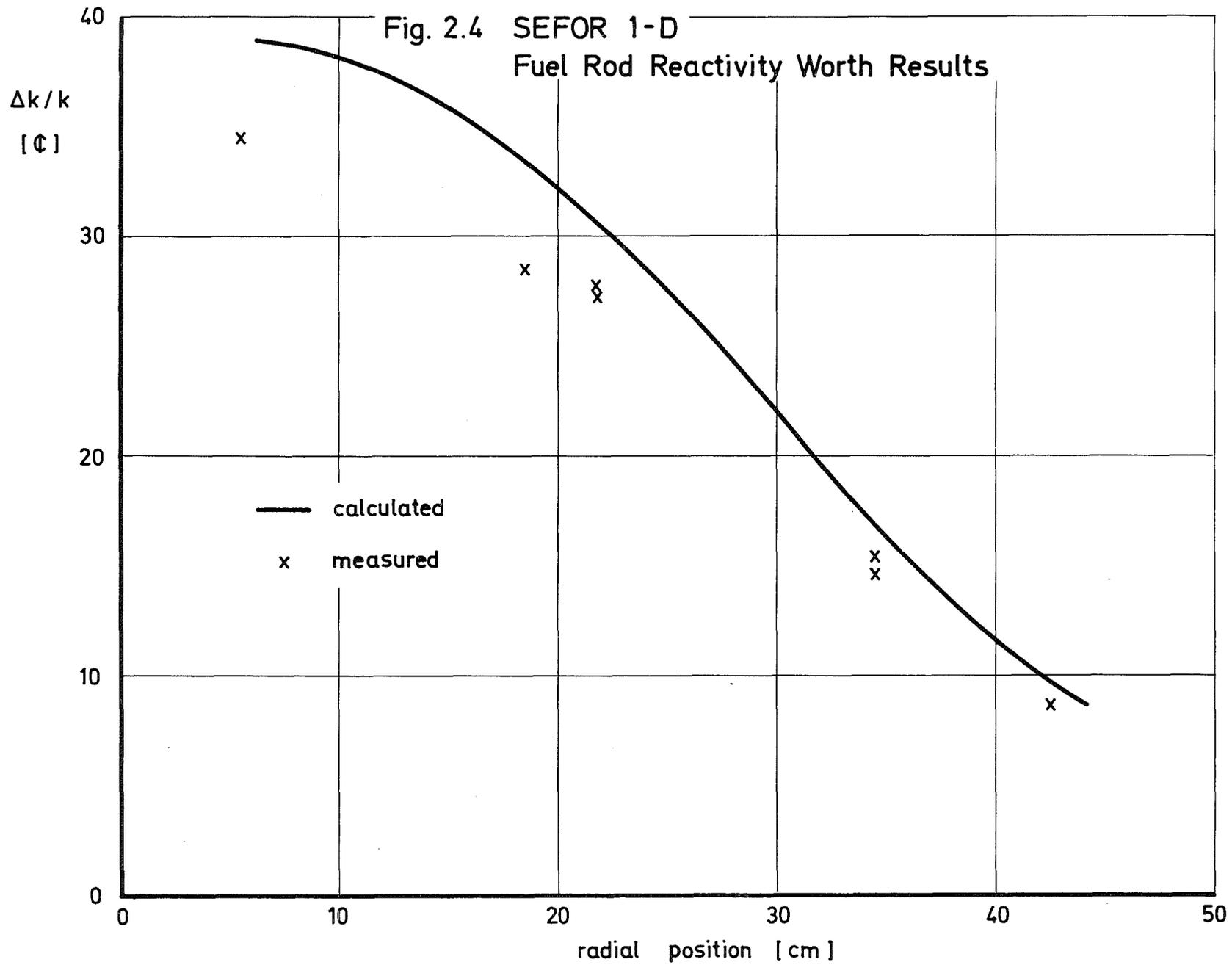
Fig. 2.5   SEFOR II-A
Fuel Rod Reactivity Worth Results
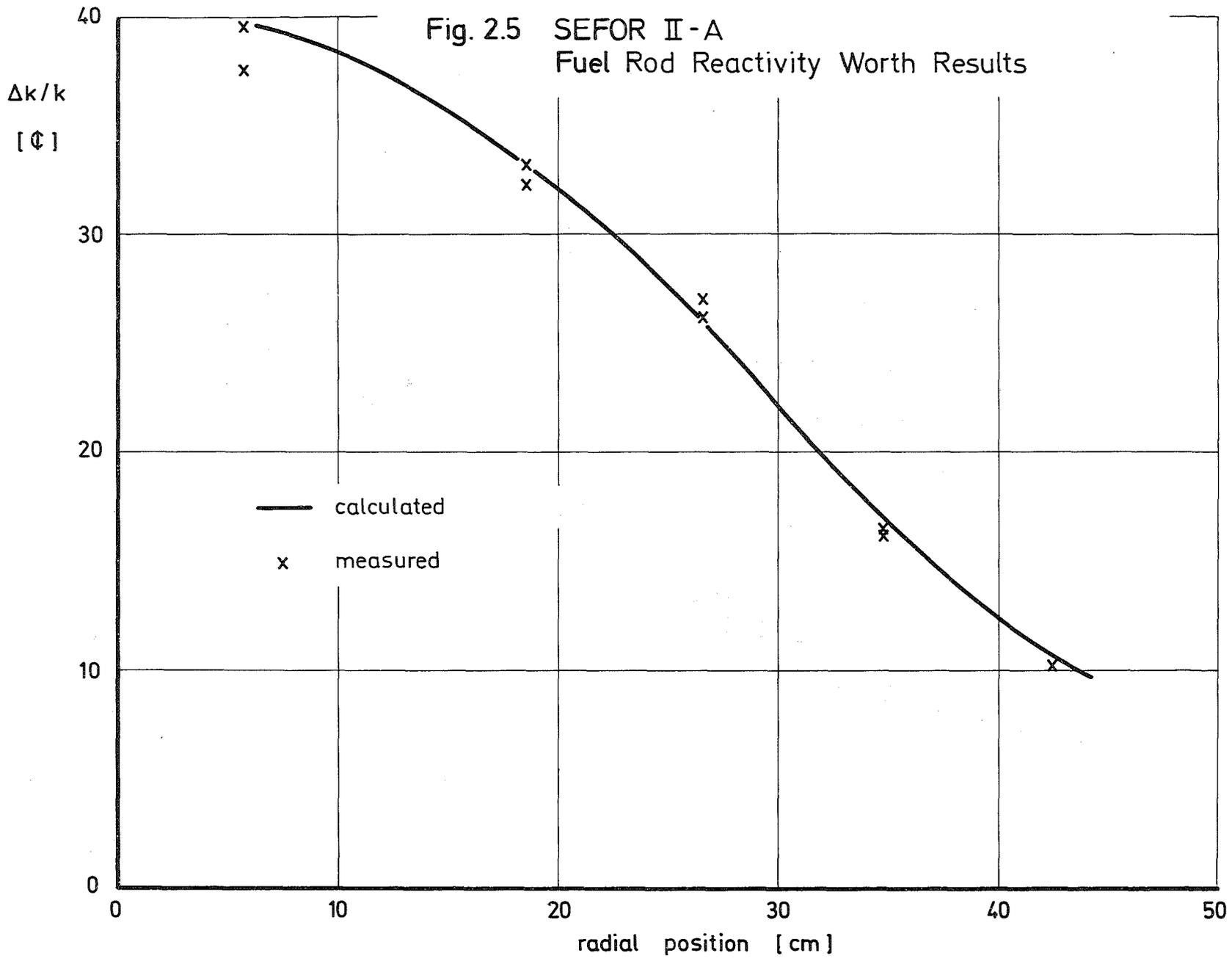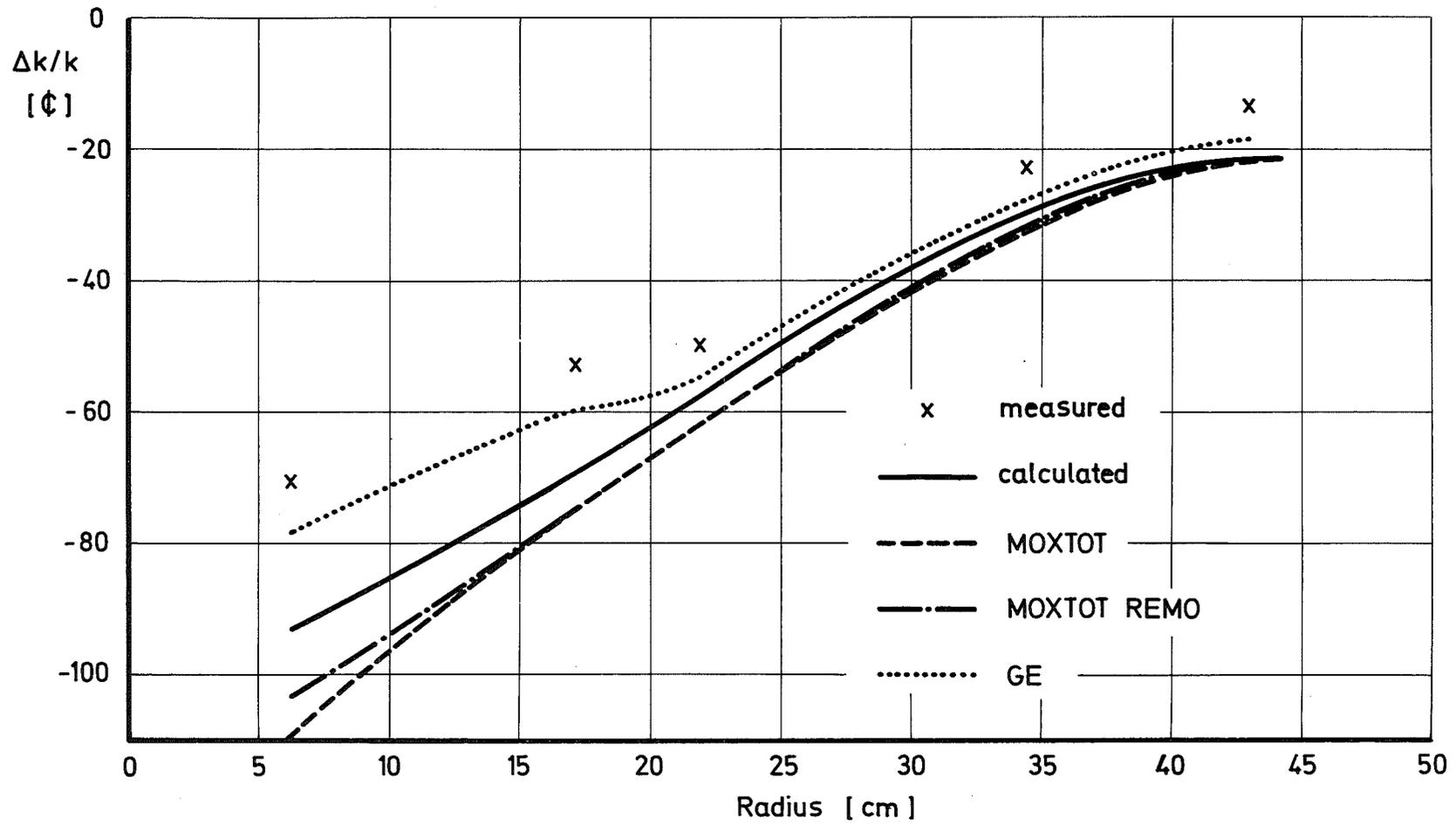
Fig. 2.6    SEFOR I-D
Boron Rod Reactivity Worth Results

Fig. 2.7    SEFOR 2-A
Boron Rod Reactivity Worth Results
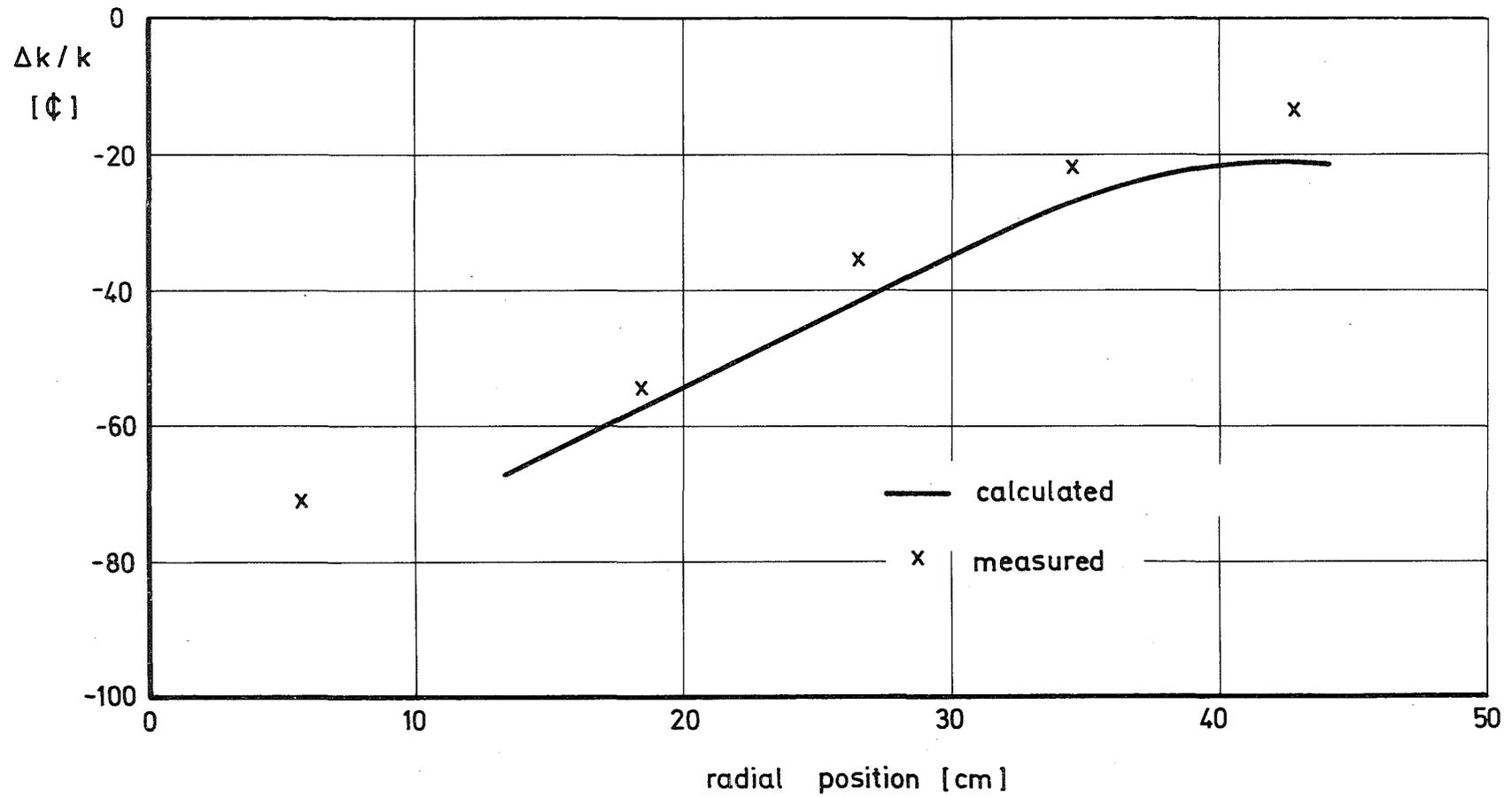
the gap between reflectors 1 and 10 (compare Fig. 2.2). If one draws
a core diameter in Fig. 2.2 perpendicular to this direction, one re-
cognizes that more boron is on the right half of the core, where most
of the measurements were done. One would then expect somewhat higher
reactivity worths at the opposite side of the core (towards the in-
tersection between reflectors 5 and 6). The only points in Fig. 2.4
and Fig. 2.6 measured at this side are those at a radial position of
21.8 cm. In fact, the ratio between calculation and experiments is
lower by 5 to 10 per cent at this position.

Two dimensional (r,z) diffusion theory calculations described in
App. B show that this order of magnitude can be expected as flux
tilting effect on rod worths caused by the asymmetrical boron load-
ing. Similar effects in Ass. 2-A are expected to be very small (com-
pare the boron loading pattern, Fig. 2.3).

All reactivity worth calculations expressed in $ are subject of
possible systematical errors introduced by errors in $\beta_{eff}$, the
fraction of delayed neutrons. More recent differential and inte-
gral measurements of this fraction at Los Alamos /7/, /8/ and in
SNEAK indicate values for $\beta_{eff}$ in U-Pu fuelled assemblies which
are higher by about 5 to 10 per cent than those calculated (as
usually) with Keepins /9/ data. This would reduce the overestima-
tion of material reactivity worths in the SEFOR assemblies by
about the same amount.

Additional reasons for the difference between calculated and measured
$B_4C$ reactivity worths must exist for Ass. 1-D. One suspects an over-
estimation of the moderating power of the BeO rods in this assembly,
which would lead to an overestimation of the low energy part of the
neutron spectrum. One should expect a similar overestimation by the
calculations of the ratio $1/\beta_{eff}$ (1 is the prompt neutron generation

time), because $1/\beta_{eff}$ is proportional to the reactivity worth of a
1/v-absorber distributed throughout the reactor. In fact, this ra-
tio was overestimated by 21% in Core I and by 8% in Core II (by two
dimensional diffusion calculations with the KFKINR-set). The table
below shows absolute numbers for $1/\beta_{eff}$.

| Cross Section Set | Ass. 1-D | Ass. 2-A |
|---|---|---|
| MOXTOT | 174.1 µs | -- |
| KFKINR | 168.7 µs | 151 µs |
| EXPERIMENT[+] | 140.5 µs | 140 µs |

+) Noise measurements

## 2.4  Fission Rate Distributions and Fission Ratios

Fission rate distributions were measured in SEFOR by foil irradia-
tion and subsequent fission product gamma activity measurements /5/,
/6/. Results of these measurements are shown together with calcula-
tional results in Figs. 2.8 through 2.13. Calculated distributions
are normalized to measurements near the core center. They are based
on two dimensional 26 group diffusion calculations with the MOXTOT
set.

The graphs show a rather consistent underestimation of the fission
rate near the axial and radial core boundary. Similar underestima-
tions were found by two dimensional 13 group diffusion calculations
performed by GE. It can also be seen that the axial fission rate
distributions are not accurately calculated in the expansion gap.
Estimated experimental uncertainties are $\pm$ 5% for the axial profiles
and $\pm$ 10% for the radial profiles /5/.

It has not been possible to explain completely the deviations shown.
Radial transport calculations ($S_8$- approximation) led to a 2.1% in-
crease of the power density near the radial core boundary, but to
negligible changes of the axial power distribution.

The complicated geometrical arrangement near the core boundary might
well be one of the reasons for calculational inaccuracies. Compara-
tive axial diffusion calculations led to an increase of the power
density at the bottom and the top of the core of 3.8%, when a more
detailed geometrical model was used to describe the material distri-
bution at and beyond the core edges (the number of axial regions was
increased in these calculations from 8 to 14). In the radial direc-

## FIG. 2.8



U238

RADIAL FISSION RATE

DISTRIBUTION

o MEASURED
--- CALCULATED (GE)
— CALCULATED (GfK)

27%

## FIG. 2.9



Pu 239

RADIAL FISSION RATE

DISTRIBUTION

o MEASURED
--- CALCULATED (GE)
— CALCULATED (GfK)

28%

FIG. 2.10



U 235

RADIAL FISSION RATE

DISTRIBUTION

○ MEASURED
---- CALCULATED (GE)
━━ CALCULATED (GfK)

$R_f$

$r\,[\,cm\,]$

FIG. 2.11



U 235

AXIAL FISSION RATE

DISTRIBUTION

○ MEASURED
---- CALCULATED (GE)
━━ CALCULATED (GfK)

$R_f$

DISTANCE ABOVE BOTTOM OF CORE [cm]

FIG. 2.12



U 238

AXIAL FISSION RATE

DISTRIBUTION

o   MEASURED
----CALCULATED (GE)
——CALCULATED (GfK)

GAP

DISTANCE ABOVE BOTTOM OF CORE [cm]

FIG. 2.13



Pu 239

AXIAL FISSION RATE

DISTRIBUTION

o   MEASURED
----CALCULATED (GE)
——CALCULATED (GfK)

GAP

DISTANCE ABOVE BOTTOM OF CORE [cm]

tion, a sufficiently detailed description of the core edge region is not possible in cylindrical geometry (compare Fig. 2.2).

Expected methodical errors are also due to the treatment of resonance shielding effects near the core boundaries, where the shielding effects are certainly less than within the core. In the first instance, these effects would increase the resonance fissions in $^{239}$Pu and $^{235}$U near the core boundaries compared to our calculational results. $^{238}$U fission rates would be increased indirectly by additional fast neutrons due to increased resonance fissions in $^{239}$Pu.

The results of one dimensional radial diffusion calculations (Table III) give an impression about the dependence of power distribution parameters on cross section sets. $P_r/\hat{P}$ is the ratio between power density at the radial core boundary and maximum power density.

Table III    $P_r/\hat{P}$ for Some Cross Section Sets

| Set | $P_r/\hat{P}$ |
|---------|---------|
| MOXTOT | .449 |
| KFKINR | .446 |
| NAPPMB | .457 |
| SNEAKOO1 | .447 |
| KFKSETO40 | .416 |

Similar (relative) differences were found for axial power distribution calculations.

An other suspected reason for the fission rate distribution discrepancies is an underestimation of neutrons returning from the nickel reflectors because of errors in the Ni cross sections used.

Approximate fission ratios at peak power position as obtained from the 1.6 MeV $^{140}$La gamma decay peak of the irradiated foils /5/ are listed in Table IV together with calculated (MOXTOT) results. The estimated experimental uncertainty ($2\sigma$) is $\pm$ 10%.

Table IV      Measured and Calculated Fission Ratios

| Ratio | Experiment | Calculation 2-dim. | Calculation 0-dim.[+] |
|-------|------------|--------------------|------------------------|
| $\sigma_{f8}/\sigma_{f5}$ | .0252 | .0236 | .0252 |
| $\sigma_{f9}/\sigma_{f5}$ | .905 | .857 | .866 |
| $\sigma_{f8}/\sigma_{f9}$ | .0278 | .0275 | .0290 |

[+]    from cell calculations with energy independent critical buckling

Convincing conclusions upon errors in the calculated neutron spectrum can not be drawn from these numbers. The difference between experimental and two dimensional results is consistent with experience obtained with the MOXTOT set by analysis of SNEAK and ZPR-III Pu assemblies.

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## 3.  Analysis of the Static Tests

### 3.1  Summary

The Doppler coefficient has been determined by the use of the measured reactivity feedback at different power levels. For this purpose the temperatures at different points in the reactor have been calculated using a model that takes into account restructuring of the fuel.
A Doppler constant of $A_D = -2.3 \pm .3$ $ for Core I, $A_D = -1.8 \pm .3$ $ for Core II has been obtained.

Experimental details of the static Doppler measurements are described in /1/ and /2/.

### 3.2  Outline of the Calculations

The reactor was described for the temperature calculations by about 100 regions. For each of these regions the linear pin power in W/cm was calculated for a given thermal power of the reactor. Fuel temperature was then calculated going from the gap inward. The thickness of the gap between fuel and cladding was calculated from the expansion of both using the average temperatures. A few iterations were necessary as the temperature drop across the gap and thus the fuel average temperature is very sensitive to the thickness of the gap. In case that there was no gap a constant heat conductivity was used. The fuel temperatures are then calculated. Restructuring of the fuel was taken into account when the fuel temperatures

exceeded threshold values of 1650°C for 97% density and 2150°C for 99% density. The central hole was then calculated from the conservation of mass. For subsequent power ascension it has been assumed that the reactor had operated at 20 MWth and the interfaces between fuels at different densities remained at the position which they reached at 20 MWth. The contribution of each region to the Doppler effect was calculated using the fuel average temperature rise and a global flux times adjoint flux weighting function.

## 3.3 Heat Transfer Model

We assumed that the gap can be described by

$$\delta = a + \gamma_c \, r_c \cdot T_c - \gamma_f \cdot r_f \cdot T_f \tag{1}$$

where  a  =  .19 mm is the initial gap at $0°C$

$r_c$  =  11.3 mm is the inner radius of the cladding at $0°C$

$r_f$  =  11.11 mm is the radius of the fuel pellets at $0°C$

$T_c$  =  is the calculated clad temperature

$T_f$  =  is the average fuel temperature

$\gamma_c$  =  17.62 x $10^{-6}$/$°C$ is the thermal expansion coefficient for SS

$\gamma_f$  =  7.54 x $10^{-6}$ (1. + 13.85 x $10^{-4}$ $T_f$) is the thermal expansion coefficient for the fuel

For the temperature drop over the gap the heat conduction formula
of Kaempf /3/ has been used

$$\Delta T = \frac{X}{2\pi r_f} \cdot \frac{\delta}{k^+} \cdot \left[ 1 - \frac{\delta}{k^+} (4 \, P_{st} \cdot T^3 + \frac{.79}{8} \cdot \frac{X}{T} \cdot \frac{1}{2\pi r_f}) \right] \qquad (2)$$

where $X$ = is the linear power rating of the pin in (W/mm)

$k^+$ = $1.58 \times 10^{-5} \, T^{.79}$

$P_{st}$ = $.378 \times 10^{-13} \, \text{Wmm}^{-2} \, {}^{o}\text{K}^{-4}$

$T$ = is the temperature of the outer surface of
the fuel ($^{o}$K)

Eq. (2) has been used for gap widths $\delta$ greater than 0.05 mm. For
$0 < \delta < .05$ mm a linear interpolation between Eq. (2) and a constant
value for the heat resistivity of $1^{o}$C $\text{cm}^2 \, \text{w}^{-1}$ has been used.

In case that the gap was closed a value of the heat resistivity
of $1^{o}$C $\text{cm}^2 \, \text{w}^{-1}$ was used.

For the fuel itself the recommended values of Schmidt /4/ for
the heat conductivity were used

$$A = (1.144x(1.-2.5x(1.-x_9))) \, /$$
$$/ \, (10.8+5.4+.0235x\Theta))x(1.-2.74x10^{-8}xT^2+2.5x10^{-14}xT^4)$$

where $\Theta$ = $T$ for $T < 1950^{o}$C

$\Theta$ = 1950 for $T > 1950^{o}$C

$x_9$ is the fraction of the theoretical density of the
fuel and has the following value depending on the
maximum local temperature reached in the history
of the reactor

$$x_9 = .92 \text{ for } T_{max} < 1650^\circ C$$

$$x_9 = .97 \text{ for } 1650 < T_{max} < 2150^\circ C$$

$$x_9 = .99 \text{ for } T_{max} > 2150^\circ C$$

The calculated temperature rise agrees to within 10% with the tem-
perature rise (Fig. 3.1) as measured in the Core II instrumented
fuel assemblies during the ascension to power. (The Core I instru-
mented fuel assemblies had certain malfunctions, and their tem-
perature data are less suited for a comparison with calculations
than the Core II data.)

## 3.4 Doppler Determination

For each region of the reactor a constant linear power per MW reac-
tor output has been assumed. It was further assumed that the power
could be separated axially and radially. The power distribution is
given in Fig. 3.2 and Fig. 3.3. A maximum rating of 65 W/mm was ob-
tained (see Table I).

The boundaries of the columniar growth region and the central void
were calculated for a reactor power of 20 MWth. Average fuel tem-
peratures were then calculated with these boundaries for different
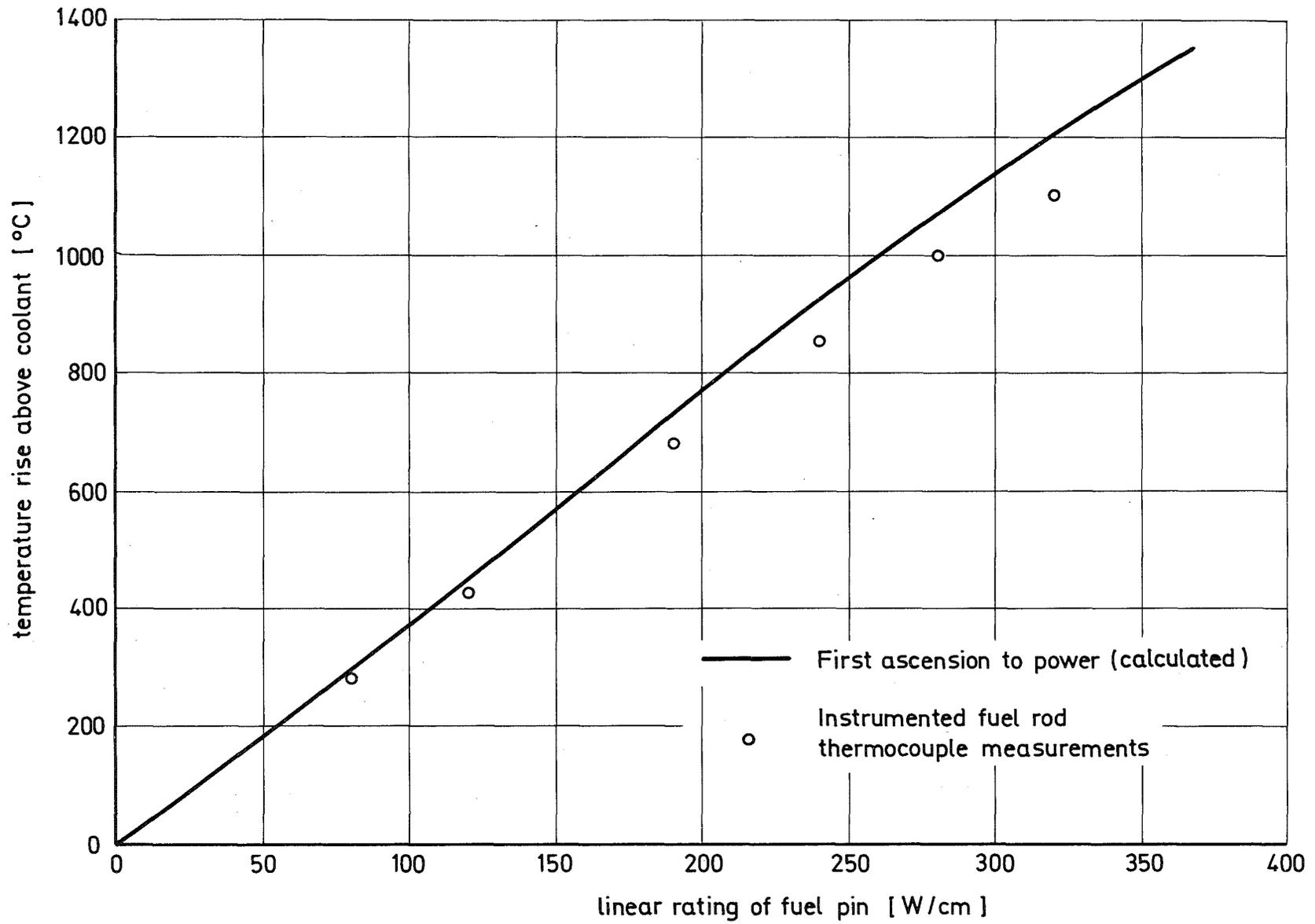
Fig. 3.1   CORE II  IFA Temperatures
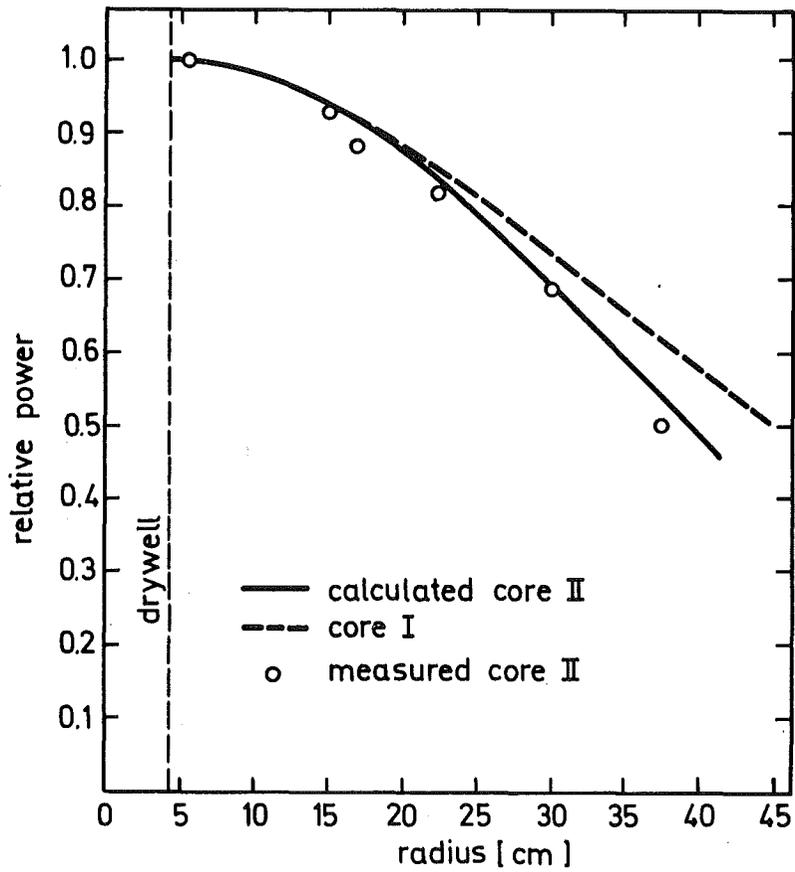
Fig. 3.2     Radial Power Distribution
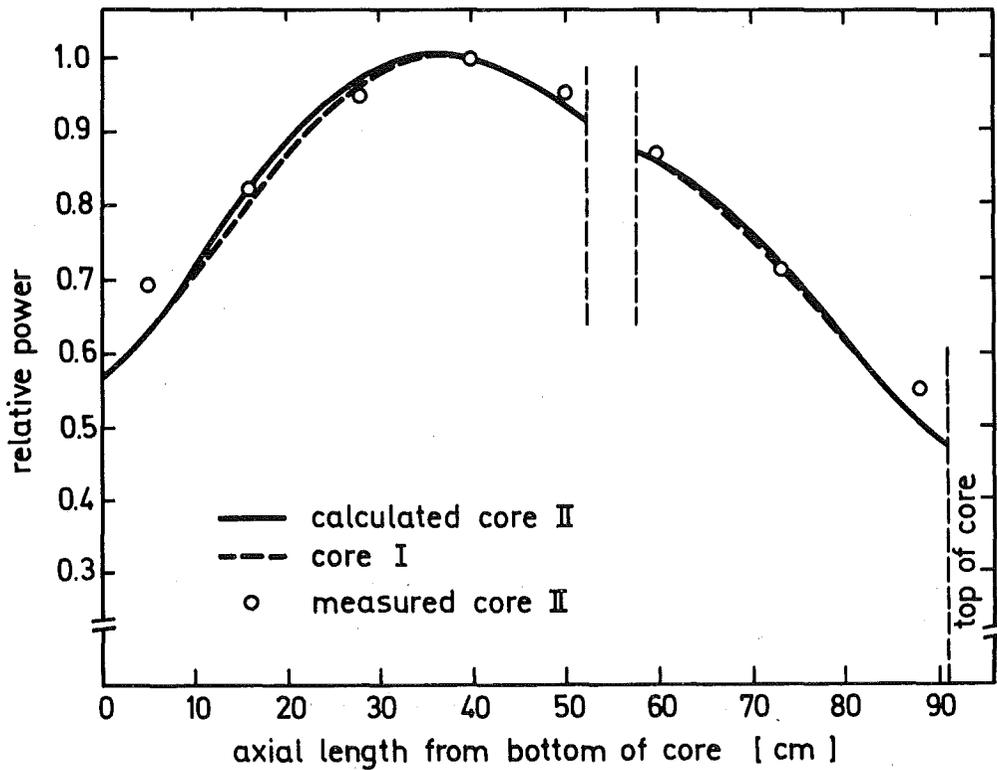


Fig. 3.3     Axial Power Distribution

Table I        Fuel Data for Temperature
                    Calculations

| Mass of the fuel | 2150 kg |
| Density | 10000 kg/m$^3$ |
| Fraction of energy released in the fuel | .997 |
| Average fuel rating at 20 MWth | 9.1 KW/kg |
| Fuel pellet diameter | 11.13 mm |
| Average linear rating | 35.4 w/mm |
| Maximum to average power | 1.83 |
| Maximum rating | 65 w/mm |

power levels. It has been assumed that the Doppler effect obeys a ln T law, where the temperature is the average pin temperature.

Each region was weighted with the flux and adjoint. Both were taken proportional to the calculated power profiles of Fig. 3.2 and Fig. 3.3.

The Doppler coefficient was then adjusted to fit the calculated Doppler feedback to the experimentally determined effect of 1.84 $ for Core I. Thus we obtained a Doppler coefficient of $A_D = -2.3 \pm .3$ for Core I and $A_D = -1.8 \pm .3$ for Core II.

A different approach for the evaluation of the Doppler coefficient in Core I is described in /6/. In this approach, an analytical relation between effective fuel temperature and reactor power was used, which was based on an analytical approximation for the dependence on temperature of the fuel thermal conductivity. The results for $A_D$ are very similar to those quoted above when the same ratio between effective temperature and average fuel temperature is used.

## 3.5 Error Analysis

The experimental Doppler feedback as a function of power had a different profile than the calculations indicate (Fig. 3.4). This could not be explained by the restructuring of the fuel. A steady loss of reactivity in the rise to power had been observed during the course of the SEFOR tests. This and the different profile

might be explained by a closing of the gap. The build up of fission gas pressure has been proposed as a mechanism that closes gradually the gap /5/. Additional uncertainties in the calculated profile of the relation between Doppler feedback and power may be caused by possible deviations from the ln T law and by other calculational uncertainties. In order to check the effect of gap closure we have assumed that no gap exists and obtained the lower curve in Fig. 3.4. The Doppler coefficient has been adjusted by 5% so that the calculated feedback agrees with the experimentally obtained value at 20 MWth. It is difficult to define correctly a gap in the case that the fuel is cracked. We feel it is reasonable to take the curves of Fig. 3.4 as upper and lower boundaries of the Doppler feedback effect.

Results of an error analysis are summarized in Table II.
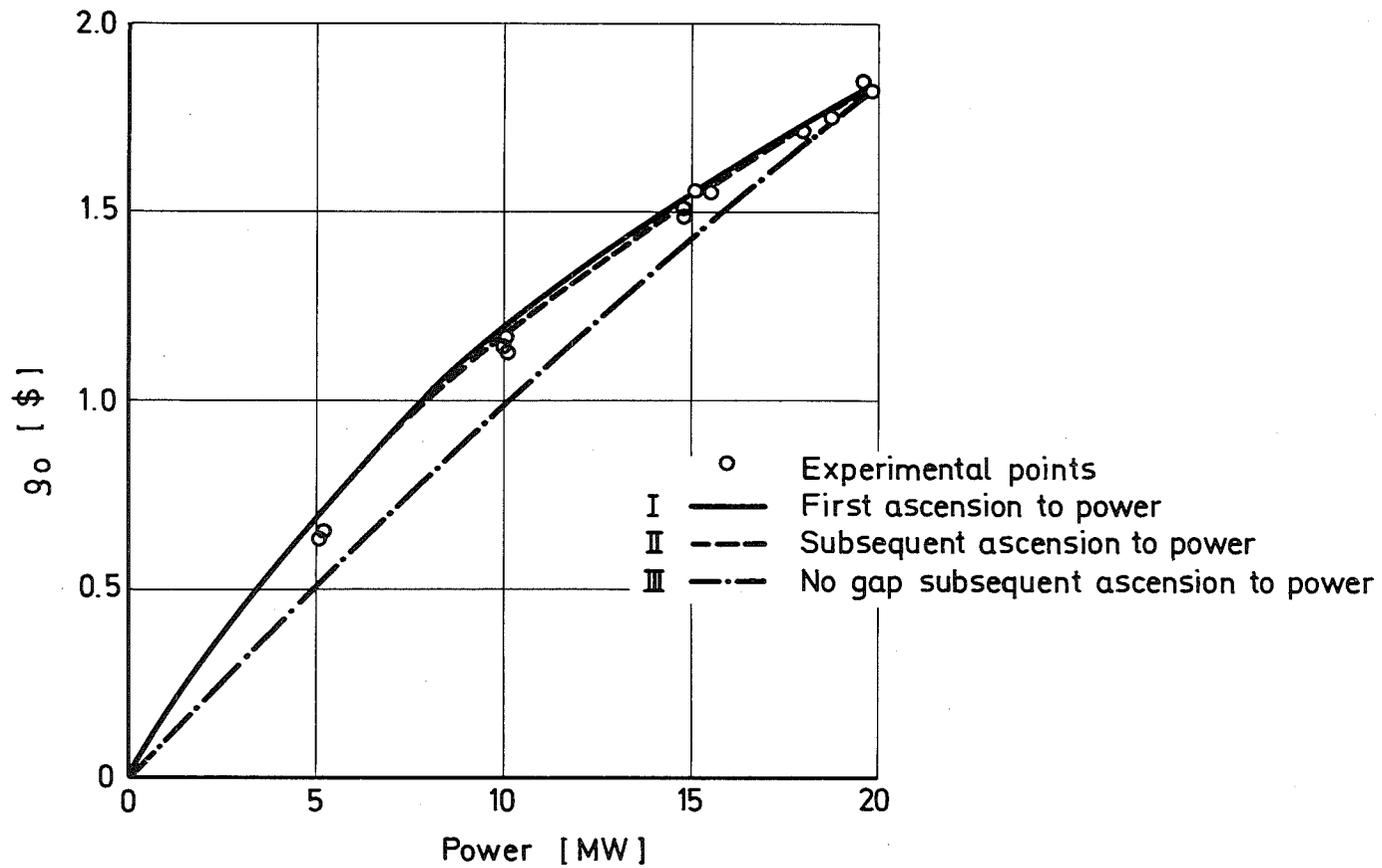
Fig. 3.4    Doppler Feedback as Function of Power for Core I

Table II          Summary of Error Analysis Results[+] for
                  Doppler Evaluation from Static Tests

| Absolute | |
|---|---|
| Absolute power | 4% |
| Non Doppler feedback | 4% |
| Reflector calibration errors | 4% |
| Changes in pellet structure + gap | 4% |
| Thermal conductivity | 8% |
| Uncertainty in oxide to metal ratio | 5% |
| Uncertainty of density variation | 4% |
| Global weighting | 5% |
| | 14% |

[+]    Uncertainties estimated for a confidence
       level of ∿ 90%

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KFK-1271/1 (1971) - Section 1216 SEFOR

4. Analysis of the Transient Tests

4.1 Summary

The transient tests were undertaken to obtain an accurate measurement of the Doppler energy coefficient in a representative reactor environment, and to provide a convincing demonstration of its effectiveness in limiting the energy release in an LMFBR power excursion. In this chapter the analysis of the transients is presented. The results are summarized in Table I for the evaluated Doppler constants, neutron generation times 1 and for the average decay constant $\bar{\lambda}$ of delayed neutrons. Details of the experiments and evaluations of GE have been published in /1-3/.

Table I

|  | $A_D$ | $1/\beta$ | $\bar{\lambda}$ (sec$^{-1}$) |
|---|---|---|---|
| Core I | 2.67 ± .2 $ | .16 ± .01 x 10$^{-3}$ | .625 ± .02 |
| Core II | 1.82 ± .2 $ | .13 ± .01 x 10$^{-3}$ | .625 ± .02 |

## 4.2 Description of the Tests

In these tests the reactor power was brought to an initial steady
state power level to establish representative fuel rod temperature
distributions, whereupon a $B_4C$ rod was ejected from the core in
approximately .1 seconds /1/. The resulting power transients were
moderated by the Doppler reactivity until a delayed scram termi-
nated the tests at $\sim$ 300 msec. A series of subprompt tests was
run for each core prior to the ejection of $B_4C$ rods with reacti-
vities greater than 1 $\$$. Typical results are shown in Fig. 4.1
and Fig. 4.2.

## 4.3 Outline of the Analysis

Inverse kinetics has been the basis to calculate from the changes
in power the net reactivity. A cell model has been used to calcu-
late temperature distributions in the fuel, clad, coolant and fuel
element boxes. The sum of the net reactivity and calculated feed-
back reactivity were required to remain constant after full ejec-
tion of the boron slug and before the delayed scram. For non
Doppler feedback reactivities calculated values were used. The
Doppler constant was adjusted so that the requirement was fullfil-
led. It was found that the Doppler constant was quite sensitive
to the assumed value of the average decay constant of the delayed
neutrons. This latter value could be determined using the sub- and
superprompt transients. The inserted reactivity should be a smooth
function of time during the ejection of the boron rod. This enabled
a determination of the reduced lifetime of the neutrons in the reac-
tor.

Fig. 4.1　Time Dependent Power for Sub-Prompt Transient 2
Core Ⅱ

Fig. 4.2   Core II   Super – Prompt Transient

## 4.4 Inverse Kinetics

The inverse kinetics equation for a point reactor model in the integral form is given by (2):

$$\rho_n = 1 - \frac{P_o}{P(t)} + \frac{\Lambda}{\beta} \cdot \frac{1}{P} \frac{dP(t)}{dt} - \frac{1}{P} \int_o^t (P(t) - P_o) \sum_{i=1}^{6} a_i \lambda_i e^{-\lambda_i(t-t')} dt' \qquad (1)$$

where  $\rho_n$  = net reactivity

P = power

$P_o$ = power at the beginning of the experiment

β = effective delayed neutron fraction

Λ = effective neutron lifetime

$a_i$ = effective fraction of the delayed neutrons of the group i

$\lambda_i$ = decay constant of the precursor nuclei of the group i

The net reactivity is the sum of the inserted (by $B_4C$ rod ejection) reactivity $\rho_{in}$ and the (negative) feedback reactivity $\rho_{fb}$. The first two terms of Eq. (1) form the familiar prompt jump approximation. The third term is the reactivity contribution due to the neutron lifetime, it is important only for superprompt tests. The last term is the contribution of the delayed neutrons.

The computer analysis of recorded power traces was based on Eq. (1) without introducing further simplifications.

## 4.5  Feedback Effects

The feedback is caused by the Doppler feedback and expansion effects
of the core. The core has been specifically designed to study the
Doppler effect and the expansion effects were therefore kept to a
minimum.

The calculation of expansion effects requires a temperature distri-
bution of all the components fuel, clad, coolant and fuel boxes. A
rough knowledge is however only required as the effects are small.
We have taken special care to obtain a fair approximation of the
time dependence of the expansion effects. We require as outlined in
4.4 that the sum of net reactivity and feedback reactivity remains con-
stant after full ejection of the boron rod. This means that errors
in the time dependence of the expansion effect will be compensated
by the Doppler coefficient, enhancing the errors in the Doppler de-
termination. The time constant for the fuel is fortunately of the
order of 30 sec. The fuel element boxes have however a time con-
stant of ca. 50 msec. Heating of the boxes results in bowing of
the fuel elements due to the clamping and results in a seizeable
reactivity effect. We have therefore made a cell calculation
(Fig. 4.3) with four zones: fuel, clad, coolant and structure.

The temperature distribution is given by

$$\frac{\partial T}{\partial t} = a \ \text{divgrad} \ T + \frac{Q}{c} \tag{2}$$

where $a = \frac{\lambda}{c}$ is the thermal diffusivity

$c$    is the heat capacity

$\lambda$    is the thermal conductivity

$Q$    is the deposited power density

| | | |
|---|---|---|
| Fuel | 11.3 mm | |
| Gap | 0.1 | |
| Clad | 1.0 | |
| Na | 2.7 | |
| Structure | 0.8 | |

Fig. 4.3    Dimensions used for Cell Calculations

The energy deposition in the different zones is given in Table II. Table III contains the volume fractions and specific weights. Table IV shows data used for heat capacity and thermal conductivity. For this evaluation a less refined model and more approximate values for the heat conductivity have been used than in the static tests.

In Fig. 4.4 we have depicted two temperature profiles. From the temperature distributions average fuel, clad, coolant and structural material temperatures were calculated. From the average temperature rise the time dependent non Doppler effects were calculated with the values of Table V.

## 4.6 Determination of the Average Decay Constant of the Delayed Neutrons

In order to simplify the argumentation, we consider simplifications (compare /10/ and /11/) of Eqs. (1) and (2). (In the actual evaluations the full equations have been used):

The approximation of the term $e^{-\lambda_i (t-t')}$ in the integrand of (1) by 1.0 introduces little error during the duration of the experiment ($\sim$ 300 msec) since the effective decay time of the delayed neutrons is about 2 sec. Eq. (1) simplifies to (3) if we further neglect the term $\frac{\Lambda}{\beta} \frac{1}{P} \frac{dP(t)}{dt}$, which is small compared to the other terms except during the short time period of superprompt criticality in the superprompt transient experiments.

Table II      SEFOR Energy Deposit in MeV/Fission

|  | Cell | Fuel | Steel | Na + $B_eO$ | Out- side Core |
|---|---|---|---|---|---|
| Kinetic energy fission products | 168. | 168. | -- | -- | -- |
| Prompt $\gamma$ from fission | 7.50 | 6.90 | .48 | .12 | -- |
| Prompt capture $\gamma$'s | 7.50 | 6.30 | .93 | .27 | -- |
| $\gamma$ from inelastic scattering | 3.27 | 3.09 | .15 | .03 | -- |
| Elastic scattering | .66 | .58 | .06 | .16 | 1.74 |
| Prompt sum | 186.93 | 184.87 | 1.62 | .58 | 1.74 |
| Delayed $\beta$'s | 7.0 | 7.0 | -- | -- | -- |
| Delayed $\gamma$'s | 6.0 | $\sim$ 5.0 | $\sim$ .70 | $\sim$ .30 | -- |
| Total | 199.93 |  |  |  |  |
| Prompt zone / total | .935 | .924 | .0081 | .0029 |  |

Table III    Volume Fractions and Specific Gravity

| | Vol fraction | Sp gr | Total mass | Fractional energy deposit / kg |
|---|---|---|---|---|
| Fuel | .432 | 10.0[+)] | 2150 | 429. x $10^{-6}$ |
| SS | .213 | 7.83 | 830 | .975 x $10^{-6}$ |
| Na + BeO | .352 | .97 | 171 | 17.5 x $10^{-6}$ |

[+)] .92 of theoretical density

Table IV     Heat Capacity and Thermal Conductivity

| Material | Heat capacity cal/9°C | Thermal conductivity |
|----------|----------------------|----------------------|
| Fuel | $.0669 + 1.125 \times 10^{-5} T$ [*] | $.015 - 1.47 \times 10^{-5} \times T + 4.96 \times 10^{-9} T^2$ [*] |
| Gap | -- | .286 |
| SS | .12 | .035 |
| Na | .305 | 45.0 |

[*]     Average fuel temperature has been used

Table V     Non Doppler Feedback Coefficients

| | |
|---|---|
| Axial clad expansion | .0013 $/°C |
| Radial core expansion | .0027 $/°C |
| Axial fuel expansion | .0002 $/°C |

Fig. 4.4    Temperature Rise in SEFOR I  Super 2

With this exception, a good approximation to Eq. (1) is:

$$\rho_n(t) = 1 - \frac{P_o}{P(t)} - \frac{\overline{\lambda}}{P(t)} \cdot E(t) \tag{3}$$

where $\overline{\lambda} = \sum_{i=1}^{6} a_i \lambda_i$ is the average decay constant of the delayed neutrons and

$E(t) = \int_0^t (P(t') - P_o)dt'$ is the transient energy

In an adiabatic situation Eq. (2) becomes:

$$\frac{\partial T}{\partial t} = \frac{Q}{c} \text{ or } T_x = T_{ox} + \frac{q_x}{c_x} \cdot E(t) \tag{4}$$

where $q_x$ is the heat deposit per MJ reactor energy per kg of the zone x

We assume furthermore that the Doppler feedback and the non Doppler feedback can be described by a linear function of the temperature rise

$$\rho_{fb} = E(t) \sum_x \frac{a_x}{c_x} = \gamma E(t) \tag{5}$$

where $\rho_{fb}$ = is the feedback reactivity

$a_x$ = is the feedback coefficient for the zone x.

The inserted reactivity is then given by

$$\rho_{in} = \rho_n(t) + \rho_{fb}(t) = 1 - \frac{P_o}{P} - \frac{\bar{\lambda}}{P} E + \gamma E \qquad (6)$$

This equation demonstrates that the evaluation of the Doppler energy coefficient $\gamma$ by analysis of $\rho_{in}$, P and E is complicated by the term $\frac{\bar{\lambda}}{P}$ E and that an error in $\bar{\lambda}$ leads to an error in $\gamma$.

Information about both, $\bar{\lambda}$ and $\gamma$ (or the related quantity $A_D$), can be obtained by analysis of different transient experiments with different ranges for $\bar{\lambda}/P$.

We combined four tests of Core II to increase the accuracy (Table VI). All tests had an initial power of 2 MW. Two data channels were used for each test. For each test $\gamma$ was determined for a fixed average decay constant. The results are shown in Fig. 4.5. We obtained an average decay constant for the delayed neutrons of $\bar{\lambda}$ = .625 ± .02. The calculated value using Keepin's data /6/, is $\lambda$ = .57 ± .06, which is about 10 per cent less than .625.

Fig. 4.5   Feedback Coefficient versus Average
           Decay Constant

Table VI    Tests Used for the Determination of
            the Average Decay Constant

| Test name | Max. reactivity |
|-----------|-----------------|
| B 1 | .5 $ |
| B 2 | .8 $ |
| S 1 | 1.12 $ |
| S 2 | 1.18 $ |

The uncertainty of the calculated value was estimated assuming
that we have only fissions in $^{239}$Pu and $^{238}$U, which is fairly
correct

$$\bar{\lambda} = ( 1 - \frac{F_8}{F_9 + F_8}) \frac{\beta_9}{\beta_{eff}} \cdot \bar{\lambda}_9 + \frac{F_8}{F_9 + F_8} \cdot \frac{\beta_8}{\beta_{eff}} \cdot \bar{\lambda}_8 \qquad (7)$$

where   $F_x$   =  is the fission rate of nuclide x

        $\bar{\lambda}_x$   =  is the corresponding average decay constant

        $\beta_x$   =  is the fraction of the delayed neutrons for
                      fissions in nuclide x

        $\beta_{eff}$   =  is the effective fraction of the delayed neu-
                          trons in the reactor

The decay constants of $^{238}U$ and $^{239}Pu$ are fairly different (Table VII).

Table VII        Effective Decay Constants

| $^{238}U$ | | | $^{239}Pu$ | | |
|---|---|---|---|---|---|
| $\lambda_i$ | $a_i$ | $a_i\lambda_i$ | $\lambda_i$ | $a_i$ | $a_i\lambda_i$ |
| 0.0132 | 0.13 | .00017 | .0129 | .038 | .0005 |
| .0321 | .137 | .00440 | .0311 | .280 | .0087 |
| .139 | .162 | .02252 | .134 | .216 | .0289 |
| .358 | .388 | .13890 | .331 | .328 | .1086 |
| 1.41 | .225 | .35955 | 1.26 | .103 | .1298 |
| 4.02 | .075 | .30150 | 3.21 | .035 | .1124 |
| sum | | .827 | | | .3888 |

Expression (7) leads to a $\overline{\lambda}$ uncertainty of 10 per cent when the estimation is based on the following partial uncertainties:

7% in the fission ratio

7% in the contribution of $^{239}$Pu to $\beta_{eff}$

7% in the effective decay constant for
the isotopes /6/.

## 4.7 Determination of the Reduced Neutron Lifetime 1/β

The inserted reactivity is known from static reactivity measurements
(compare /7/, p. 6 - 74) to be a smooth function of position and
hence a smooth function of time. When the inserted reactivity is cal-
culated from power traces of the transient experiments using Eq. (1)
and Eq. (6), it is fairly sensitive to the reduced neutron lifetime
in the region between 80 and 120 msec (Fig. 4.6). Visual inspection
of reactivity plots with different reduced neutron lifetimes gave
the following values.

|         | GfK             | ORNL noise /7/ |
|---------|-----------------|----------------|
| Core I  | .16 ± .01 msec  | .13            |
| Core II | .13 ± .01 msec  | .14            |

They compare well with the GE values but differ markedly from the
ORNL measurements. This might well be caused by the effects of the
reflectors on the lifetime. ORNL has measured at low power with the
reactor below critical. The transients were performed with nearly
all the reflectors in.

Fig. 4.6    Core I Transient No. 2
Effect of the Neutron Lifetime

## 4.8  Doppler Evaluation

The full Eq. (1) and the temperature model of 4.5 were used to de-
termine the Doppler coefficient. The temperature model described in
4.5 is more exact than the usual adiabatic approximation because it
includes effects of heat conduction on the development of the tem-
perature pattern in a fuel rod cell (compare Fig. 4.4). These effects
were found to be not negligible for the interpretation of the tran-
sient experiments.

The calculational procedure for the Doppler coefficient determina-
tion for each experiment included the steps:

- Temperature calculations for fuel rod cells for different
  amounts of transient energy depositions (to describe dif-
  ferent positions in the core) with the same time develop-
  ment of the energy deposition as derived from correspond-
  ing experimental power traces.

- Doppler- (based on an lnT law) and non Doppler feedback
  calculations.

- Flux and adjoint flux weighting of the results for dif-
  ferent positions in the reactor, where the energy depo-
  sition was assumed to be proportional to the flux.

- Comparison of the results with the results of feedback
  reactivity calculations obtained from Eq. (1) and appli-
  cation of a least square fit procedure for determination
  of the Doppler constant.

Table VIII presents the results of the Doppler constant determination for several transient experiments in Core I and Core II.

Table VIII          Doppler Constant Determination

| Transient number | Initial power (MW) | $A_D$ | |
|---|---|---|---|
| Core I | | | |
| 1 | 2 | 2.60 | |
| 2 | 2 | 2.52 | |
| 4 | 5 | 2.60 | 2.67 ± .2 |
| 5 | 5 | 2.67 | |
| 6 | 10 | 2.82 | |
| 7 | 10 | 2.81 | |
| Core II | | | |
| 1 | 2 | 1.84 | |
| 2 | 2 | 1.74 | 1.82 ± .2 |
| 4 | 5 | 1.88 | |

## 4.9 Error Analysis and Discussion

Table IX summarizes the results of an error analysis for the deter-
mination of the Doppler constant from transient experiments. Our re-
sults are comparable to the analysis results of GE and HEDL /8/. It
should however be noted that the agreement is partially accidental as
our analysis differs in several details. Some of the more important
differences between GE and our analysis are given in Table X. Special
care has therefore to be taken should these results be used to nor-
malize predictions of the Doppler effect that the same methods are
used as given in this chapter /9/.

Table IX    Results of uncertainty[+] estimates for
the evaluation of the Doppler coeffi-
cient from transient experiments

| | |
|---|---|
| Noise of signal and non linearity of amplifiers | 4% |
| Absolute power | 5% |
| Non Doppler feedback | 5% |
| Decay of delayed neutrons | 7% |
| Specified heat of the fuel | 8% |
| Global temperature weighting | 4% |
| Thermal conductivity and gap coefficient | 3% |
| $\beta_{eff}$ | 10% |
| Total | 16% |

[+]   estimated for a confidence level of $\sim$ 90%

Table X          Model Differences for Doppler Coefficient
                 Determination

| | KA | GE | $\delta A_D$ |
|---|---|---|---|
| Prompt energy released in the fuel | 92.4% | 88% | - 5% |
| Average decay constant for the delayed neutron precursors | .625 | .572 | + 8% |
| Linearity between Doppler and non Doppler feedback | no | yes | - 8% |
| lnT weighting over the pellets | no | yes | - 4% |
| Difference in reference temperature at the beginning of the transient | | | 2% |

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5. Analysis of Oscillator Experiments

This section summarizes results of the analysis of SEFOR oscillator experiments and experiences obtained during the analysis. A detailed description of dynamical models developed for and used in the analysis is published elsewhere /1/, /2/.

Conventional oscillator experiments (C.O.E.) and two kinds of balanced oscillator experiments have been performed one with the coolant temperature (I.B.O.E.) and the other with the reactor power kept constant (II.B.O.E.).

The analysis of the I.B.O.E. showed that a better interpretation of the measured data was possible by taking into account the temperature dependence of the fuel to gap heat transfer coefficient $h_g$. This can be clearly demonstrated with the transfer function between coolant flow and power. The fit to the experimental points, plotted in Fig. 5.1, is much better in the case in which the temperature dependence of $h_g$ (K = 0.3) was taken into account than in the case in which this dependence was neglected (K = 0).

The fuel time constant $\tau$ has been determined from the transfer function between power and reactivity. The results are in good agreement with corresponding calculations as shown in Table I. This table includes also the data evaluated by GE.

The results of the B.O.E. agree very well, the small differences being probably due to the different models. The different calculated results are due to the different evaluation methods. The bigger value results from a thermal resistance calculated with a linear interpolation, whereas the smaller value is obtained by the using the tangent for the thermal resistance. The latter method is

1st B.O.E.

Transfer function between the coolant flow and the power

$P_0 = 10$ MW

K = 0,3

K = 0

Phase

ω (rad/sec)

Fig. 5.1

Table I        Fuel Time Constants Evaluated from
               Oscillator Experiments

| $P_o$ (MW) | $\overline{\Theta}_o$ (°F) | Fuel time constant (sec) | | | | |
|---|---|---|---|---|---|---|
| | | GE | | | GfK | |
| | | Conv. | B.O.E. | Calc. | B.O.E. | Calc. |
| 10 | 560 | | 27.1 | | 27.7 | 26.7 |
| 19 | 760 | 24.5 | 22.1 | 26.5 | 19 | 19.7 |

GE        $\tau = \dfrac{T_{av} - \Theta_o}{P_o} \; V_f \, \rho \, c$

GfK       $\tau = \dfrac{\delta \, T_{av}}{\delta \, P_o} \; V_f \, \rho \, c$

$T_{av}$ = average fuel temperature
$\Theta_o$ = average sodium temperature
$V_f$ = volume of fuel
$\rho$ = density of fuel
$c$ = heat capacity of fuel

more correct and agrees much better with the I.B.O.E. However the
results of the C.O.E. remain still different from those of the
I.B.O.E. and of the calculations. A similar situation is reflected
in the results of the Doppler power coefficient shown in Table II.

This table compares values for the Doppler power coefficient eva-
luated by GE and GfK.

Table II    Doppler Power Coefficient Results

| $P_o$ (MW) | $\overline{\Theta}_o$ (°F) | Doppler power coefficient ($\cent$/MW) | | | | |
|---|---|---|---|---|---|---|
| | | GE | | | GfK | |
| | | Conv. | B.O.E. | Calc. | B.O.E. | Calc. |
| 10 | 560 | | 9.65 | | 9.55 | 9.3 |
| 19 | 760 | 4.2 | 4.9 | 5.1 | 4.8 | 4.9 |

From a comparison of the static experiments, the C.O.E. and the
I.B.O.E. it appears very clearly that the results of the I.B.O.E.
agree with those calculated from the static tests much better than
with those from the C.O.E. This may be explained by the existance
of a reactivity structure coefficient with a very long time con-
stant which has already been attenuated at the frequencies used
for the oscillator experiments.

Since the coolant temperature was kept constant, both in the I.B.O.E.
and in the static tests, the results of these tests are not affected

by this hypothetical structure coefficient, whereas the conventional
oscillator experiments are affected. The structure coefficient, used
in the analysis of the C.O.E., was obtained from the static tests.
As no attenuation was included in the model, this effect was prac-
tically overcompensated in the analysis. Qualitatively this conclu-
sion holds also for the results of the II.B.O.E.

During the 2nd B.O.E. it was first tried to keep the power constant
by oscillating the reactivity and the primary flow. However it was
observed that, at low frequencies, inlet and outlet temperatures
were in opposition of phase so that the amplitude of the reactivity
oscillations was very small and the Doppler coefficient could not
be evaluated in this way. This effect is due to the delay constants
of the primary loop. This result is confirmed by the calculations
carried out with the SEFAN code (Fig. 5.2). Here the amplitude of
the average reactor coolant temperature $((T_i+T_o)/2)$ is shown as a
function of the frequency. This amplitude is low at lower frequencies
and high at higher frequencies, due to the different phase between
outlet and inlet temperature. It was then tried to keep the inlet
coolant temperature constant by oscillating the secondary flow.

However, due to the large thermal capacity of the structure between
core outlet and reactor outlet, the temperature oscillations at reac-
tor outlet are strongly attenuated and this effect did not allow a
good control of the reactor inlet temperature by means of the secon-
dary flow.

All results of the oscillator experiments are of general importance
because of their direct relation to general questions of reactor
stability and plant dynamic behaviour. The analysis of the C.O.E.
and the B.O.E. showed that the plant dynamic behaviour including
the coolant systems is well understood. This confirms the accuracy
of the analytical model and the calculational methods used and gives
confidence in its application to other LMFBR's.

Fig. 5.2

## References for Section 5

/1/ L. Caldarola, P. Ferranti, F. Mitzel

Fast Reactor Transfer Functions with Special Reference
to the Nonlinearities and to the Spatial Dependence of
the Heat Transfer Process

KFK-2027 (1974) (Topical Report)

/2/ L. Caldarola, P. Ferranti, F. Mitzel

Fast Reactor Transfer Functions with Special Reference
to the Nonlinearities and to the Spatial Dependence of
the Heat Transfer Process

(to be published in Nuclear Technology)

# 6. Nuclear Doppler Calculations

## 6.1 Summary

Calculated results for $A_D = -T \, d\rho/dT$ are

$$A_D = 8.4 \cdot 10^{-3} \quad \text{for Core I}$$

$$\text{and} \quad A_D = 6.3 \cdot 10^{-3} \quad \text{for Core II.}$$

These numbers were obtained with the MOXTOT cross section set. KFKINR results are lower by about 5 per cent.

Agreement with experimental numbers is within the range of experimental uncertainty, with a slight preference for the KFKINR set.

Numerical studies demonstrate the importance of sufficiently detailed geometrical models and of the application of several corrections for Doppler calculations. Several questions are discussed concerning the calculation of the temperature dependence of effective group cross sections.

## 6.2 Studies Concerning the Temperature Dependence of Effective Cross Sections

The calculation of effective resonance capture and fission cross sections for the temperature range in question is one of the principle steps in nuclear Doppler calculations.

The straight forward way to perform this step with the present GfK code system is the use of the interpolation routine /1/ for shielded cross sections (f-factors), which are tabulated for the temperatures 300°K, 900°K, and 2100°K.

Questions arise about possible errors of the interpolation procedure and about possible errors in the calculation of self-shielding factors at the reference temperatures.

Concerning the first question, comparison calculations were performed for the Doppler coefficient of SEGOR 1-D (reactivity differences based on perturbation theory) with

      a)   using the interpolation routine /1/, and

      b)   using the DOPRO /2/ code

with the major difference, that b) avoids errors of interpolation. The difference of the results was found to be less than 2 per cent (.00785 for b) and .00770 for a)), which is considered to be negligible compared to other possible errors.

Regarding the second question, information was gained by a comparison between DOPRO and RABID /3/ calculations; with the same input of nuclear resonance parameters. The methodical difference is, that DOPRO is based on the "Narrow Resonance" (NR) approximation, while the more elaborate RABID code numerically, and more generally, treats the more complicated moderation effects occuring mainly in wider resonances.

Table I permits a comparison of results /4/ for individual $^{238}U$ resonances:

Table I    Comparison of NR with RABID Results for Effective
$^{238}U$ Capture Cross Sections


(Differences Between Resonance Integral Contributions $\delta RI$ for $500^\circ K$ and $1100^\circ K$ in $10^{-3}$ Barns)

| $E\ [ev]$ | $\Gamma_n\ [ev]$ | $\delta RI\ (500^\circ K\ -\ 1100^\circ K)$ | | Relative Difference |
| --- | --- | --- | --- | --- |
| | | NR | RABID | |
| 1802 | 0.670 | 0.7 | 0.73 | 4% |
| 1710 | 0.646 | 2.0 | 2.1 | 5% |
| 1255 | 0.236 | 2.6 | 2.77 | 6.5% |
| 793 | 0.0024 | 0.50 | 0.53 | 6% |
| 697 | 0.057 | 9.5 | 10.0 | 5% |
| 584 | 0.026 | 12.4 | 12.73 | 3% |

Condensed information is presented in Table II for those energy groups, which predominantly contribute to the Doppler effect.


Table II    Comparison of NR with RABID Results
for Three Important Energy Groups

| | $\delta\sigma_c\ (500^\circ K\ -\ 1100^\circ K)$ $[b]$ | |
| --- | --- | --- |
| Energy Group | DOPRO | RABID |
| 2.15 — 1.0 keV | 0.103 | 0.105 |
| 1.0 — 0.465 keV | 0.184 | 0.202 |
| 0.465 — 0.215 keV | 0.209 | 0.233 |

It can be estimated from Table I and Table II (by extrapolating the results of these tables to other, less important energy groups), that a "non-NR") correction of about 5% should be applied to calculated NR results for the Doppler coefficient for SEFOR.

An other study using the DOPRO /2/ code gives information about the relation between uncertainties of the SEFOR Doppler coefficient and uncertainties of the p wave strength function $S_1$. The range of numerical values in discussion is about $1.5 \cdot 10^{-4} < S_1 < 2.5 \cdot 10^{-4}$. The SEFOR Doppler coefficient was found to be 9 per cent higher for $S_1 = 2.5 \cdot 10^{-4}$ than for $S_1 = 1.5 \cdot 10^{-4}$.

Local fuel Doppler reactivity effects are usually calculated for the average temperature of the fuel rod section in question without explicitly taking into account the radial temperature profile across the fuel rod. This approximation was checked for SEFOR rods by comparative RABID calculations. Effective $^{238}U$ capture cross section differences $\Delta\sigma_{c8}$ were calculated for temperature transitions (averaged over the rod cross section) from $500^{\circ}K$ to $1000^{\circ}K$ with

a) specially constant temperature across the rod, and
b) a parabolic temperature profile with the same average temperatures as in a).

The results for three important energy groups are shown in Table III.

Table III      Comparison of $\Delta\sigma_{c8}$ for Flat and Parabolic Temperature Profiles

| Energy Range keV | $\Delta\sigma_{c8}$ [barns] | |
|---|---|---|
| | without T profile | with parabolic T profile |
| 2.15 - 1.0 | .092 | .091 |
| 1.0 - 0.465 | .176 | .175 |
| 0.465 - 0.215 | .200 | .197 |

The results of Table III demonstrate that errors due to the flat
temperature approximation can be neglected in the case of SEFOR
rods.

## 6.3 Variation of Doppler Results with Geometrical Models

The interpolation routine /1/ was used and perturbation theory was
applied to obtain the Doppler coefficients listed in Table IV for
differently detailed geometrical models of SEFOR I-D. Flux and
adjoint distribution were calculated with MOXTOT cross sections in
diffusion theory approximation.

Table IV:    Doppler Results for Different Geometrical Models

| Geometrical Model | $A_D = - T \, d\rho/dT$ |
|---|---|
| Sperical model | .00610 |
| Cylindrical model (with vertical bucklings) | .00670 |
| Detailed r-z model | .00686 |

It can be seen that a rather detailed geometrical model of the
reactor is required for accurate Doppler calculations.

## 6.4 Comparison of Results for Different Cross Section Sets

The comparison was based on one dimensional diffusion calculations for a spherical model /5/ of SEFOR. The MOXTOT set has been chosen arbitrarily as a basis for the comparison, the results of which are shown in Table V.

Table V:    Relative Differences in Calculated Doppler Coefficients
            for Different Cross Section Sets

| Set | $\Delta A_D / A_D$ |
|---|---|
| MOXTOT | (basis) |
| KFKINR | - 5 % |
| NAPPMB | - 5 % |
| ABN (GfK) | - 11 % |
| ABN (1964) | + 21 % |
| ENDF/B-II | + 15 % |
| MOD.ENDF/B-II | + 12 % |
| FTR III | + 35 % |

The Doppler coefficient results for the last 5 sets were taken from /5/. It can be seen that there are remarkable differences in the Doppler results for some of the sets in the table. The big difference between ABN (GfK) and the older (1964) set is essentially caused by different nuclear data for $^{239}$Pu.

## 6.5 Errors Introduced by the Perturbation Theory Approximation

The perturbation theory approximation neglects second order effects on the flux and adjoint distribution caused by the transition from "unperturbed" to "perturbed" cross sections. Corresponding errors can be avoided by successive criticality calculations for some discrete temperature levels.

Doppler effect calculations for SEFOR 1-D have been performed with both methods. Flux and adjoint distributions were based on one dimensional (radial) diffusion calculations. The results are well fitted by

$$T \, d\rho/dT = - A_D (T/T_o)^{-\alpha} \tag{1}$$

where $d\rho/dT$ is the temperature (T) derivative of reactivity $\rho$ , $T_o$ is a reference temperature, $A_D$ is the Doppler coefficient at $T_o$ and $\alpha$ is an exponent which in the case $\alpha \neq 0$ describes the deviation of the calculated results from a simple $\ln T$ - law for $\rho$. $T \, d\rho/dT$ is close to $-A_D$ when T is close to $T_o$. Eq. (1) permits the translation of $A_D$ from $T_o$ to other reference temperatures.

The calculations yielded $A_D = .0070$ (as expected for both methods of $\Delta\rho$ calculations) and

$\alpha = 0$     for the perturbation theory results,

$\alpha = 0.08$ for successive reactivity calculations

## 6.6 Additional Corrections

Perturbation theory applied to two dimensional (r,z) flux and adjoint distributions yielded $A_D$ = .0083, when REMO corrected (compare section 2.2) MOXTOT cross sections were used. The corresponding result without REMO correction is $A_D$ = .0075. Both values refer to core I-D and to a Doppler reference temperature $T_o$ = 300 °K.

A transport correction was estimated by comparison of one dimensional radial and axial $S_8$ calculations with radial and axial diffusion calculations. (Flux weighted transport cross sections were used in the $S_8$ calculations.) The comparison led to a transport correction of +5.4 % for the Doppler coefficient.

A heterogeneity correction of + 6% for core I-D and +5 % for core II-A for the Doppler coefficient $A_D$ was derived from radial diffusion theory Doppler calculations for core I-D with

   a) heterogeneity corrected ZERA cross sections for the actual cell geometry, and

   b) ZERA cross sections with the cell dimensions reduced by a factor $10^3$ ("quasi homogeneous" ZERA cross sections).

## 6.7 Comparison of Calculated with Experimental Results

Table VI summarizes the calculated results for the Doppler temperature coefficient $A_D$ for core I-D and core 2-A for the MOXTOT set.

Table VI: Results of Nuclear Doppler Temperature Coefficient Calculations

| Calculational Step | $A_D$ | |
| --- | --- | --- |
| | I-D | 2-A |
| 2 Dim. Diffusion, Perturbation Calculation, Including REMO Correction, $T_o$ = 300 °K | .0083 | .0062 |
| Correction for Extrapolation to $T_o$ = 1000 °K | -.0008 | -.0006 |
| Heterogeneity Correction | +.0005 | +.0003 |
| Transport Correction | +.0004 | +.0004[*] |
| Corrected Result | .0084 | .0063 |

[*] Assumed to be the same as calculated for I-D

The extrapolation to 1000 °K was performed, because this temperature is more representative for the experimental conditions in most of the SEFOR Doppler experiments.

The corrected results should be compared with the experimental results for $A_D$ described in sections 3 and 4, which are listed in Table VII together with GE evaluation results / 6 /.

Table VII:  Doppler Temperature Coefficient Evaluations
            from Static and Transient Experiments

|  | Core I | | Core II | |
|---|---|---|---|---|
|  | GfK | GE | GfK | GE |
| Static Power Experiments | .0073 | .0079 | .0058 | .0060 |
| Transient Experiments | .0084 | .0081 | .0058 | .0060 |
| Average | .0079 | .0080 | .0058 | .0060 |
| Uncertainty of Evaluation | $\pm$.0012 | | $\pm$.0009 | |

It can be seen that calculated an evaluated Doppler temperature
coefficients agree within the range of uncertainty of the evaluated
numbers for both cores. A slight systematical overestimation (5 to 10 %)
of $A_D$ by MOXTOT calculations can be recognized. It can be expected from
Table V that the overestimation would be reduced to the range 0 to 5 %
in the case of the KFK INR set.

The calculated results as shown in Table VI do not contain the
"non-NR" correction mentioned in section 6.2. This correction would
increase the calculated coefficients by about 5 %.

The indicated overestimation of the Doppler temperature coefficient
is in line with the overestimation of $1/\beta_{eff}$ and of fuel and boron
reactivity worths (compare section 2). An increase of $\beta_{eff}$ by about
5 % would generally improve the agreement between calculated and
measured SEFOR parameters.

## References for Section 6

/1/     H. Huschke

Gruppenkonstanten für dampf- und natriumgekühlte
schnelle Reaktoren in einer 26-Gruppendarstellung

KFK-770 (1968)


/2/     E.A. Fischer

Interpretation von Dopplerproben-Messungen in schnel-
len kritischen Null-Energie-Anlagen

KFK-844 (1969)


/3/     P.H. Kier, A.A. Robba

RABBLE, A Program for Computation of Resonance Absorp-
tion in Multiregion Reactor Cells

ANL-7326 (1967)


/4/     H. Bogensberger, E.A. Fischer, A.M. Raberain, D. Wintzer

Dopplerrechnungen für schnelle Reaktoren - Standardver-
fahren und notwendige Korrekturen

Reaktortagung Hamburg (1972)


/5/     O'Dell

Calculation of the Isothermal Doppler Coefficient
for SEFOR

(Personal communication)


/6/     L. Caldarola et al.

SEFOR Experimental Results and Application to LMFBR's

Proc. of the Int. Conf. on Engineering of Fast Reactors,
Karlsruhe, Oct. 9 - 13, 1972

## 7. Summary and Conclusions

The analysis of the SEFOR zero power experiments leads to ratios
between calculated and experimental results, which mostly are con-
sistent with experience obtained by the analysis of more simple
Plutonium fuelled configurations in zero power assemblies with the
cross section sets MOXTOT and KFKINR. Exceptions with more pro-
nounced differences between calculation and experiments are boron
rod reactivity worths and fission rate distributions in Core I.

Nevertheless, the conclusion is permitted that there are no major
errors in data and calculational procedures.

Doppler temperature coefficients derived from static power experi-
ments and from transient experiments are in good agreement with
GE evaluations, although different approximations have been used
for the evaluation. Ratios between nuclear Doppler coefficient
calculations and experimental numbers are in the range between
1.0 and 1.1 (for the KFKINR set and the MOXTOT set).

The transfer of experience obtained by SEFOR Doppler effect analysis
to predictions for other LMFBR's requires the use of data and cal-
culational procedures as described in this report, if the predic-
tions should be based on the MOXTOT set or on the KFKINR set. It is
suggested to use the Doppler temperature coefficient as calculated,
with the only modification that the calculated effective fraction
of delayed neutrons should be multiplied by 1.05. In the case of
SEFOR, this procedure would reduce the difference to experimental
Doppler coefficient data to unsignificant amounts for both cross
section sets.

Error estimates as described in /1/ indicate that this procedure of extrapolation to other LMFBR's would limit the uncertainty of the predicted Doppler power coefficient and of the Doppler energy coefficient to $\pm$ 25% with a confidence level of about 90%.

A consistent transfer of the SEFOR analysis results for Doppler effect predictions in transient situations requires a correction of the calculated average decay constant of delayed neutrons and the inclusion of non-adiabatic effects in temperature calculations.

Reference for Section 7

/1/       W.J. Oosterkamp, D. Wintzer
          Übertragbarkeit der SEFOR-Dopplerergebnisse auf den SNR
          KFK-1272/4 (1972), section 1216 SEFOR

A p p e n d i x   A

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Geometrical Models and Material Densities for

Neutrons Physics Calculations

Models for Two Dimensional Reactor Calculations

Figures 2.1 through 2.3 show the core loading pattern of Ass. 1-C,
1-D and 2-A. Corresponding two dimensional geometrical models of
material distributions are shown in Fig. A1 and Fig. A2 for Ass. 1-C
and 1-D.

Comments

Regions O, P and Q represent the central channel with slightly vary-
ing average steel concentrations along the channel axis. Regions A
and B represent the regular core lattice with slightly lower BeO con-
centration and slightly higher steel concentration in B than in A.
The core regions C and D differ from A and B because of the slightly
(4%) lower Plutonium enrichment in the central 52 fuel rods. Regions
E and F (in Ass. 1-D) differ from regions A and B because of the

# FIG. A1  SEFOR - IC
## GEOMETRICAL MODEL
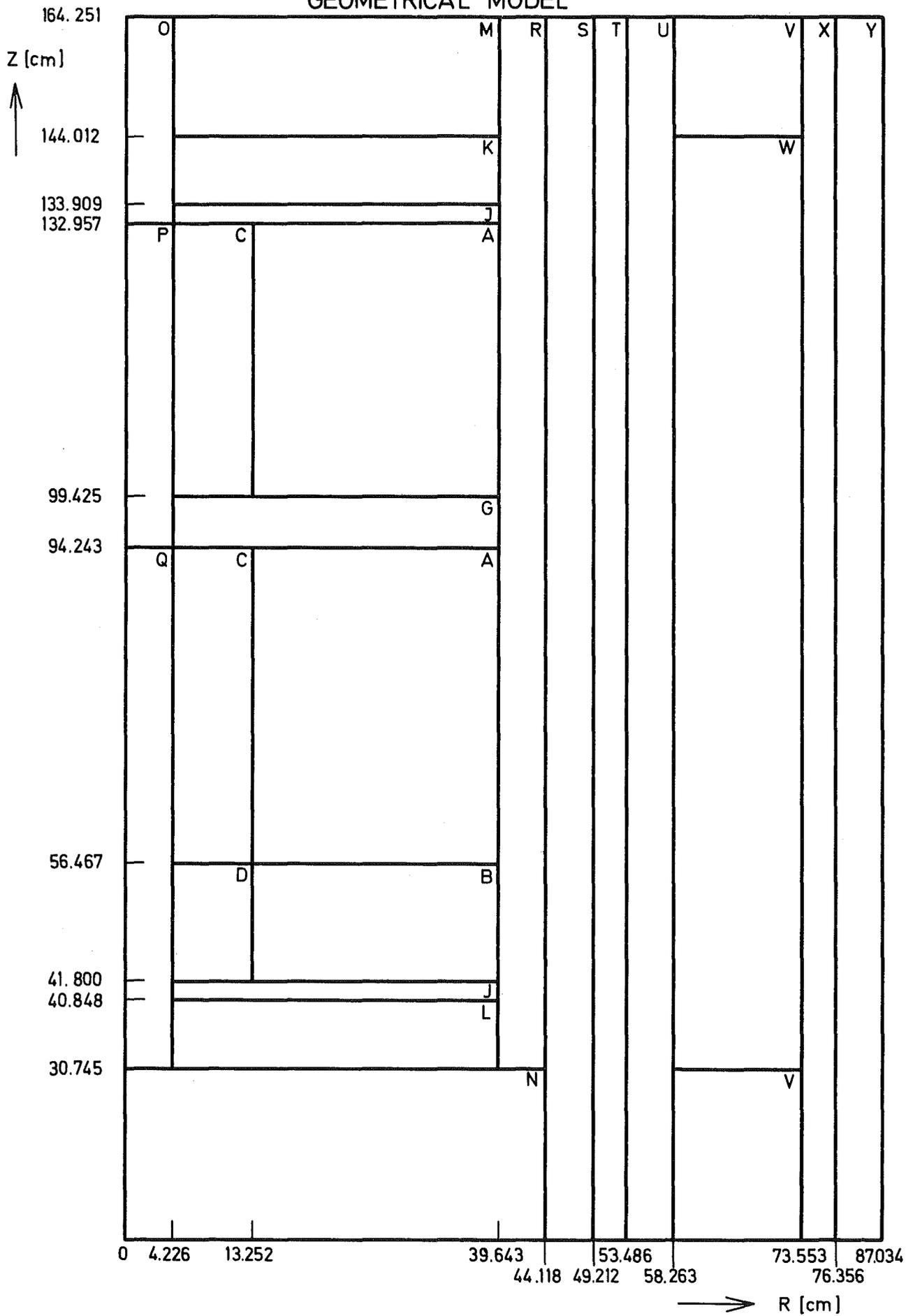
FIG. A2   SEFOR - ID
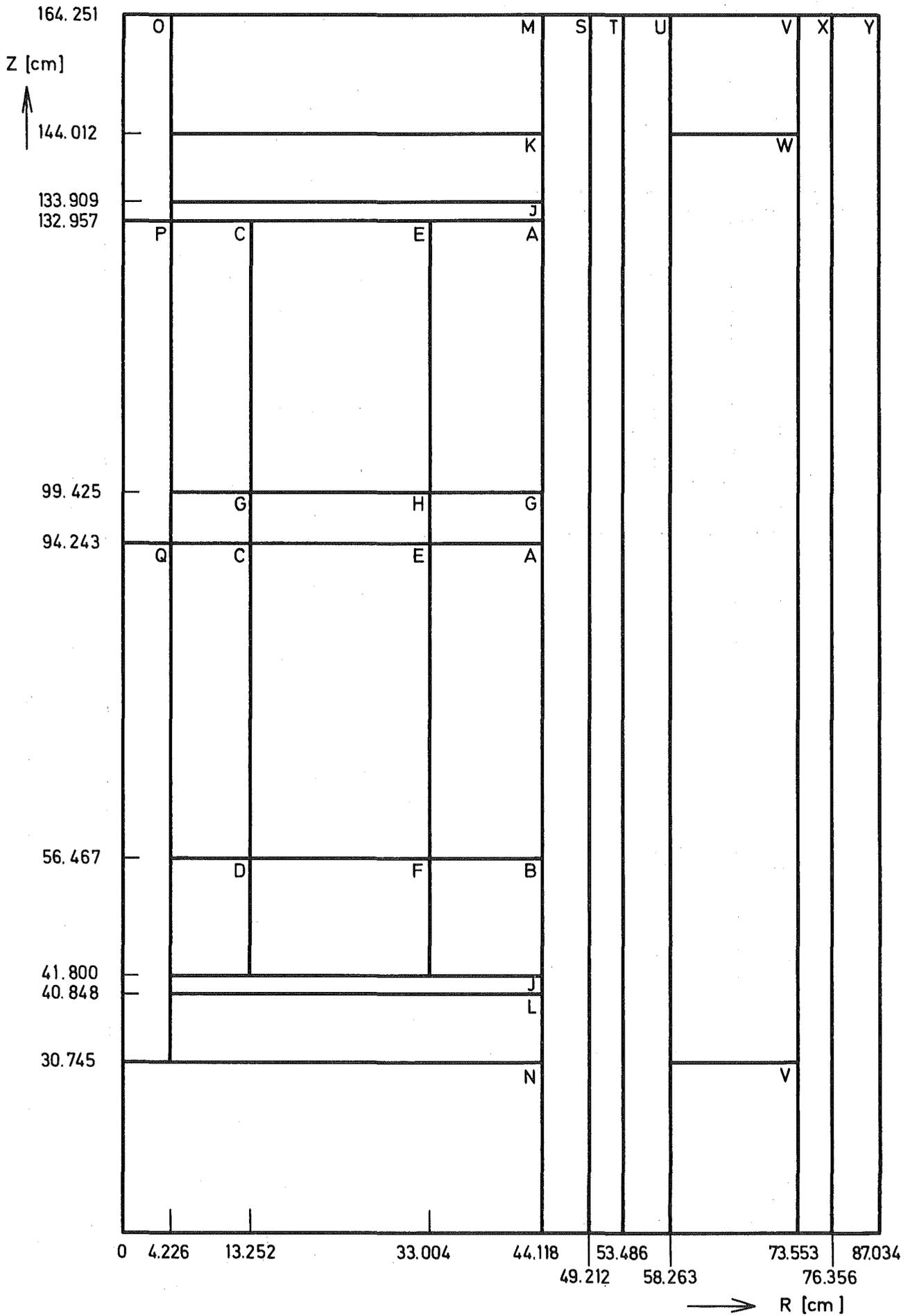
presence of the boron absorber rods in this region. The upper and lower parts of the core are separated by the gap regions G and H which essentially contain the springs of the expansion gaps and the $UO_2$ insulator pellets (compare Fig. A3). Regions I and J are for the upper and lower insulator pellets of the fuel rods. Nickel reflector regions are K, L (axial) and W (radial).

Region R represents that part of the core channel structure which was loaded with fuel elements in Ass. 1-C. The core edge plates, the shell around the core, torque rods and the corresponding volume of Na are homogenized in region S. Regions T and U represent the material of the, and around the, inner and outer vessel. Regions M, N, V, X and Y roughly describe various material structures beyond the axial and radial Ni reflectors.

The core composition of Ass. 2-A differs from Ass. 1-D because of the insertion of steel rods instead of the BeO rods (compare Fig. A4) to produce a harder neutron spectrum. In addition, Ass. 2-A contains 7 boron rods instead of 14 in Ass. 1-D.

Atom densities of the reactor regions are given in Table I. In addition to the regions shown in Figs. A1 and A2, Table I contains atom densities for additional regions above ($M^+$) and below ($N^+$, $N^{++}$) regions M and N, respectively. The axial extensions of these regions are:

$M^+$     125.5 cm

$N^+$     40.4 cm

$N^{++}$     14.0 cm

Comparative criticality calculations yielded a reactivity increase of $\Delta k$ = .0014 due to addition of these regions. Similar calculations showed a reactivity increase of $\Delta k$ = .0013 for a transition from a more simple 4 region model for the radial material distribution beyond the core radius to the 6 region model (S, T, U, W, X, Y) as shown in Figs. A1 and A2.

FIG. A3   SEFOR  FUEL  ROD

REFLECTOR

INSULATOR

FUEL

INSULATOR

INSULATOR

FUEL

INSULATOR

REFLECTOR

49-5/8 IN.

| CLADDING  O.D. | 0.970 |
| CLADDING  MATL. | STN. STL. |
| CLADDING  THICKNESS | 0.040 |
| CORE  LENGTH | 36.56 |
| FUEL  LENGTH | 33.81 |

CENTRAL BeO PELLETS
0.777 ±0.003 O.D.

FUEL PELLET
0.875 ± 0.001 DIA.

TIGHTENER SLEEVE
0.960 ±0.005 INSIDE

ACROSS FLATS X
0.035 ±0.004 WALL

CENTRAL BeO CLAD
0.875 ±0.002 O.D. X
0.040 ±0.002 SST WALL

SIDE ROD
SST 0.250 ± 0.001 DIA.

FUEL ROD CENTERS
LOCATED ON 2.190 DIA.

FUEL CLAD SST
0.890 ± 0.0015 I.D.
0.040 ± 0.002 WALL

3.150 ± 0.003
HEX. CHANNEL

0.060 ±0.002 WALL SST

3.160
HEX.
MEAN CELL

FIG. A4    SEFOR FUEL CHANNEL

## Cell Models for Heterogeneity Calculations

Most of the Karlsruhe calculations for SEFOR are done with heterogeneity corrected microscopic cross sections for the core regions, which are computed by the cell code ZERA.

Two region models are used to describe the lattice cell (see Figs. A4 and A5). Cell region 1 represents a fuel rod (without clad) and cell region 2 represents Na, 1/6 of a BeO rod and the rest of the structural material. The cross sectional area of the unit cell is 1/6 of the cross sectional area of an element cell.

Table II contains the atom densities in the two cell regions for the cell models used. Homogenization of the atom densities within the cell yields the atom densities of the corresponding compositions of Table I.

The correspondence is:

| Cell No | Homog. Composition |
|---------|--------------------|
| Z 1 | A |
| Z 2 | B |
| Z 3 | C |
| Z 4 | D |
| Z 7 (Z 5) | E |
| Z 8 (Z 6) | F |

In Z 7 and Z 8, the lattice distortions due to the presence of some $B_4C$ rods were approximated by slightly increasing (by a factor 22.05/21.05 - see Fig. A6) the unit cell area and by distributing the corresponding amount of $B_4C$ in cell region 2.

For comparison calculations, another type of cell models (Z 5 and Z 6) is used for the homogeneous compositions E and F. They describe a boron rod (see Fig. A7) in the center of the cell (cell region 1), the rest of the usual rod cell in cell region 2 and homogenized core material without $B_4C$ in a third cell region. The cross sectional area of Z 5 and Z 6 is 1/14 of the cross sectional area occupied by compositions E and F.
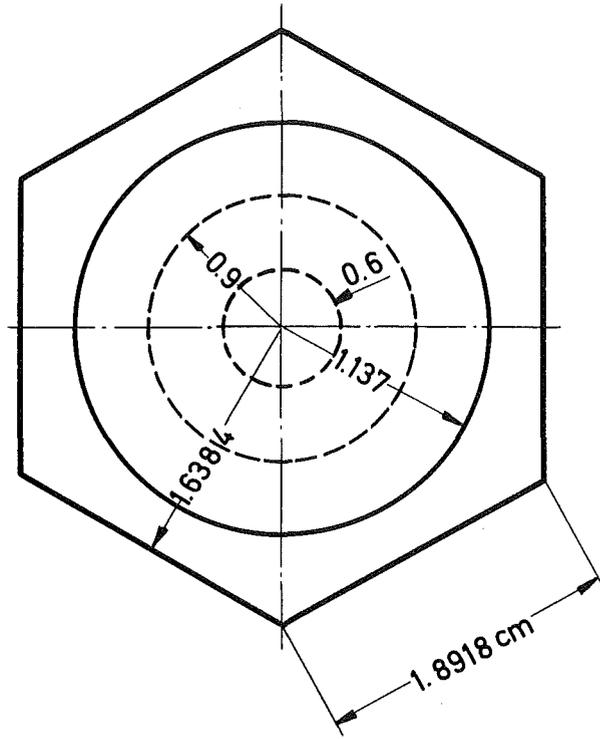
FIG. A5
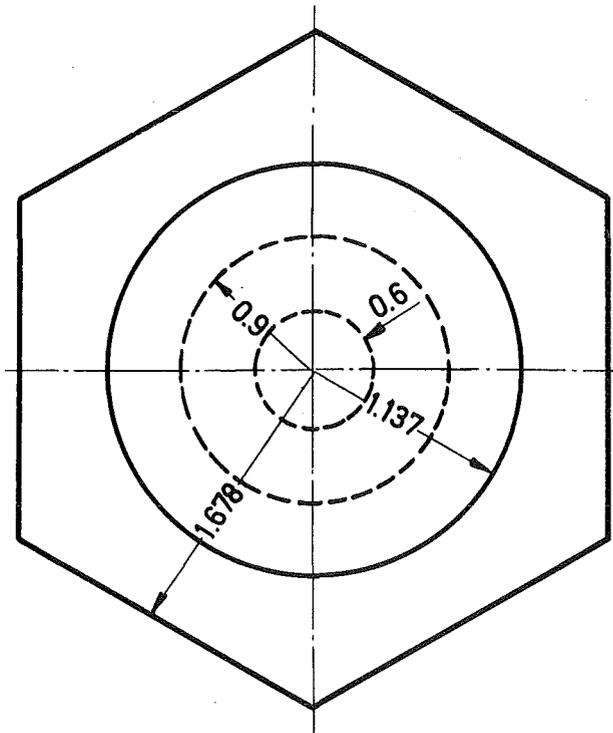


FIG. A6
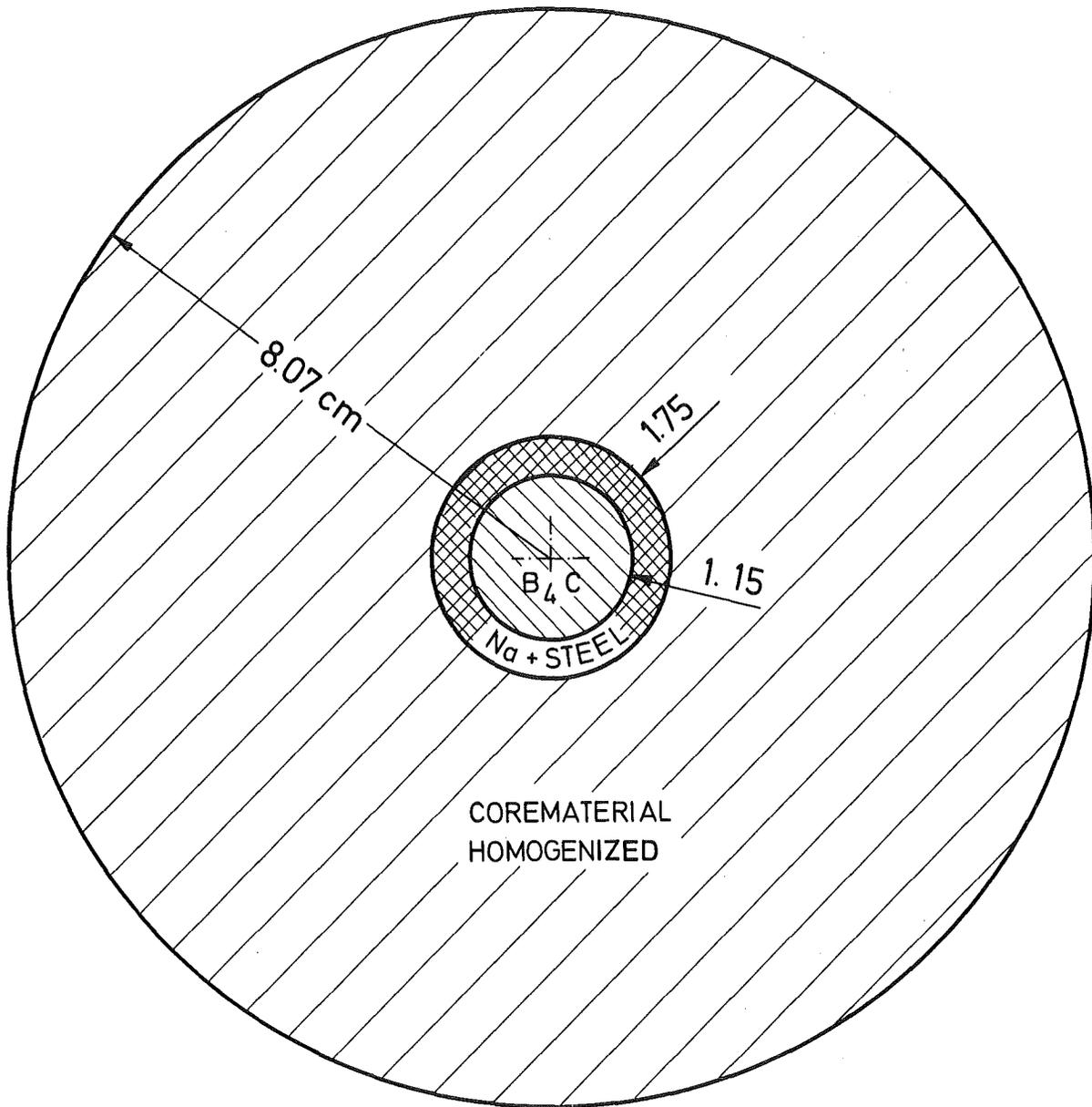
FIG. A7 CELL MODEL FOR BORON
ROD CELL CALCULATIONS

## TABLE I

### Atom Densities in SEFOR Compositions

| | A | B | C | D | E | F | G | H |
|---|---|---|---|---|---|---|---|---|
| Be | 3.7769-3 | 2.6840-3 | 3.7769-3 | 2.6840-3 | 3.7769-3 | 2.6840-3 | 3.7769-3 | 3.7769-3 |
| B-10 | – | – | – | – | 1.4402-4 | 1.4402-4 | – | 1.4402-4 |
| B-11 | – | – | – | – | 5.7968-4 | 5.7968-4 | – | 5.7968-4 |
| C | – | – | – | – | 1.8092-4 | 1.8092-4 | – | 1.8092-4 |
| O | 2.2146-2 | 2.1052-2 | 2.2146-2 | 2.1052-2 | 2.1313-2 | 2.0219-2 | 1.0742-2 | 1.0360-2 |
| Al | – | – | – | – | – | – | – | – |
| Na | 6.8099-3 | 6.8099-3 | 6.8099-3 | 6.8099-3 | 6.8099-3 | 6.8099-3 | 6.8099-3 | 6.8099-3 |
| Cr | 3.7929-3 | 3.9587-3 | 3.7929-3 | 3.9587-3 | 3.9322-3 | 4.0978-3 | 5.1147-3 | 5.1939-3 |
| Fe | 1.3368-2 | 1.3952-2 | 1.3368-2 | 1.3952-2 | 1.3846-2 | 1.4430-2 | 1.7330-2 | 1.7628-2 |
| Ni | 2.2297-3 | 2.3272-3 | 2.2297-3 | 2.3272-3 | 2.3011-3 | 2.3987-3 | 3.6265-3 | 3.6345-3 |
| Mo | 1.1915-4 | 1.2436-4 | 1.1915-4 | 1.2436-4 | 1.1915-4 | 1.2436-4 | 1.1915-4 | 1.1915-4 |
| U-235 | 1.6154-5 | 1.6154-5 | 1.6303-5 | 1.6303-5 | 1.5421-5 | 1.5421-5 | 7.6230-6 | 7.2773-6 |
| U-238 | 7.3328-3 | 7.3328-3 | 7.4005-3 | 7.4005-3 | 7.0002-3 | 7.0002-3 | 3.4578-3 | 3.3010-3 |
| Pu-239 | 1.7215-3 | 1.7215-3 | 1.6595-3 | 1.6595-3 | 1.6434-3 | 1.6434-3 | – | – |
| Pu-240 | 1.5506-4 | 1.5506-4 | 1.4948-4 | 1.4948-4 | 1.4803-4 | 1.4803-4 | – | – |

TABLE I    (continued)

Atom Densities in SEFOR Compositions

| | I | J | K | L | M | M$^+$ | N | N$^+$ | N$^{++}$ | O | P |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Be | 3.7705-3 | 2.6912-3 | - | - | - | - | - | - | - | - | - |
| B-10 | - | - | - | - | - | 6.579-3 | - | - | - | - | - |
| B-11 | - | - | - | - | - | 2.632-2 | - | - | - | - | - |
| C | - | - | - | - | - | 8.224-3 | - | - | - | - | - |
| O | 2.2748-2 | 2.1669-2 | - | - | - | - | - | - | - | - | - |
| Al | - | - | - | - | - | - | - | - | - | - | - |
| Na | 6.8099-3 | 6.8099-3 | 6.8759-3 | 6.9716-3 | 1.3526-2 | 9.83-3 | 1.081-2 | 2.40-2 | 3.43-3 | 1.6615-2 | 1.6615-2 |
| Cr | 3.7929-3 | 3.9587-3 | 3.9153-3 | 3.6o7o-3 | 5.9192-3 | 2.40-3 | 9.46 -3 | - | 1.51-2 | 3.176o-3 | 2.0624-3 |
| Fe | 1.3368-2 | 1.3952-2 | 1.3784-2 | 1.2732-2 | 1.8825-2 | 8.41-3 | 3.00 -2 | - | 5.2o-2 | 1.0894-2 | 7.0745-3 |
| Ni | 2.2297-3 | 2.3272-3 | 4.2413-2 | 4.306-2 | 3.1419-3 | 1.20-3 | 5.30 -3 | - | 7.77-3 | 1.6285-3 | 1.0575-3 |
| Mo | 1.1915-4 | 1.2436-4 | 1.29o9-4 | 1-1974-4 | 1.2872-4 | 5.48-5 | - | - | - | - | - |
| U-235 | 2.0670-5 | 2.0670-5 | - | - | - | - | - | - | - | - | - |
| U-238 | 9.4207-3 | 9.4207-3 | - | - | - | - | - | - | - | - | - |
| Pu-239 | - | - | - | - | - | - | - | - | - | - | - |
| Pu-240 | - | - | - | - | - | - | - | - | - | - | - |

TABLE I    (continued)

Atom Densities in SEFOR Regions

| | Q | R | S | T | U | V | W | X | Y |
|---|---|---|---|---|---|---|---|---|---|
| Be | – | – | – | – | – | – | – | – | – |
| B-10 | – | – | – | – | – | – | – | – | 6.63-3 |
| B-11 | – | – | – | – | – | – | – | – | 2.72-2 |
| C | – | – | – | – | – | – | – | – | 8.45-3 |
| O | – | – | – | – | – | – | – | – | – |
| Al | – | – | – | – | 1.1791-2 | – | 2.1303-3 | 2.7290-2 | – |
| Na | 1.6615-2 | 2.0790-2 | 1.4943-2 | 1.4496-2 | – | – | – | – | – |
| Cr | 5.3564-3 | 2.2978-3 | 6.4971-3 | 4.1208-3 | 2.7620-3 | 2.659-3 | 5.9730-4 | 1.0186-3 | 1.86-3 |
| Fe | 1.8373-2 | 7.882o-3 | 2.0600-2 | 1.4135-2 | 9.4740-3 | 9.093-3 | 2.o489-3 | 3.4939-3 | 5.73-3 |
| Ni | 2.7465-3 | 1.1782-3 | 5.4525-3 | 2.1129-3 | 1.4160-3 | 1.364-3 | 7.3936-2 | 5.2226-4 | 1.o1-3 |
| Mo | – | – | – | – | – | – | – | – | – |
| U-235 | – | – | – | – | – | – | – | – | – |
| U-238 | – | – | – | – | – | – | – | – | – |
| Pu-239 | – | – | – | – | – | – | – | – | – |
| Pu-240 | – | – | – | – | – | – | – | – | – |

# TABLE II

## Cell Compositions for Heterogeneity Calculations

| | CELL Z1 | | CELL Z2 | | CELL Z3 | | CELL Z4 | |
|---|---|---|---|---|---|---|---|---|
| | Region 1 | Region 2 | Region 1 | Region 2 | Region 1 | Zone 2 | Zone 1 | Zone 2 |
| Be | – | 6.5o1-3 | – | 4.62o-3 | – | 6.5o1-3 | – | 4.620-3 |
| O | 4.383-2 | 6.501-3 | 4.383-2 | 4.620-2 | 4.383-2 | 6.501-3 | 4.383-2 | 4.620-3 |
| Na | – | 1.172-2 | – | 1.172-2 | – | 1.172-2 | – | 1.172-2 |
| Cr | – | 6.529-3 | – | 6.814-3 | – | 6.529-3 | – | 6.814-3 |
| Fe | – | 2.3o1-2 | – | 2.402-2 | – | 2.301-2 | – | 2.402-2 |
| Ni | – | 3.838-3 | – | 4.006-3 | – | 3.838-3 | – | 4.006-3 |
| Mo | – | 2.051-4 | – | 2.141-4 | – | 2.051-4 | – | 2.141-4 |
| U-235 | 3.855-5 | – | 3.855-5 | – | 3.89o-5 | – | 3.89o-5 | – |
| U-238 | 1.750-2 | | 1.75o-2 | – | 1.766-2 | – | 1.766-2 | – |
| Pu-239 | 4.108-3 | – | 4.108-3 | – | 3.960-3 | – | 3.960-3 | – |
| Pu-240 | 3.700-4 | – | 3.700-4 | – | 3.567-4 | – | 3.567-4 | – |
| | | | | | | – | | – |

Cell Geometry : 1 (see Fig. 7A)

TABLE II    (continued)

CELL COMPOSITIONS FOR HETEROGENEITY CALCULATIONS

### CELL Z5

|        | Region 1  | Region 2 | Region 3  |
|--------|-----------|----------|-----------|
| Be     | -         | 6.501-3  | 3.7769-3  |
| B-10   | 7.5775-3  | -        | -         |
| B-11   | 3.0500-2  | -        | -         |
| C      | 9.5192-3  | -        | -         |
| O      | -         | 6.5o1-3  | 2.2146-2  |
| Na     | -         | 1.172-2  | 6.8o99-3  |
| Cr     | 7.3262-3  | 6.529-3  | 3.7929-3  |
| Fe     | 2.513o-2  | 2.3o1-2  | 1.3368-2  |
| Ni     | 3.7564-3  | 3.838-3  | 2.2297-3  |
| Mo     | -         | 2.o51-4  | 1.1915-4  |
| U-235  | -         | -        | 1.6154-5  |
| U-238  | -         | -        | 7.3328-3  |
| Pu-239 | -         | -        | 1.7215-3  |
| Pu-24o | -         | -        | 1.55o6-4  |

### CELL Z6

|        | Region 1  | Region 2 | Region 3  |
|--------|-----------|----------|-----------|
| Be     | -         | 4.62o-3  | 2.684o-3  |
| B-1o   | 7.5775-3  | -        | -         |
| B-11   | 3.0500-2  | -        | -         |
| C      | 9.5192-3  | -        | -         |
| O      | -         | 4.62o-3  | 2.1o52-2  |
| Na     | -         | 1.172-2  | 6.8o99-3  |
| Cr     | 7.3262-3  | 6.814-3  | 3.9587-3  |
| Fe     | 2.5130-2  | 2.402-2  | 1.3952-2  |
| Ni     | 3.7564-3  | 4.oo6-3  | 2.3272-3  |
| Mo     | -         | 2.141-4  | 1.2436-4  |
| U-235  | -         | -        | 1.6154-5  |
| U-238  | -         | -        | 7.3328-3  |
| Pu-239 | -         | -        | 1.7215-3  |
| Pu-24o | -         | -        | 1.55o6-4  |

CELL GEOMETRY: 3    (see Fig. 7C)

TABLE II        (continued)

CELL COMPOSITIONS FOR HETEROGENEITY CALCULATIONS

|        | Cell Z 7 | | Cell Z 8 | |
|--------|----------|----------|----------|----------|
|        | Region 1 | Region 2 | Region 1 | Region 2 |
| Be     | –        | 6.2953-3 | –        | 4.4737-3 |
| B-1o   | –        | 2.4005-4 | –        | 2.4oo5-4 |
| B-11   | –        | 9.6621-4 | –        | 9.6621-4 |
| C      | –        | 3.0156-4 | –        | 3.0156-4 |
| O      | 4.383-2  | 6.299-3  | 4.383-2  | 4.475-3  |
| Na     | –        | 1.1351-2 | –        | 1.1351-2 |
| Cr     | –        | 6.554-3  | –        | 6.83o2-3 |
| Fe     | –        | 2.3o79-2 | –        | 2.4o52-2 |
| Ni     | –        | 3.8355-3 | –        | 3.9932-3 |
| Mo     | –        | 1.986-4  | –        | 2.o728-4 |
| U-235  | 3.855-5  | –        | 3.855-5  | –        |
| U-238  | 1.75o-2  | –        | 1.75o-2  | –        |
| Pu-239 | 4.1o8-3  | –        | 4.1o8-3  | –        |
| P-24o  | 3.7oo-4  | –        | 3.7oo-4  | –        |

CELL GEOMETRY: 2 (see Fig. 7 B)

# A p p e n d i x   B

## Supplementary Studies for $B_4C$ Rods in SEFOR 1-D

## Flux Depressions in and Around Boron Rods

ZERA cell calculations for a representative boron rod containing core cell were performed for evaluation of neutron flux and reaction rate depressions in and around the SEFOR $B_4C$ rods. Cell geometry and material densities are shown in Fig. A7 and Table II (cell Z 5) of Appendix A. The ratio of core material over $B_4C$ in the cell corresponds to the average ratio in core region E of Ass. 1-D (see Fig. A2). Figures B1 and B2 show radial flux distributions for some neutron energy groups. Flux depressions in the vicinity of the $B_4C$ rods are substantial at high energies (because of lacking neutron sources) and at low energies (because of high B-10 absorption cross sections at low energies). About 2/3 of the boron absorption occurs in the energy range between 20 keV and 200 keV, where the flux depression is not substantial. The resulting reduction of the $B_4C$ absorption compared to a corresponding homogeneous case is 6 per cent.

## Effects of Asymmetrical B$_4$C Loadings in the Core

It was pointed out in section 2.3 that the B$_4$C rod loading in
Ass. 1-D was not quite symmetrical, and it was suspected that
this fact contributed to the overestimation of the reactivity
worths of B$_4$C rods and fuel rods in this assembly by the cal-
culations. An attempt was made to estimate the resulting asym-
metry effect by r- geometry diffusion calculations. Fig. B3
shows the model of material distribution for this study. The
boron containing core region extends from a radius of 13.252 cm.
The model represents 8 boron rods in the right half section of
the core (see Fig. B3) and 6 boron rods in the left half section.

The calculated angular distribution of the neutron flux in energy
groups 1 and 4 is shown in Fig. B4 for a radial position of 33 cm
(compare Fig. B3). Calculated angular distributions of fission
rates are shown in Figs. B5 through B7 for $^{238}$U, $^{239}$Pu, and $^{235}$U.
The square of the $^{239}$Pu fission rate distribution is a rough esti-
mate of the reactivity worth distribution and indicates a reacti-
vity worth difference in the order of 8 per cent between the left
hand side and the right hand side (Fig. B3) of the core at r = 33 cm.

The material distribution model for these calculations somewhat
overestimates the actual asymmetry of the B$_4$C loading in Ass. 1-D,
and a more exact description of the boron distribution would prob-
ably lead to a reactivity difference of about 6% instead of 8%. It
was mentioned in section 2.3 that this is the order of magnitude
of the asymmetry effect on reactivity worth measurements, which
was indicated by some experimental numbers. The effect is probably
one of the reasons for the more pronounced overestimation of ma
terial reactivity worths in Ass. 1-D than in Ass. 2-A (the latter
had a fairly symmetrical boron loading).

# FIG. B1 FLUX DISTRIBUTION AROUND B$_4$C RODS



GROUP 1
6.5 - 10 MeV

GROUP 8
0.1 - 0.2 MeV

HET
HOM

$\dfrac{\phi_n}{\phi}$    n = 1...7

RADIAL POSITION [CM] ⟶

# FIG.B2 FLUX DISTRIBUTION AROUND B$_4$C RODS



GROUP 14
1.0 - 2.15 KeV

GROUP 17
100 - 215 eV

——— HET
– – – HOM

RADIAL POSITION [CM] ⟶
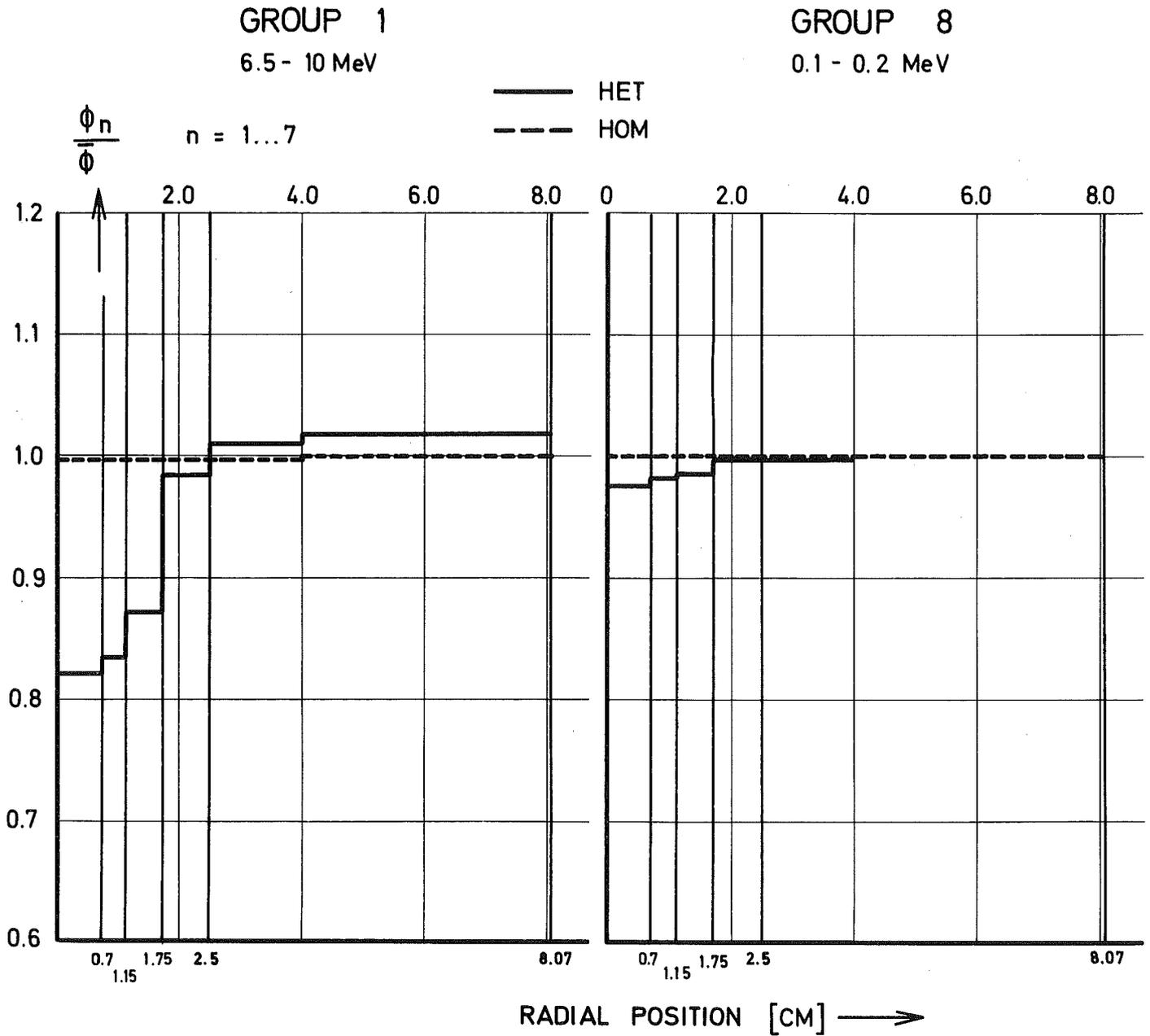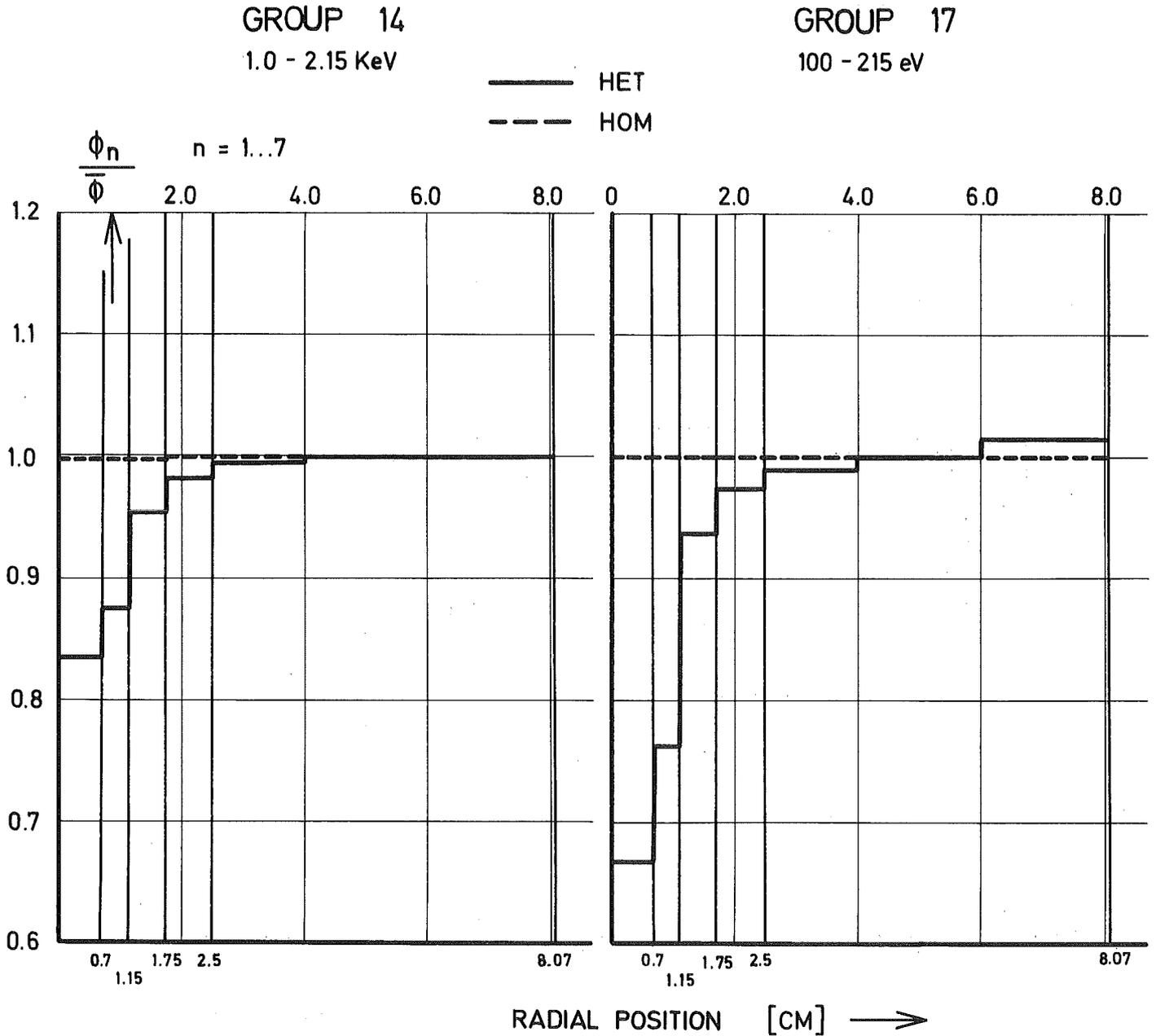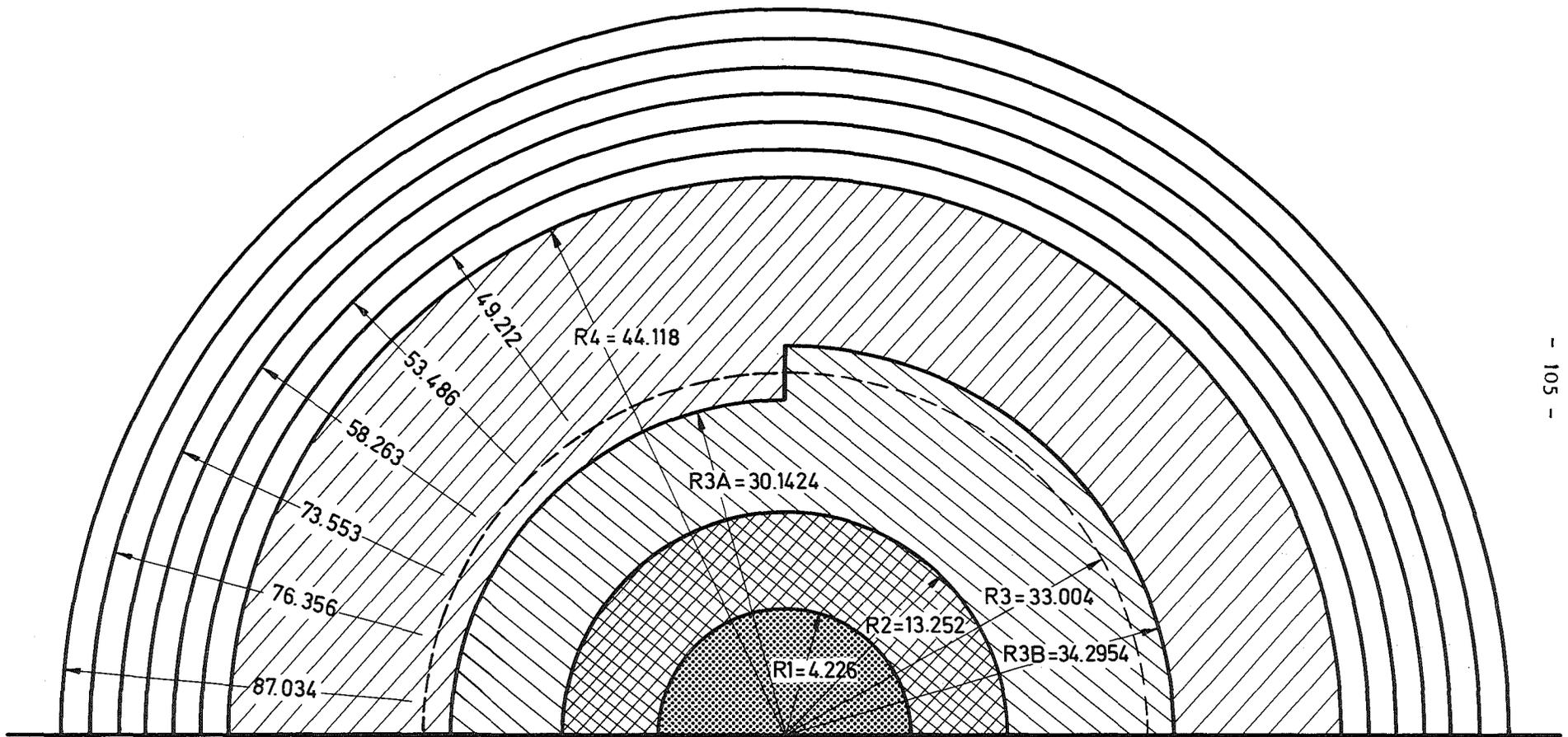
FIG. B3    Core  1D  R  Θ - Geometrie
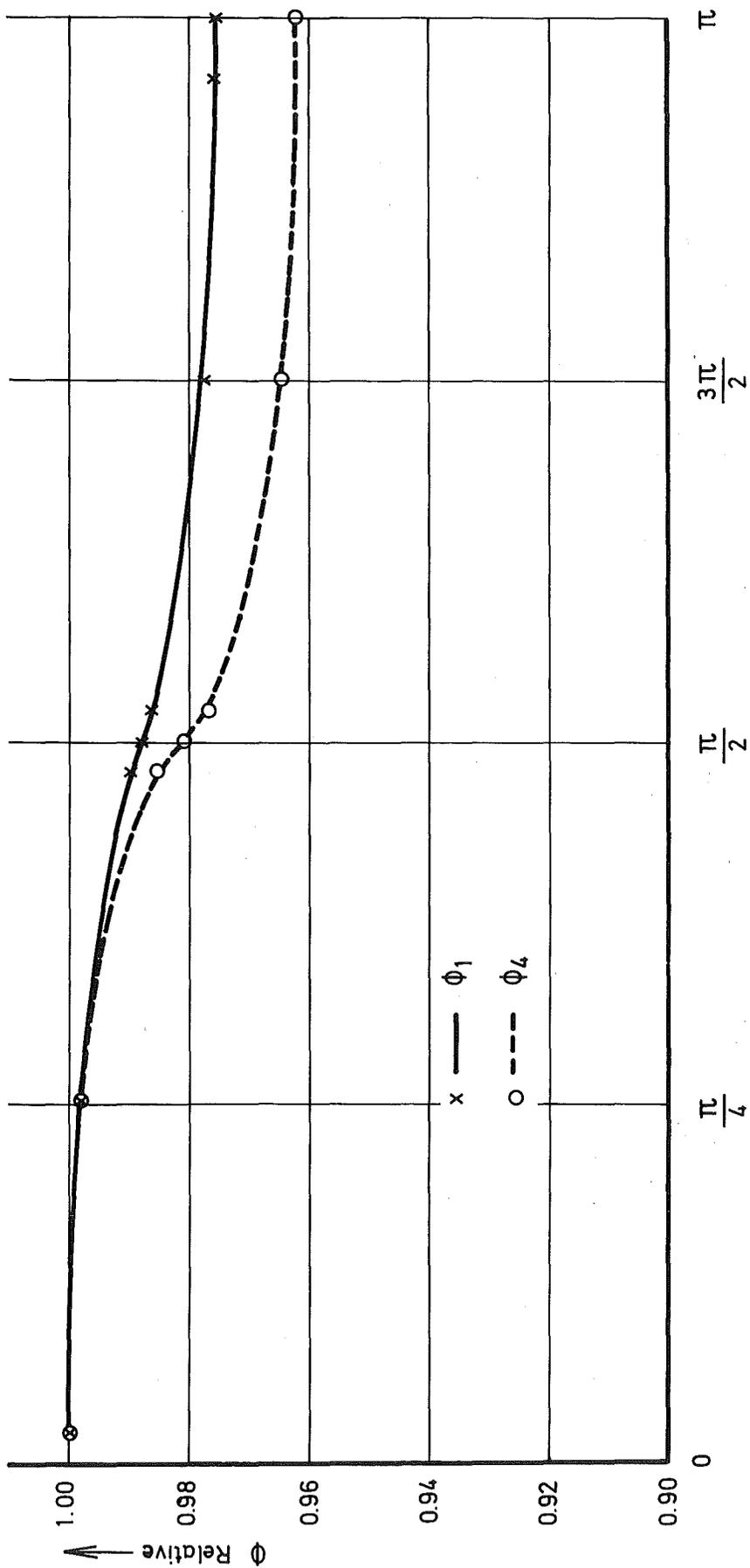
Fig. B4    Relative Flux Distribution
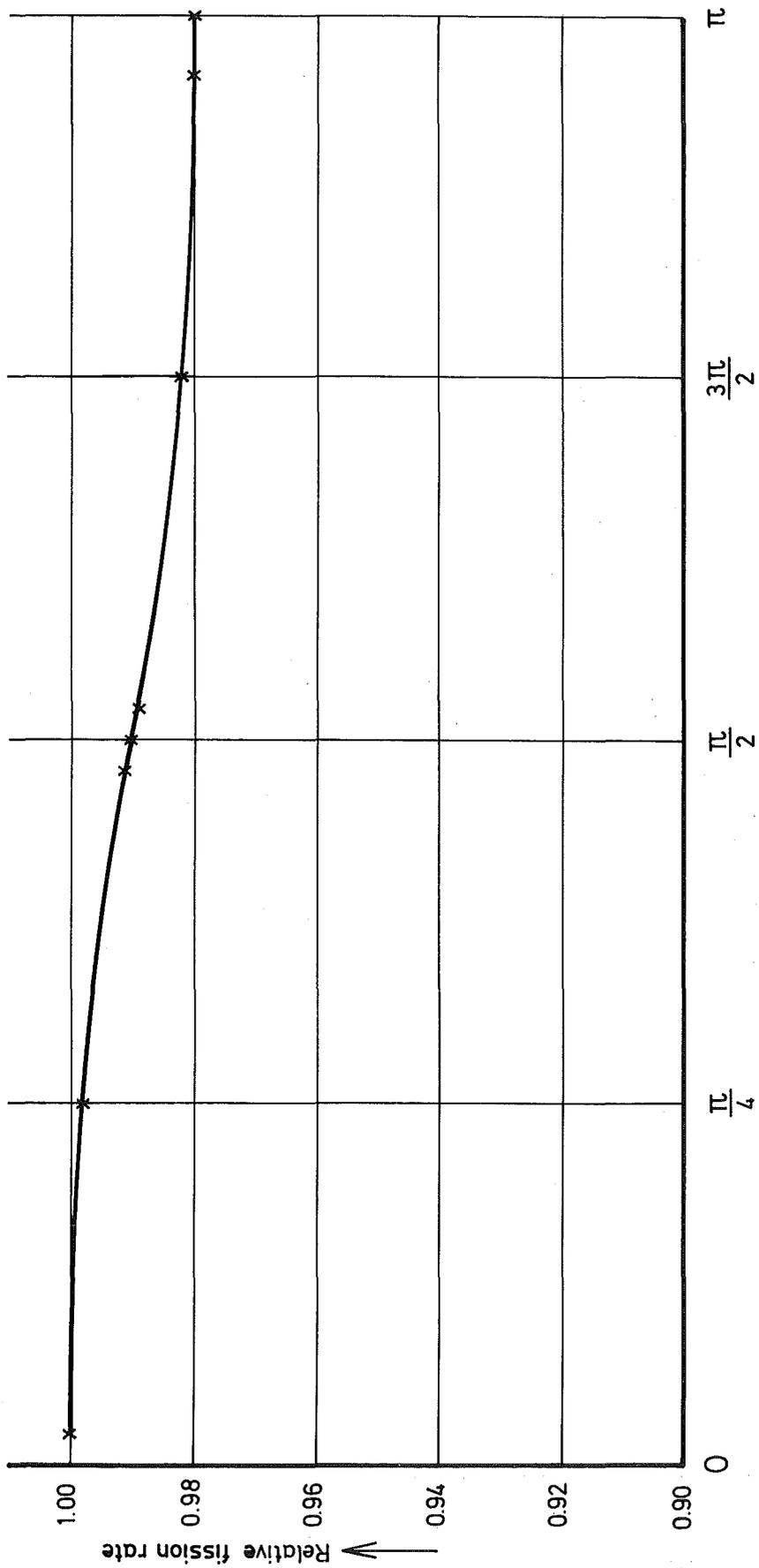
Fig. B5    Fission Rate Traverse  U238
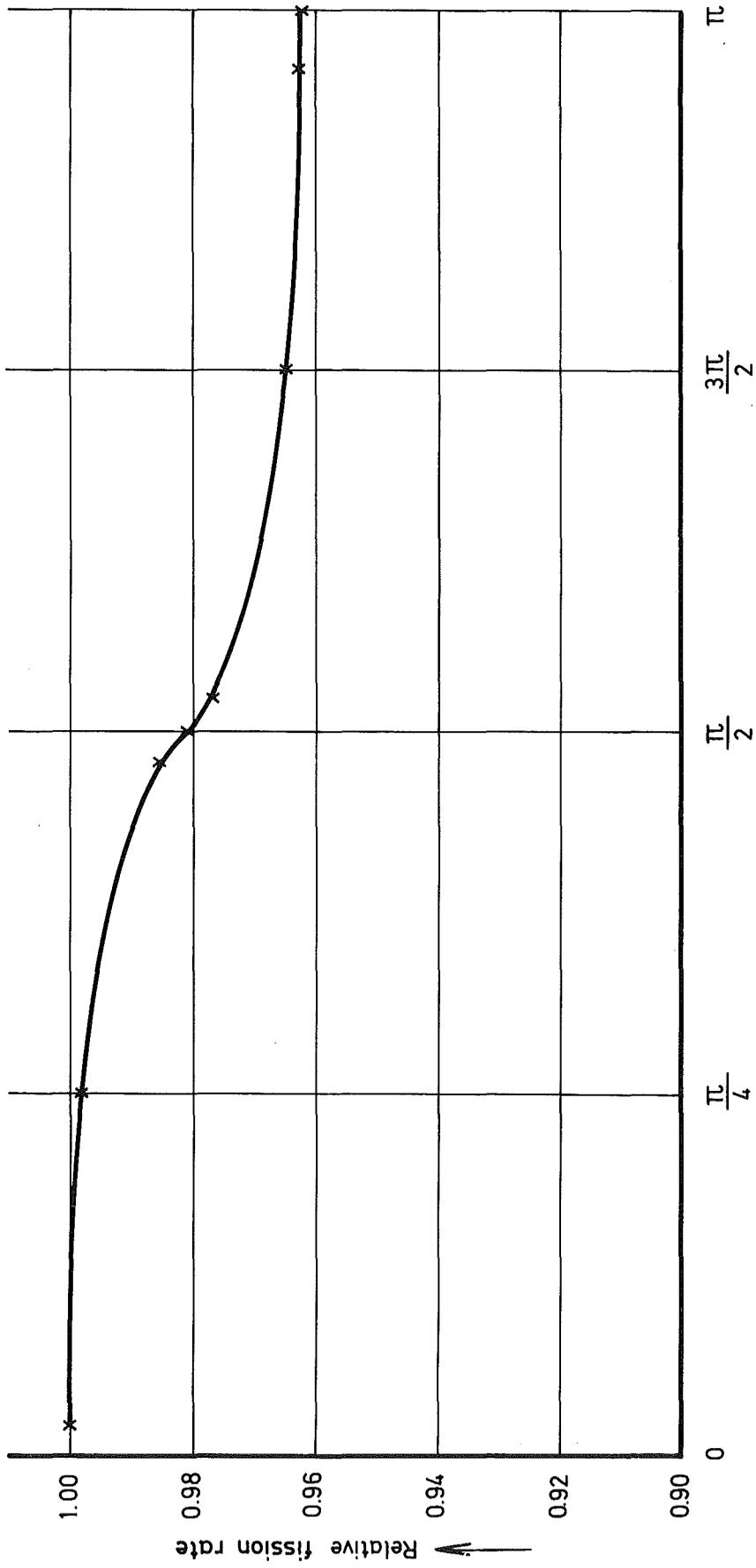
Fig. B6    Fission Rate Traverse  Pu 239
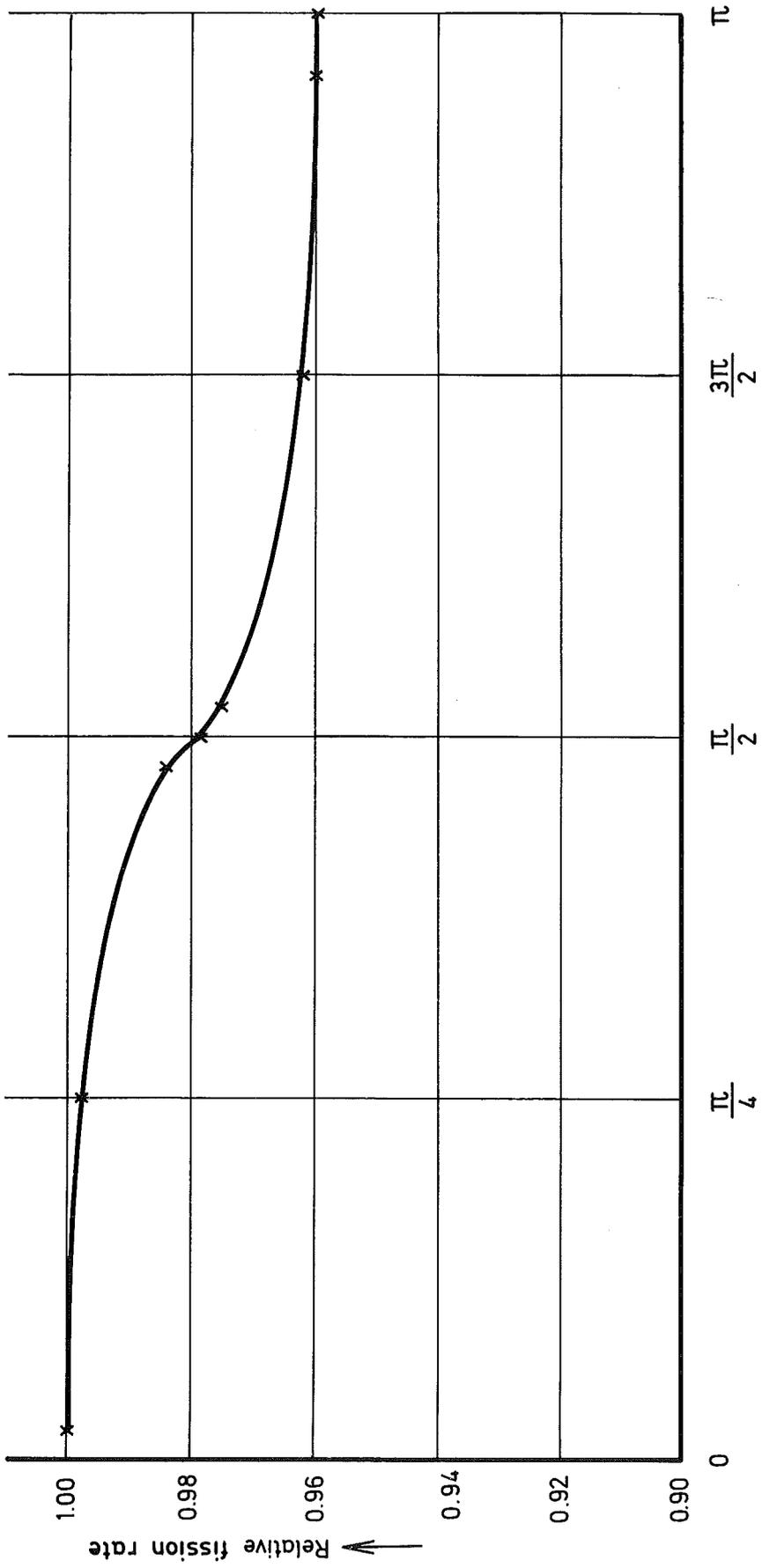
Fig. B7    Fission Rate Traverse  U 235