KERNFORSCHUNGSZENTRUM

KARLSRUHE

Juli 1972

KFK 1633

Institut für Angewandte Systemtechnik und Reaktorphysik
Projekt Schneller Brüter

# Control Rod Experiments in SNEAK-Assembly 6

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Compiled by

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With contributions of

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## - A B S T R A C T -

The present report summarizes the results of evaluation of the control rod experiments performed in the SNEAK-6 program (February - September 1970). The assembly SNEAK-6, devoted to studies related to the design of the SNR (Schneller Natriumgekühlter Reaktor), comprises a 90 cm high central plutonium fuelled zone (typical of the central SNR core zone), surrounded by a uranium driver zone.

The control rod materials investigated were $B_4C$, Na, Ta, Al and steel. A simulated rod, made of two parts, absorber and follower, was partially or totally inserted in the core at the center of the plutonium zone. Reactivity changes and power traverses were measured. This program was a continuation of a first control rod campaign performed in the assembly SNEAK-2, part C.

From the results of evaluation, conclusions are drawn in order to be applied on the design of the SNR control rods.

The work covered by this report was performed in close cooperation between the Gesellschaft für Kernforschung, the industrial consortium for the SNR and the French MASURCA group at Cadarache.


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**14. Juli 1972**

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

Dieser Bericht faßt die Auswertungsergebnisse der im Rahmen des SNEAK-6 Programms durchgeführten Kontrollstabexperimente (Februar - September 1970) zusammen. Die SNEAK-6 Anordnung, die für die Entwurfstudien für den S N R (Schneller Natriumgekühlter Reaktor) diente, enthielt eine 90 cm hohe zentrale Plutonium-Zone (typisch für die zentrale S N R Core zone), umgeben von einer Uran-Treiberzone.

Die untersuchten Kontrollstabmaterialien waren : $B_4C$, Na, Ta, Al und Stahl. Der simulierte Stab bestand aus einem Absorberteil und einem Nachfolgerteil und wurde teilweise oder ganz ins Zentrum der Plutonium-Corezone eingezogen. Gemessen wurden Reaktivitätsänderungen sowie Leistungsverteilungen. Dieses Programm bildete die Weiterführung einer ersten Reihe von Kontrollstabexperimenten, die in der Anordnung SNEAK-2, Teil C, durchgeführt wurden.

Aus diesen Auswertungsergebnissen werden Schlüsse gezogen im Hinblick auf den Entwurf der S N R-Kontrollstäbe.

Die in diesem Bericht beschriebenen Untersuchungen wurden in enger Zusammenarbeit zwischen der Gesellschaft für Kernforschung, dem Industriekonsortium für den S N R und der französischen MASURCA Gruppe in Cadarache durchgeführt.

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## - R E S U M E -

Ce rapport présente les résultats d'évaluation des expériences de barres de contrôle effectuées au cours du programme SNEAK-6 (février - septembre 1970). L'assemblage SNEAK-6, consacré à des études liées à la conception du SNR (Schneller Natriumgekühlter Reaktor), comprend une zone centrale haute de 90 cm à combustible plutonifère (représentation de la zone centrale du coeur SNR), entourée d'une zone nourricière à l'uranium.

Les matériaux de barres de contrôle étudiés ont été : $B_4C$, Na, Ta, Al et acier. La barre de simulation, composée d'une partie absorbante et d'une partie prolongatrice, a été partiellement ou totalement insérée dans le coeur au centre de la zone au plutonium. On a mesuré des changements de réactivité ainsi que des distributions de puissance. Ce programme constituait la continuation d'une première campagne de barres de contrôle réalisée dans l'assemblage SNEAK-2, partie C.

A partir de ces résultats d'évaluation, on déduit des conclusions à appliquer à la conception des barres de contrôle du SNR.

Le travail couvert par ce rapport a été réalisé en étroite collaboration par la Gesellschaft für Kernforschung, le consortium industriel SNR et le groupe français de MASURCA à Cadarache.


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# - TABLE OF CONTENTS -



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## 0.- INTRODUCTION.

The present report summarizes the results of evaluation of the control rod experiments performed in the SNEAK-6 program (February - September 1970). The assembly SNEAK-6, devoted to studies related to the design of the S N R (Schneller Natriumgekühlter Reaktor), is a sequence of 90 cm high cores comprising a central plutonium fuelled zone (typical of the central S N R core zone), surrounded by a uranium driver zone.

A separate KFK report [1] describes in detail the evaluation of all the experiments belonging to the SNEAK-6 program, with the exception of the control rod experiments. Since, the latter formed a special part of the program with a very specific purpose, it appeared appropriate to describe them in a separate report.

The control rod materials investigated were $B_4C$, Na, Ta, Al and steel. A simulated rod, made of two parts, absorber and follower, was partially or totally inserted in the core at the center of the plutonium zone. Reactivity changes associated with this insertion and power traverses with and without the presence of the rod were measured.

This program was a continuation and extension of a first control rod campaign performed in the assembly SNEAK-2, part C [2] ; evaluations of the two programs are compared.

As was the case for past SNEAK assemblies devoted to the S N R, the experiments were performed by the SNEAK experimental group and evaluated in close cooperation by groups from GfK (I A R), BELGONUCLEAIRE and INTERATOM, using the same basic methods.

The experiments were also placed into the framework of the cooperation SNEAK-MASURCA. The MASURCA group of the French C E A (Cadarache), who contributed directly to SNEAK-6 by furnishing 150 kg plutonium fuel and by exchanging experimental equipment, designed the experimental program for SNEAK-6D and evaluated these experiments with their own methods ; a C E A report [3] compares the evaluations made on both sides on SNEAK-6D including the control rod experiments.

The present report attempts to gather the results of all evaluations performed for SNEAK-6 control rod experiments :

- for SNEAK-6A, by the group BN-IA (Chapter 2) ;
- for SNEAK-6D, by the group GfK-BN including comparisons with the French results (Chapter 3).

Finally, conclusions are drawn, in order to be applied on the design of the S N R control rods (Chapter 4).

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# 1.- EXPERIMENTAL TECHNIQUES AND CORE CONFIGURATIONS.

## 1.1. REACTION RATE MEASUREMENTS.

Most of the experimental techniques applied in the SNEAK-6 control rod experiments were already utilized in SNEAK-2 ; they are described in [2] .

The distribution of $U^{235}$, $U^{238}$ and $Pu^{239}$ fission rates were measured, either axially in a position close to the control rod full or half inserted or radially for its full insertion.

These fission rate traverses in general were measured by fission chambers, but in SNEAK-6D foils have also been used in the radial direction. One may find all details relative to fission chambers and foils in [1] .

The traverses measured for the individual isotopes were combined to give the power density traverse, using in general measured values of fission ratios at a reference point. The accuracy of the power density traverses derived from experiments and normalized at a reference point is 0.3 % in the core region.

## 1.2. REACTIVITY MEASUREMENTS.

1.2.1. In the SNEAK-6 control rod experiments, the quasi-critical method was used, as before in SNEAK-2C [2] , for almost all the steps of the measurements.

The principle of this method is that, in order to compensate for the negative reactivity insertion due to the control rod, core elements are progressively added at the core edge, in such a way that the core is always close to criticality (within $\pm$ 1 \$). The reactivity changes are measured between successive configurations in which the control rod (which means mostly 4 SNEAK-elements, 10.88 x 10.88 cm size) is made, respectively, of core medium (reference state), follower, absorber partially or fully inserted. These configuration changes are realized in steps small enough to allow a measurement of the corresponding reactivity changes by calibrated SNEAK shim rods. The measurement of such a step is assessed to have an accuracy of $\pm$ 5 %.

1.2.2. The major causes for uncertainties of the measured reactivities using the quasi-critical method are associated with the calibration of the SNEAK shim rods, with the ionisation chambers used for flux readings and with other minor influences.

The calibration of the SNEAK shim rods should ideally be repeated for each reactivity step to be measured. As this is not done for all cases, an error is introduced by the interaction between the shim rods and the simulated SNR rods. It may reach 3 %, according to previous experience.

Also the calibration is affected by spatial effects associated with the position of the $BF_3$ ionisation chambers used for flux readings ; these chambers are located outside of the radial blanket and these readings are affected by the core size and asymmetrical flux distributions. Errors of the order of 3 % are associated with such effects.

Further, there are minor influences, due to the electronic equipment, to the inverse kinetics method used, to temperature drifts, etc., which are estimated to 1 %.

A quadratic combination of these independent errors, leads to the figure of $\pm$ 5 %.

1.2.3. A second method was used complementarily, for a few cases, in SNEAK-6A and -6D : the subcritical method, based on source multiplication. Due to the presence of the plutonium spontaneous neutron source Q, there is a simple relation between the flux $\Phi$ measured by a $BF_3$ ionisation chamber (see Fig. II-1) and the subcriticality margin $\Delta k$ ($\Phi$ = constant . $Q/\Delta k$). Flux values are therefore measured for the subcritical configuration and converted into $\Delta k$ using subcriticalities achieved with calibrated SNEAK rods as a standard. This method has an accuracy within 10 % for subcriticality margins up to 5 $.

1.2.4. The following results in SNEAK-6A allow a comparison between quasi-critical and subcritical techniques :

Separation of 4 $B_4C$ absorbers from position 0 to 1 (Fig. II-1) :

| | |
|---|---|
| quasi-critical method | 106.5 ¢ |
| subcritical method | 105.0 ¢ |

Further separation of the 4 $B_4C$ absorbers from position 1 to 2 :

| | |
|---|---|
| quasi-critical method | 13.8 ¢ |
| subcritical method | 13.8 ¢ |

For such relativelly small reactivities, the agreement between the results of the two techniques is good.

One large reactivity step was measured by the subcritical method. This was the transition from one to four SNEAK elements made of $B_4C$ at the center of a large void zone (60 elements) in the core of 400 elements. The subcritical result was : 580.9 ¢.

This result may not be compared directly to any quasi-critical measurement, but an approximate comparison in the following way is possible. One may divide the total reactivity difference into several small reactivity steps which were measured by the quasi-critical method in different core sizes (see the details of these steps in Chapter 2, Tables I and III) and sum up the results. This sum amounts to 523.4 ¢.

The difference between the two techniques is 10 %. The reactivity steps were however measured in cores of different sizes. Using a calculated correlation between rod reactivity and core size, a correction is found which reduces the discrepancy to 5 %.

For the evaluation which follows, when the results of the two techniques are available, their average is retained.

1.2.5. In the course of the evaluation, the GfK-BN-IA group in general considers the reactivities in ¢ obtained using the calibration of the SNEAK shim rods as the basic experimental results. The comparison calculation-experiment therefore must use a calculated average delayed neutron fraction $\beta_{eff}$.

Let us note that for its evaluation the MASURCA group converts all measured reactivities into equivalent numbers of edge element worth or for fine adjustments into variations of the critical core radius, using a calculated correlation ; these quantities are taken as the basic experimental results to be compared with calculations.

## 1.3. CORE CONFIGURATIONS SNEAK-6A AND -6D.

The following summary presentation is extracted from [1], which gives a full description of the cores realized in the series SNEAK-6 and from [3], which is specially devoted to SNEAK-6D.

In SNEAK-6A (see Fig. II-1) as well in SNEAK-6D (see Fig. III-1), the 90 cm high core is made of a central plutonium fuelled zone ($Z_1$) surrounded by a uranium driver zone ($R_1$). The axial blanket, 40 cm high, contains $UO_2$, Na and steel and simulates a breeder blanket. The radial blanket, about 33 cm thick, is made of depleted uranium blocks.

The central plutonium zone is in fact made of two concentric zones having practically the same homogeneized compositions but differing by their heterogeneity : SNEAK plates or MASURCA rodlets. In SNEAK-6A, the $Z_1$ SNEAK zone is at the center, surrounded by the $Z_1$ MASURCA zone as a buffer, while in SNEAK-6D these positions are reversed. The control rods investigated are always placed at the core center, that is respectively, in $Z_1$ SNEAK (6A) and in $Z_1$ MASURCA (6D).

The main dimensions of the two critical cores 6A and 6D are given in the following Table.

| Core zones | SNEAK-6A | SNEAK-6D |
|---|---|---|
| Critical core zone (Pu) | | |
| designation | $Z_1$ SNEAK | $Z_1$ MASURCA |
| equivalent radius (cm) | 37.3 | 30.1 |
| height (cm) | 89.4 | 91.45 |
| Pu buffer zone | | |
| designation | $Z_1$ MASURCA | $Z_1$ SNEAK |
| equivalent radius (cm) | 49.1 | 49.1 |
| height (cm) | 60.3* | 89.4 |
| U driver zone | | |
| designation | $R_1$ | $R_1$ |
| equivalent radius (cm) | 59.2 | 58.95 |
| height (cm) | 89.3 | 89.3 |

*Does not extend along the total core height, is axially surrounded by $R_1$.

The homogeneized compositions are listed in Appendix

For cell structures and other, more detailed information about the cores, one should refer to [1] and [3].

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## 2.- SNEAK-6A CONTROL ROD EXPERIMENT.

### 2.1. SCOPE OF THE CONTROL ROD EXPERIMENT IN SNEAK-6A.

After the control rod experiments in SNEAK-2C, the evaluation of which is reported in [2], a large complementary program was set up and realized in SNEAK-6A.

Both programs are complementary :

- the radial positions of control rods in the SNR were mocked-up as closely as possible in 2C, while a central position only is considered in 6A ;
- the core height of 60 cm in 2C becomes 90 cm in 6A (SNR height : 95 cm).

The same compositions for absorbers ($B_4C$, Ta) and followers (Na, Al) are used.

In 2C, the mock-up type representation led to a complicated zoned sector core, while the 6A core, characterized by only one enrichment level, is as cylindrical as possible.

The 6A experiments, avoiding any geometrical disturbance, allow therefore a better check of the methods of calculation used in control rod design, and particularly of the absorber cross-sections.

Table II-1 gives a recapitulation of the experiments performed in SNEAK-6A - date, designation, traverses - with the respective core sizes (in number of core elements, 5.44 x 5.44 cm each); it is to be noticed that :

1°) a special experiment was devoted to the study of the self-shielding effects within an absorber rod : for both rods $B_4C$ and Ta full inserted, reactivity changes were measured also when the four elements constituting the rod were split up radially, by one and by two element widths. Fig. II-1 shows the $B_4C$ rod split up in these two steps ;

2°) the insertion of the Na - $B_4C$ rod was considered not only in a reference core but also in the core with the large sodium void zone of 60 elements described in [1].

The entire program included the loading of 10 core configurations (+4 with sodium void), and the measurement of 29 reactivity changes and 6 fission rate traverses, as detailed in Table II-1. The experimental methods are recalled in Chapter 1. The number of core elements varied from 372 to 482, which represents an increase of the equivalent core radius from 59.2 cm to 67.4 cm.

The compositions of the different core zones, blankets and control rods used are gathered in Appendix ; the control rod compositions are the same as for SNEAK-2C (see [2]).

- T A B L E II-1 -

RECAPITULATION OF THE SNEAK-6A CONTROL ROD EXPERIMENTS.

| Date (in 1970) | Number of core elements | Designation of the steps (SC = subcritical react. meas.) | Traverses measured |
|---|---|---|---|
| 17/2 | 372 | Critical core | |
| 20/2...3/4 | 376 | Reference core | radial, axial |
| | | Control rod experiments in the sodium void | |
| 10/4 | 376 | Reference, Na void | |
| 13/4 | 376-380-392--400 | Na.-$B_4C$ characteristic (1 element) in Na void | |
| 14/4 | 400 | $B_4C$ in Na - void (SC) | |
| 17/4 | 468 | $B_4C$ in Na-void | |
| | | Normal control rod experiments | |
| 23/4 | 468 | $B_4C$ | Radial |
| 27/4 | 482 | $B_4C$, split-up, 1. step | |
| 27/4 | 482 | $B_4C$, split-up, 2. step | |
| 28/4 | 468 | 3/4 $B_4C$ | |
| 29/4 | 458 | 3/4 $B_4C$ | |
| 29/4 | 458 | 1/2 $B_4C$ | |
| 30/4 | 442 | 1/2 $B_4C$ | Axial |

TABLE II-1 (continued).

| Date (in 1970) | Number of core elements | Designation of the steps (SC = subcritical react. meas.) | Traverses measured |
|---|---|---|---|
| 13/5 | 416 | 1/4 $B_4C$ | |
| 13/5 | 416 | 1/2 $B_4C$ (SC) | |
| 14/5 | 416 | 1/2 Ta | Axial |
| 15/5 | 412 | 1/4 Ta | |
| 19/5 | 412 | Na | |
| 19/5 | 412 | Al | |
| 20/5 | 412 | 1/2 Ta (SC) | |
| 20/5 | 428 | 1/2 Ta | |
| 21/5 | 428 | 3/4 Ta | |
| 21/5 | 432 | 3/4 Ta | |
| 21/5 | 432 | Ta | Radial |
| 21/5 | 432 | Ta, split-up, 1.step | |
| 22/5 | 432 | Ta, split-up, 2.step | |
| 26/5 | 392 | Na | |
| 26/5 | 392 | Reference (392) | |

Fig. II-1 and II-2 illustrate this control rod experiment by representing, respectively, a typical cross-section of the core (corresponding to $B_4C$ full inserted), and the axial distribution of the materials Na - $B_4C$ and Al - Ta.

## 2.2. METHODS USED IN THE CALCULATIONS.

The whole control rod experiment was calculated using diffusion theory with 4-group condensed cross-sections ; refined methods, mainly transport theory and more energy groups were applied to some typical cases of the series ; the conclusions thereof are gathered in paragraph 2.6.

The cross-sections are condensed from the NAPPMB set [6] as explained in [2] . The condensation scheme used is :

1 to 4, 5 and 6, 7 to 10, 11 to 26.

The computer programs used allowed calculations in the following geometries :

- one dimension, radial : used only for condensation, for checks and for obtaining some corrections (see also paragraph 2.6.) ;
- two dimensions (RZ) ;
- two dimensions (XY) ;
- three dimensions (X Y Z), synthesis technique, computer program KASY [4][5] .

The basic calculations were performed in 2-D (RZ) and 3-D (X Y Z). In fact, for the central rod problem of SNEAK-6A, 2-D (RZ) is generally sufficient (except for the split-up cases). The KASY synthesis technique in the geometry (X Y Z) was extensively used in SNEAK-2C ; it is therefore also used here and its results are compared with the 2-D (RZ) results in order to further check the method. It is clear that the usual fast breeder cores, containing many control rods, require a 3-D model.

The (XY) results serve mainly to derive trial functions for the synthesis ; they are also directly applicable to the split-up experiment.

In (XY) or (X Y Z) geometry, a quarter of the core section is considered each time the experimental configuration was symmetrical, or close to symmetry ; in few cases, the full size model was needed.

An adequate mesh spacing is used, small enough to avoid any mesh effect, but not unnecessarily small, in view of the computer time needed : the spacing was about 1 cm in the control rod, in the main part of the core and at the core-blanket boundary.

## 2.3. EVALUATION OF THE REACTIVITY CHANGES.

Calculations of the different core configurations produce multiplication factors $k_{eff}$, which are given as illustrative figures in the following Table II-2 for all the cases considered ; the 2-D (XY) results are listed together with the KASY 3-D (XY/Z) results. The numerical accuracy of $k_{eff}$ is $10^{-5}$.

- T A B L E II-2 -

SNEAK - 6A CONTROL ROD EXPERIMENT.
DIRECT RESULTS : $k_{eff}$ FROM THE 2-D (XY) AND 3-D (X Y Z) CALCULATIONS.

| Core loading (number of core elements) | D e s i g n a t i o n | 2-D (XY) | 3-D (XY/Z) |
|---|---|---|---|
| 376 | Reference | .96898 | .97064 |
| | sodium void | .96725 | .97234 |
| 400 | (sodium void)* | .97741 | .98272 |
| | $B_4C$ in sodium void | .94043 | .94408 |
| 468 | (Na.) | .99286 | |
| | $B_4C$ | .96686 | |
| 482 | $B_4C$ | .97263 | |
| | $B_4C$, first split-up | .96790 | |
| | $B_4C$, second split-up | .96839 | |
| 458 | $(B_4C)$* | .96341 | .96556 |
| | (Na.) | .98972 | |
| | $B_4C$ in sodium void | .96336 | .96769 |
| | 3/4 B4C | | .96956 |
| | 1/2 $B_4C$ | | .97843 |
| 416 | $B_4C$ | .94759 | |
| | (Na.)* | .97540 | .97900 |
| | 1/2 B4C | | .96301 |
| | 1/4 B4C | | .97314 |
| 412 | Na. | .97373 | .97732 |
| | Al | .97430 | .97864 |
| | (Ta.)* | .96026 | .96208 |
| | 1/4 Ta. | | .97494 |
| | (1/2 Ta.)* | | .96929 |
| | (3/4 Ta.)* | | .96404 |
| 428 | (Al) | .98042 | |
| | (Ta.)* | .96631 | .96880 |
| 432 | Ta. | .96777 | |
| | Ta., first split-up | .96592 | |
| | Ta., second split-up | .96755 | |
| | (Al) | .98130 | |
| 392 | reference | .97581 | .97799 |
| | Na. | .96593 | .96920 |

( ) : step not realized in this core ; calculation made in order to produce a trial function for the synthesis.
( )* : step not realized in this core, but in a very similar core.

The $k_{eff}$ obtained by DIXY are slightly too small (by 2 to 5 ‰). A universal $B^2$ value of $6.44\ 10^{-4}$ $cm^{-2}$ was used. The advantage of KASY not to need very accurate $B^2$ values in the calculation of the trial functions, was already underlined [5].

From the preceding $k_{eff}$ values, reactivity changes expressed as simple differences of $k_{eff}$ are easily derived. From a series of comparisons which are not described in details here, the following observations can be made :

- XY/Z and RZ results for reactivity changes are in fairly good agreement. The XY/Z results are generally higher by 1 or 2 %, and by 4 % in the extreme case (Na - 1/4 $B_4C$). The reactivity ref. - Na is calculated 1.8 % larger with XY/Z than with RZ, ref. - $B_4C$ 1.3 % larger and Na - $B_4C$ 1.1 % larger. This means that the cylindrisation of the control rod has no significant influence on tha reactivity. It also means that the synthesis procedure is fully satisfactory as far as the treatment of the axial neutron leakage is concerned ; this conclusion was already drawn from comparisons with direct 3-D models [4][5] ;
- simple XY calculations using universal $B_Z^2$ values overestimate largely (by 12 and 24 %) the reactivity of the followers Na and Al, with respect to XY/Z or RZ results. This conclusion was already obtained for SNEAK-2C [2] ;
- the sodium void reactivities calculated by XY/Z and XY disagree completely. This is mainly due to the use of a constant $B^2$ value in the XY calculations. It should be noted that these results are only taken as indicative figures. As a matter of fact such simplified XY calculations (with respect to $B^2$) are not applicable to a sodium effect, and even the XY/Z calculations are too approximate (too few energy groups). The goal was here to apply the standard methods used in the SNR design to the insertion of a $B_4C$ control rod in a sodium voided core compared to its insertion in the normal core (safety aspect).

Two important remarks apply to the directly calculated reactivities :

1°) One must take into account the variation of $\beta_{eff}$ due to the core size increase and to the insertion of the control rod.

The value of $\beta_{eff} = 421\ 10^{-5}\ \Delta k$ for the core of 376 elements is taken from [1] as a basic value.

The relative variations of $\beta_{eff}$ have been estimated using the method described in [2]. One observes that $\beta_{eff}$ increases significantly when the core becomes larger. This is to be expected since the olume ratio U zones to Pu zones increases.

The maximum effect is 9 %, which is significant.

For a fixed core geometry the insertion of the central control rod does not generally change $\beta_{eff}$ by more than 1 %.

The reactivity changes calculated in $\Delta k$ have been converted to ¢ using the $\beta_{eff}$ value estimated for the individual step.

2°) The single reactivity steps calculated (for one core configuration) are directly comparable with the experimental ones.

Sums of single steps, corresponding for example to the replacement of fuel elements by an absorber fully inserted, were derived from measured values as well as from calculated values for the individual steps involved. This comparison is meaningful, since for each step, the exact core configuration and size was simulated as close as possible in the calculational model.

Calculated and measured reactivity changes (in ¢) are compared in three different parts of the following Table II-3 :

Part A : single steps ;

Part B : fuel replaced by follower or absorber rod ;

Part C : follower progressively replaced by absorber.

Part A allows the comparison for single steps of reactivity, calculated in the conditions of the measurements.

Part B is well adapted to an evaluation.

Part C is of direct interest to the SNR Project.

"Calculation" refers always to the XY/Z synthesis model.

These SNEAK-6A measurements being very close to those in SNEAK-2C for a central control rod (position 5), the corresponding SNEAK-2C ratios calculation-over-experiment [2] have also been indicated for Parts B and C.

The reactivity changes measured are accurate within $\pm$ 5 %.

The calculated values are affected by calculation approximations :

- up to 2 % due to condensation ;
- negligibly due to the mesh spacing used.

- T A B L E II-3 -

SNEAK - 6A CONTROL ROD EXPERIMENT.
COMPARISON OF CALCULATION AND MEASURED REACTIVITY CHANGES (in ¢),
(with a negative sign).

| Part A - Single steps | | | |
|---|---|---|---|
| 1°) Insertion of a $B_4C$ absorber into a Na follower | Experiment | Calculation | Ratio C/E |
| Reference - Na | 183.9 | 204.8 | 1.11 |
| Na - 1/4 $B_4C$ | 153.7 | 133.7 | 0.87 |
| 1/4 $B_4C$ - 1/2 $B_4C$ | 206.0 | 231.1 | 1.12 |
| 1/2 $B_4C$ - 3/4 $B_4C$ | 186.9 | 195.3 | 1.04 |
| 3/4 $B_4C$ - $B_4C$ | 87.6 | 88.1 | 1.01 |
| 2°) Insertion of a Ta absorber into a Al follower | | | |
| Reference - Al* | 196.2 | 174.1 | 0.89 |
| Al - 1/4 Ta* | 70.6 | 84.9 | 1.20 |
| 1/4 Ta - 1/2 Ta | 121.5 | 129.5 | 1.07 |
| 1/2 Ta - 3/4 Ta | 100.7 | 118.9 | 1.18 |
| 3/4 Ta - Ta | 41.0 | 44.4 | 1.08 |
| 3°) Shadowing effect with $B_4C$ | | | |
| $B_4C$ to split-up 1 | 105.7 | 102.1 | 0.97 |
| Split-up 1 to split-up 2 | -13.8 | -10.6 | 0.77 |
| Total | 91.9 | 91.5 | 0.99 |
| 4°) Shadowing effect with Ta | | | |
| Ta to split-up 1 | 43.1 | 41.8 | 0.97 |
| Split-up 1 to split-up 2 | -30.7 | -36.9 | 1.20 |
| Total | 12.4 | 4.9 | 0.40 |
| 5°) Insertion of a $B_4C$ absorber in a large central sodium void zone** | 851.5 | 894.9 | 1.05 |

* In the case denoted Al, the upper 20 cm in the axial blanket were filled with $B_4C$. The exact compositions in the rod position were always accounted for in the calculations presented.

** This is in fact a sum of : 270.6 ¢ (insertion of 1 element $B_4C$, cores 376, 380, 392 and 400) ;
and : 580.9 ¢ (insertion of 3 new elements $B_4C$, core 400).

TABLE II-3 (continued).

| Part B - Fuel replaced by follower or absorber rod | | | | |
|---|---|---|---|---|
| 1°) Insertion of a $B_4C$ absorber into a Na follower | Experiment | Calculation | Ratio C/E | Corresponding ratio in SNEAK-2C |
| Na | 183.9 | 204.8 | 1.11 | 1.25 |
| $B_4C$ | 818.1 | 853.0 | 1.04 | 1.01 |
| 2°) Insertion of a Ta absorber into an Al follower | | | | |
| Al | 196.2 | 174.1 | 0.89 | 1.13 |
| Ta | 530.0 | 551.8 | 1.04 | 1.08 |
| 3°) Insertion of a $B_4C$ absorber in a large central sodium void zone | | | | |
| $B_4C$ (core 400) | 851.5 | 883.2 | 1.04 | |
| $B_4C$ (core 468) | 794.0 | 846.6 | 1.07 | |
| **Part C - Follower progressively replaced by absorber** | | | | |
| 1°) Insertion of a $B_4C$ absorber into a Na follower | | | | |
| Na - 1/4 $B_4C$ | 153.7 | 133.7 | .87 | |
| Na - 1/2 $B_4C$ | 359.7 | 364.8 | 1.01 | .86 |
| Na - 3/4 $B_4C$ | 546.6 | 560.1 | 1.02 | |
| Na - $B_4C$ | 634.2 | 648.2 | 1.02 | .93 |
| 2°) Insertion of a Ta absorber into an Al follower | | | | |
| Al - 1/4 Ta | 70.6 | 84.9 | 1.20 | |
| Al - 1/2 Ta | 192.1 | 214.4 | 1.12 | .99 |
| Al - 3/4 Ta | 292.8 | 333.3 | 1.14 | |
| Al - Ta | 333.8 | 377.7 | 1.13 | 1.04 |

## 2.4. CONCLUSIONS ON THE REACTIVITY CHANGES.

Point A1/B1 - $B_4C$ absorber and Na follower.

The reactivity of Na against fuel is still overestimated by calculation (11 %), but not so much as for SNEAK-2C (25 %). The overestimate was attributed to an incorrect treatment of the axial neutron leakage in this light medium ; since the core height is larger in SNEAK-6A (90 cm) than in SNEAK-2C (60 cm), the overestimate is reduced.

The reactivity of $B_4C$ is overestimated (4 %). In SNEAK-2C, the overestimate was smaller (1 %). Such a difference lies within the experimental accuracy (± 5 %).

The reactivity steps corresponding to the progressive replacement Na - $B_4C$ are either over - or underpredicted. The picture becomes clearer if one examines Fig. II-3, which represents the characteristic curve on the basis of the few points measured and calculated. The zero of the reactivity scale is fixed at the core midplane in the linear portion of the curve.

One observes that the slope of the characteristic curve is overpredicted by calculation :

9.7 ¢/cm calculated
8.9 ¢/cm measured
} (estimate using the three central points).

Such a result was already found in SNEAK-2C, where two characteristic curves Na - $B_4C$ (one SNEAK element) were directly measured [2].

Point A2/B2 - Ta absorber and Al follower.

The reactivity of Al against fuel is underpredicted by 11 %, while it was overpredicted by 13 % in SNEAK-2C.

The reason given hereabove for Na (axial leakage smaller with a larger core height) makes a trend downwards for the ratio C/E plausible. In fact, both cases are not really comparable ; the case designed Al in SNEAK-6A does not correspond to a pure Al follower, but to a partial insertion of an absorber ($B_4C$ in the upper 20 cm of blanket).

The progressive replacement Al - Ta too is illustrated on Fig. II-3. The same trend as for Na - $B_4C$ is observed : the slope of the characteristic curve is overpredicted by calculation :

5.6 ¢/cm calculated
5.0 ¢/cm measured
} (estimate using the three central points).

The reactivity of Ta versus fuel is slightly overestimated (4 %). In SNEAK-2C, the overestimate was larger (8 %). Such a difference lies with the experimental accuracy (± 5 %). It cannot be due to a systematic influence associated with the determination of $\beta_{eff}$, since the reactivity of $B_4C$ is overestimated by 4 % in SNEAK-6A, but only 1 % in SNEAK-2C.

This means that one of the main conclusions of the SNEAK-2C evaluation, which was that the reactivity of Ta was significantly overestimated with respect to that of $B_4C$, is no longer supported by the present evaluation.

Points A3 and A4 - Shadowing effect with $B_4C$ and with Ta.

For $B_4C$ which is a non-resonant absorber, the reactivity steps associated with a progressive split-up of the rod are well reproduced by the standard design methods used.

For Ta which is a resonant absorber, the same conclusion may also be drawn.

The large percentage error for some reactivity differences stems from their small absolute size due to compensating effects. The prediction of the total reactivity worth of a $B_4C$ or Ta rod is equally good if the rod is split-up or not.

Point A5 - Insertion of a $B_4C$ absorber in a large sodium void zone.

The reactivity worth of a $B_4C$ control rod was measured twice : in the normal core and in the core comprising a central zone voided of Na.

The influence of a sodium void on the $B_4C$ worth is here evidenced.

Experimentally, one obtains :

| | |
|---|---|
| $B_4C$ worth in reference core (quasi-critical method, series of cores 468 to 392) | 818.1 ¢ |
| $B_4C$ worth in Na void zone | |
| a. $B_4C$ in Na void - $B_4C$ (core 468) | - 49.9 ¢ |
| b. $B_4C$ - reference | 818.1 ¢ |
| c. reference - Na void zone (core 376) | 25.8 ¢ |
| | 794.0 ¢ |

The decrease of $B_4C$ worth due to the spectral hardening associated with a sodium void in the surrounding core (60 elements) is therefore of 3 %.

On the other hand, the calculation gives the following results :

| | | |
|---|---|---|
| $B_4C$ in normal core, calculation : | 853.0 ¢ | C/E = 1.04 |
| $B_4C$ in Na - voided core, calculation : | -46.8 ¢ | |
| | 853.0 ¢ | |
| | 40.0 ¢ | |
| | 846.6 ¢ | C/E = 1.07 |

The latter reactivity was also measured by the subcritical technique in another core size ; the result was : 851.5 ¢. The calculation gives 883.2 ¢ or a ratio C/E = 1.04.

It seems, within the experimental accuracy, that the standard methods of calculation used here (not specially adapted to sodium void) produce the same or a slightly larger overestimate of the experimental value for a $B_4C$ rod in a sodium voided zone than in a normal core.

## 2.5. EVALUATION OF THE POWER DENSITY DISTRIBUTIONS WITH CONCLUSIONS.

The following traverses were measured in SNEAK-6A, control rod experiments :

| | | |
|---|---|---|
| Radial | reference core (376 elements) | see [1] |
| Radial | $B_4C$ fully inserted (468 elements) | Fig. II-4 |
| Radial | Ta fully inserted (432 elements) | Fig. II-5 |
| Axial | reference core (376 elements) | see [1] |
| Axial | $B_4C$ half inserted (458 elements) | Fig. II-6 |
| Axial | Ta half inserted (428 elements) | Fig. II-7 |

In fact, fission traverses for isotopes Pu 239, U 238 and U 235 were directly measured : power density traverses were derived in general by combining the measured traverses with measured values of fission ratios.

The accuracy of the power density traverses derived from experiments is 0.3 % in the core region.

Measured and predicted traverses are plotted together on Fig. II-4 to II-7.

The normalization points chosen are in the radial direction a point equidistant from the rod and from the following core zone, and in the axial direction at half the follower height.

Radial traverses.

The agreement calculation/experiment is fairly good in the radial direction. The largest deviations are :

1 % in the reference core ;
1 % with $B_4C$ fully inserted ;
2 % with Ta fully inserted.

Therefore, the conclusions already drawn for SNEAK-2C [2] are confirmed, including the trend to overestimate the predicted peak-to-average power ratio.

The predicted curves are generally as good in the close vicinity of the central control rod as in the center of the fuel zone ; there are also no significantly larger deviations near the outer boundary of the central zone (the surrounding zone, $Z_1$ MASURCA, has practically the same fuel enrichment) ; the homogeneous calculations performed agree with measurements in spite of the actual change in heterogeneity.

The presence of a central control rod results only in a slightly worse prediction (of about 1 %) than for the reference case.

Axial traverses.

The picture is not so favourable in the axial direction.

In the reference case, one observe the following (with the normalization point chosen 24 cm below the core midplane) :

- the maximum is 1 % overestimated ;
- the maximum deviation is about 3 % in the core far from the boundary ;
- the deviation becomes larger in the vicinity of the axial blanket boundary :
  . 7 to 9 % underestimate 3 cm inside the core ;
  . 11 to 14 % underestimate at the boundary ;
- the measured curve is slightly asymmetrical which is not reproduced by calculation.

A renormalization to the same core average value modifies slightly the deviations but the trend to an underestimate near the boundary remains.

In case of a half inserted $B_4C$ rod (Na - $B_4C$), one observes a tilting effect, which leads to a general worsening of the prediction :

- 3 % deviation at the maximum of the curve ;
- 4 % deviation in the zone surrounding the transition Na - $B_4C$ (from + 15 to - 15 cm) ;
- 3 % deviation in the fuel zone far away from any boundary ;
- the deviation near the blanket boundary is reduced or even reversed (near the $B_4C$ absorber).

This comparison bears some similarity with the corresponding cases in SNEAK-2C [2], except in the vicinity of the blanket. But the axial blanket is of another type (breeder blanket $UO_2$ - Na here, depleted U in SNEAK-2C).

In the case of a half inserted Ta rod (Al - Ta), one observes a fairly good agreement between the calculated curve and the measured points. Surprisingly the agreement is better than in the case Na - $B_4C$, and close to that in the reference case :

- the maximum is 1 % overestimated (with the normalization chosen at the point Z = - 24 cm) ;
- the maximum deviation is also about 1 % far from core boundaries ;
- the deviation becomes larger in the vicinity of the axial blanket :
  . 5 to 8 % underestimate 3 cm inside the core ;
  . 5 to 10 % underestimate at the boundary.

## 2.6. DIFFERENT METHOD COMPARISONS, PARTICULARLY TRANSPORT-DIFFUSION.

Many complementary calculations were made for the SNEAK-6A control rod experiment, in order to compare different methods, or to verify the validity of the approximation used. Comparisons were made between 26 groups and few groups (13 and 4), between diffusion results obtained with different geometrical models : 1-D (radial), 2-D (RZ), 3-D (X Y Z), and between transport and diffusion theory in one and two dimensions.

One has verified as in [2] that the effects of a few-group condensation have been minimized with the methods used and are small, as well for the control rod worths as for the power distribution in the core region.

The geometrical models 2-D (RZ) and 3-D (X Y Z) give very similar reactivity changes ; typical relative deviations are :

1.8 % (ref - Na) ; 1.3 % (ref - $B_4C$) ; 1.1 % (Na - $B_4C$).

The comparisons between transport and diffusion results merit a more careful attention. Table II-4 gathers the deviations observed on reactivity worths $\Delta k$ :

- in one dimension (radial) ; the transport calculations correspond to the $S_4$ order of approximation, to isotropic scattering cross-sections, and are performed using two definitions of transport cross-sections :

  1°) STR, current weighted (used in diffusion)(results given in the Table) ;

  2°) STRTR, flux weighted ;

- in two dimensions (RZ).

The deviations between one and two dimensions diffusion results are also given.

- T A B L E II-4 -

SNEAK - 6A CONTROL ROD EXPERIMENT.
COMPARISONS OF TRANSPORT AND DIFFUSION RESULTS.

| Relative changes of the reactivities (%) | Diffusion 2 - D → Diffusion 1 - D | Diffusion 1 - D → Transport 1 - D | Diffusion 2 - D → Transport 2 - D |
|---|---|---|---|
| ref - $B_4C$ | - 3.3 | -3.5 | -5.0 |
| ref - Na | + 13.0 | -1.8 | -8.6 |
| $B_4C$ - Na | - 8.3 | -4.0 | -4.1 |

The definition of the transport cross-section influences the results by about 1 %, the STRTR results being systematically larger.

For the reactivity worth of the absorber $B_4C$, the one- and two-dimensional transport effects are comparable (3.5 and 5.0 %), and they compare relatively well with the 7 % effect obtained for SNEAK-2C [2] .

For the reactivity worth of the sodium follower, the transport effects in one (1.8 %) and in two dimensions (8.6 %) are completely different, since only in 2-D, the advantage of transport theory in correctly describing the axial leakage becomes effective. They also differ from the 26 % effect found for SNEAK-2C [2] . The latter is clearly due to the different core heights.

For the sodium follower reactivity worth, one deduces that a geometrical model which contains the axial direction (i.e., two-dimensional in this case) is needed in order to give a meaningful transport correction (transport value - diffusion value).

As far as relative power density traverses are concerned, the comparisons made give the following conclusions :

- one- and two-dimensional results agree fairly well, as well in diffusion, as in transport ; the use of a second dimension in the model brings therefore here no significant advantage ;
- the transport results obtained with the two different definitions of transport cross-sections considered (STR and STRTR) are identical (within 0.1 %) ;
- for all the points located far from core boundaries, the transport and the diffusion results agree fairly well (within 0.3 %) ;
- at boundaries, one observes the following maximum deviations (transport-diffusion/diffusion) :

| | |
|---|---|
| core/blanket, axial traverse | - 0.5 % |
| $B_4C$ rod/core, radial traverse | - 1.4 % |
| Na rod/core, radial traverse | - 0.5 % |

Fig. II-8 illustrates this comparison of diffusion and transport results.

For the cases considered (full insertion of rods), the use of transport instead of diffusion, in the calculations, does not improve significantly the prediction of the experimental power density traverses.

*

* *

## 3.- CONTROL ROD EXPERIMENTS.

### 3.1. SCOPE OF THE CONTROL ROD EXPERIMENT IN SNEAK-6D.

The assembly 6D was utilized for further control rod experiments. A particular wish of the MASURCA-team was the consideration of only fully inserted control rods and, respectively, rod followers. In order to preserve symmetry, the upper axial blanket, as well as the lower axial blanket, were kept free from absorber, and respectively, rod follower material as far as it was possible in view of safety requirements. In control rod experiments, a number of different core types had to be built-up. A core type is defined as an assembly with a certain absorber or follower in the core center and a defined number of additionally loaded radial core elements (additionally loaded core elements in relation to the "clean critical"). Therefore, in Table III-1 are given, for the entire control rod experiment, the realized core types with their reactivity difference in comparison to the previously given core type. All measured reactivity differences have an accuracy of $\pm$ 5 % on the absolute worths. Through addition of the reactivity differences, one derives, for example, the worth of

a. four sodium elements with respect to the reference core ;

b. four aluminium elements with respect to the reference core ;

c. four tantalum elements with respect to four aluminium elements ;

d. four boron carbide elements with respect to four sodium elements and so forth.

The reference core is here, by definition, identical with the "clean critical" plus five additional radial core elements. Concerning the loading of the control rods, the following remarks are given :

a. a sodium element was inserted in the core region as a SNEAK element tube filled with MASURCA sodium rodlets.

   The Na element in 6D differs therefore from the Na element of 6A by its heterogeneous structure ; the homogeneized compositions are very close to each other (see Appendix).

b. An aluminium element was a SNEAK element tube filled with SNEAK aluminium matrices (as in SNEAK-6A).

c. A tantalum element was a SNEAK element tube filled with SNEAK aluminium matrices containing tantalum rodlets (as in SNEAK-6A).

d. A boron carbide element was a SNEAK element tube filled with boron carbide (as in SNEAK-6A).

e. A stainless steel element was a SNEAK element tube filled with steel blocks of 2.5 cm height. This element, which simulates the steel dummy elements used at the start-up of the PHENIX reactor was not used in SNEAK-6A.

The compositions of all these media are given in Appendix.

## 3.2. EVALUATION OF THE REACTIVITY CHANGES IN 26 GROUPS.

The evaluation of the control rod experiments was performed in 26 groups with the two-dimensional diffusion theory program DIXY in RZ geometry. The cross-section set employed was the MOXTOT set [7]. The experimental steps quoted in Table III-1 were followed as well as possible in the calculations. Every step in Table III-1 corresponds to a criticality calculation. Some experimental steps were omitted in order to save computer time.

In the diffusion theory calculations, the shim and safety rods of SNEAK were considered fully withdrawn (absorber out) with the followers smeared over their respective zones.

Table III-2 contains the core types considered in the calculational evaluation and their criticality constants*.

The core type numbers are distinguished from those in Table III-1 by an apostrophe. In addition to the $k_{eff}$ worths for the core types of Table III-2, the delayed neutron fraction $\beta_{eff}$, was also calculated, in order to obtain an impression of the variation of this quantity with increasing number of radial core elements and absorber rods and in order to be able to convert the calculated reactivity differences $\Delta k$ to cents with as realistic as possible $\beta_{eff}$ worths. The calculated reactivity difference in ¢ between two core types corresponds to the difference of two $k_{eff}$ worths divided by the $\beta_{eff}$ of the final core type.

*The core type 13 (Table III-2) was not realized in the experiment. The worth of four boron carbide elements with respect to four tantalum elements was measured in the subcritical range.

- T A B L E III-1 -

SUMMARY OF THE EXPERIMENTAL RESULTS OF THE CONTROL ROD EXPERIMENTS IN SNEAK - 6D.

| | Core type | Reactivity difference with respect to the preceding core type ($\phi$) |
|---|---|---|
| 1. | "Clean critical" + 5 EL. | 0.0 |
| 2. | "Clean critical" + 5 EL. + 1 Na | - 52.3 |
| 3. | "Clean critical" + 19 EL. + 1 Na | + 139.8 |
| 4. | "Clean critical" + 19 EL. + 4 Na | - 145.4 |
| 5. | "Clean critical" + 21 EL. + 4 Na | + 16.1 |
| 6. | "Clean critical" + 19 EL. + 4 Na | - 16.0 |
| 7. | "Clean critical" + 19 EL. + 4 Al | - 13.2 |
| 8. | "Clean critical" + 25 EL. + 4 Al | + 56.6 |
| 9. | "Clean critical" + 25 EL. + 3 Al + 1 SS | - 5.7 |
| 10. | "Clean critical" + 25 EL. + 4 SS | - 17.6 |
| 11. | "Clean critical" + 23 EL. + 4 SS | - 16.3 |
| 12. | "Clean critical" + 28 EL. + 4 SS | + 40.9 |
| 13. | "Clean critical" + 28 EL. + 4 SS (+) | - 23.8 |
| 14. | "Clean critical" + 26 EL. + 4 SS (+) | - 15.3 |
| 15. | "Clean critical" + 31 EL. + 4 SS (+) | + 42.8 |
| 16. | "Clean critical" + 31 EL. + 3 SS (+) + 1 $B_4C$ (+) | - 220.3 (subcritical measurement) |
| 17. | "Clean critical" + 31 EL. + 1 B4C (+) | + 162.2 |
| 18. | "Clean critical" + 32 EL. + 1 $B_4C$ (+) | + 8.8 |
| 19. | "Clean critical" + 32 EL. + 4 Ta (+) | - 230.2 (subcritical measurement) |
| 20. | "Clean critical" + 52 EL. + 4 Ta (+) | + 181.7 |

TABLE III-1 (continued)

| Core type | Reactivity difference with respect to the preceding core type (¢) |
|---|---|
| 21. "Clean critical" + 57 EL. + 4 Ta (+) | + 44.9 |
| 22. "Clean critical" + 59 EL. + 4 Ta (+) | + 17.0 |
| 23. "Clean critical" + 59 EL. + 4 Ta" (+) | - 6.4 |
| 24. "Clean critical" + 60 EL. + 4 Ta" (+) | + 6.0 |
| 25. "Clean critical" + 61 EL. + 4 Ta" (+) | + 6.8 |
| 26. "Clean critical" + 59 EL. + 4 Ta" (+) | - 12.7 |
| 27. "Clean critical" + 60 EL. + 4 TA" (+) | + 8.1 |
| 28. "Clean critical" + 61 EL. + 4 Ta" (+) | + 7.6 |
| 29. "Clean critical" + 62 EL. + 4 Ta" (+) | + 7.6 |
| 30. "Clean critical" + 62 EL. + 4 $B_4C$ (+) | - 345.7 (subcritical measurement) |
| 31. "Clean critical" + 101 EL. + 4 $B_4C$ (+) | + 342.9 |
| 32. "Clean critical" + 102 EL. + 4 $B_4C$ (+) | + 5.2 |

(+) The upper axial blanket part of these elements is filled with $B_4C$ because of safety reasons.

Footnotes : 1°) "Clean critical" is the core consisting of 369 core elements with $\beta_{eff} = 397.34 \times 10^{-5}$ ;

2°) 4 SS means : the four central MASURCA elements were replaced by elements which were filled with steel blockswithin the core region ;

3°) The 4 Ta" elements have the same homogeneous atom density as the 4 Ta elements, but another cell structure : the small Ta pins are displaced downwards (see Fig. II-2).

- T A B L E III-2 -

CORE TYPES CONSIDERED IN THE CALCULATIONS, WITH THEIR CALCULATED CRITICALITY CONSTANTS AND DELAYED NEUTRON FRACTIONS.

| C o r e | $k_{eff}$ | $\beta_{eff}$ | $\frac{\beta_{eff}}{397.34\text{-}5}$ |
|---|---|---|---|
| 1'. "Clean critical" + 5 EL. | 0.9914/0 | 399.03-5 | 1.004 |
| 2'. "Clean critical" + 5 EL. + 1 Na | 0.9891/1 | 400.15-5 | 1.007 |
| 3'. "Clean critical" + 19 EL. + 1 Na | 0.9956/0 | 404.67-5 | 1.018 |
| 4'. "Clean critical" + 19 EL. + 4 Na | 0.9893/5 | 407.96-5 | 1.026 |
| 5'. "Clean critical" + 19 EL. + 4 Al | 0.9890/3 | 407.54-5 | 1.025 |
| 6'. "Clean critical" + 25 EL. + 4 Al | 0.9918/0 | 409.50-5 | 1.030 |
| 7'. "Clean critical" + 25 EL. + 4 SS | 0.9908/1 | 409.40-5 | 1.030 |
| 8'. "Clean critical" + 28 EL. + 4 SS | 0.9921/7 | 410.35-5 | 1.032 |
| 9'. "Clean critical" + 28 EL. + 4 SS (+) | 0.9911/4 | 410.62-5 | 1.033 |
| 10'. "Clean critical" + 31 EL. + 4 SS (+) | 0.9924/8 | 411.59-5 | 1.035 |
| 11'. "Clean critical" + 31 EL. + 1 $B_4C$ (+) | 0.9893/8 | 411.99-5 | 1.036 |
| 12'. "Clean critical" + 31 EL. + 4 Ta (+) | 0.9777/0 | 416.83-5 | 1.049 |
| 13'. "Clean critical" +101 EL. + 4 Ta (+) | 1.0065/9 | 438.78-5 | 1.104 |
| 14'. "Clean critical" +101 EL. + 4 $B_4C$ (+) | 0.9930/7 | 445.39-5 | 1.120 |

For explanations, see Table III-1.

Therefore, for example, the reactivity of a sodium element with respect to the reference core (core type 1' and core type 2') is :

$$\Delta k \text{ (1 Na} \rightarrow \text{Core reference)} = \frac{0.98911 - 0.99140}{0.0040015} \times 100 \text{ ¢}$$

$$= - 57.2 \text{ ¢}$$

In a similar manner, one obtains from core type 3' and core type 4', the worth $\Delta k$ (4 Na → 1 Na) and from core type 4' and core type 5', the worth $\Delta k$ (4 Al → 4 Na), etc. Through combination of such generated reactivity differences, one gets the control rod worths, which are shown in Table III-3 and which should correspond to the experimental control rod worths of Table III-1. Table III-3 gives a comparison of calculated and corresponding measured control rod worths, in addition the ratio of calculational to experimental worths is shown in the last column.

- T A B L E III-3 -

EVALUATION OF THE CONTROL ROD EXPERIMENTS IN SNEAK-6D IN 26 GROUPS (COMPARISON OF EXPERIMENT AND CALCULATION).

| Types of measurement | Experimental results E | Calculational results C | C/E |
|---|---|---|---|
| 1 Na - core reference | - 52.3 ¢ | - 57.2 ¢ | 1.09 |
| 4 Na - core reference | - 197.7 ¢ | - 210.4 ¢ | 1.06 |
| 4 Al - core reference | - 210.9 ¢ | - 218.3 ¢ | 1.03 |
| 4 SS - core reference | - 234.2 ¢ | - 242.5 ¢ | 1.03 |
| 4 SS (+) - core reference | - 258.0 ¢ | - 267.6 ¢ | 1.03 |
| 4 SS (+) - 4 SS | - 23.8 ¢ | - 25.1 ¢ | 1.05 |
| 1 $B_4C$ (+) - core reference | - 316.1 ¢ | - 342.8 ¢ | 1.08 |
| 1 $B_4C$ (+) - 1 Na | - 263.8 ¢ | - 285.6 ¢ | 1.08 |
| 4 Ta" (+) - core reference | - 552.7 ¢ | - 623.0 ¢ | 1.12 |
| 4 Ta (+) - core reference | - 546.3 ¢ | | 1.14 |
| 4 Ta" (+) - 4 Al | - 341.8 ¢ | - 404.7 ¢ | 1.18 |
| 4 Ta (+) - 4 Al | - 335.4 ¢ | | 1.20 |
| 4 $B_4C$ (+) - core reference | - 898.4 ¢ | - 926.5 ¢ | 1.03 |
| 4 $B_4C$ (+) - 4 Na | - 700.7 ¢ | - 716.1 ¢ | 1.02 |

For explanation, see Table III-1.

For most of the control rod calculations, one half of the reactor was considered, with 81 mesh points in the R-direction and 111 mesh points in the Z-direction.

For each criticality calculation, the source iteration was continued until the difference between two succeeding $k_{eff}$ values was not greater than $5.10^{-5}$ (on most cases, it was $2.10^{-5}$).

Utilizing the smallest $\beta_{eff}$ worth of $399.03.10^{-5}$, one thus finds a maximum numerical error of $\pm$ 2.5 ¢ for the analytical results.

From the tabulated $\beta_{eff}$ worths of Table III-2 for the various core types, one sees that $\beta_{eff}$ does not only increase with the increasing number of radial core elements but also with increasing shut-down reactivity of the central control rod even though the first effect dominates. The latter effect is attributed to the fact that the control rod "displaces" the flux from the center to the outer uranium zone. In relation to the "clean critical" $\beta_{eff}$ increases about 12 % in the case of 101 radial core elements and 4 boron carbide rods in the center.

## 3.3. RADIAL REACTION RATE DISTRIBUTION IN THE PRESENCE OF CONTROL RODS.

In the presence of four central SNEAK elements filled with boron carbide, the radial fission rate distribution per atoms for $U^{235}$ and $U^{238}$ as well as the radial capture rate distribution of $U^{235}$ were measured. They were also measured for the case of four steel block filled SNEAK elements. Within the control rod, the rate measurements were performed with foils. About 10 cm from the center, the measurements were continued with chambers. The measured rates, normalized to a value outside the rod about 8.5 cm off-center, are shown in Fig. III-2, III-3 and III-4 indicated by crosses. The horizontal length of the bars corresponds either to the length of the chamber or to the size of the foil while the length of the vertical bar is a measure for the experimental error. The measurements were made in the core mid-plane.

The solid curves represent the S8-corrected rates per atom which were calculated in (X,Y)-geometry using composition dependent bucklings in 26 groups. The normalization point for the calculated rates is shifted by about half a centimeter to the right with respect to the normalization point of the experimental results.

The fairly good agreement between calculation and experiment inside the control rods was brought about by the S8-correction. The S8-correction outside the rod is negligible.

In the calculational model, $10^{16}$ (atoms/cm$^3$) were added to the compositions of the control rods to calculate the rates. This corresponds essentially to infinite dilution or negligible self-shielding which is a reasonable assumption, except for the $U^{238}$ capture rate in steel, where the self-shielding of the foil cannot be neglected. One can account approximately for the real self-shielding of the foil by introducing for $U^{238}$ in steel an atom density $N_8'$ which is defined in the following way :

$$\frac{1}{2\,d\,.\,N_8} = \frac{\Sigma_t}{N_8'}$$

Here is $d$ thickness of the foil ;

$N_8$ atom density of $U^{238}$ in the foil ;

$\Sigma_t$ total cross-section of steel.

In the present core, one gets :

$$N_8' = 3.66.10^{20}\ (cm^{-3})$$

With this atom density, one achieves a satisfactory agreement between the calculated and measured $U^{238}$ capture rate traverse inside the steel rod. Due to the different self-shielding factors of the rod and the zone Z1M, there appears a discontinuity in the calculation of the $U^{238}$ capture rate traverse at the surface of the rod.

The radial traverses for $U^{235}_{fission}$, $U^{238}_{fission}$ and $U^{238}_{capture}$ in the presence of a $B_4C$ rod show a minimum in the center of the rod which lies about 30-40 % below the maximum of the respective traverse in the Z1M zone. With a central steel rod, the $U^{238}_{capture}$ and $U^{235}_{fission}$ traverses increase towards the rod center while the $U^{238}_{fission}$ traverse is decreased by 40 % with respect to its maximum.

## 3.4. COMPLEMENTARY CALCULATIONS USING OTHER METHODS.

The control rod experiment in SNEAK-6D is evaluated hereabove using diffusion theory, 26 group cross-sections of the MOXTOT set [7] without any condensation, and 2-D (RZ) geometrical models.

The preceding evaluations of control rod experiments in SNEAK-2C [2] and SNEAK-6A (see Chapter 2) were made using also diffusion theory, but NAPPMB [6] cross-sections reduced to four groups and generally other geometrical models such as 2-D (XY) and 3-D (XY/Z).

It was therefore attempted to correlate all these evaluations of control rod experiments. For this purpose, the following complementary calculations were performed :

1°) the cases studied with the 2-D (RZ) model were recalculated with the 2-D (XY) and with the 3-D (XY/Z synthesis) models (paragraph 3.4.1.);

2°) the most typical cases of the 26 group study were recalculated in four groups, with both cross-section sets NAP and MOXTOT. Paragraph 3.4.2. is concerned with $k_{eff}$ and $\Delta k$, paragraph 3.4.3. with the power distributions.

This allows a methodical study of :

- comparable results found with (RZ), (XY) and (XY/Z) models ;
- the condensation effect with different geometrical models ;
- the effect of using the cross-section sets MOXTOT and NAP.

This study completes similar investigations made already for SNEAK-2C [2].

3.4.1. 26 group calculations in the (RZ), (XY) and (XY/Z) geometries.

The series of calculations made using the MOXTOT cross-section set in the (RZ) geometry, was repeated in the (XY) and (XY/Z) geometries. The main goal was to apply the KASY synthesis technique to an experiment evaluation in 26 groups. The (XY) calculations were made mainly in order to generate trial functions for the synthesis.

In order to save computer time, the synthesis calculations with 26 groups were made using only one trial function. Since only fully inserted rods were considered in SNEAK-6D (in the core part), the trial function was chosen representative of a core cross-section.

However, in the SNEAK-6D measurements, there were also some cases in which the upper part of the control rod inside the axial blanket was changed, by filling it with $B_4C$ for safety reasons.

For such changes, a synthesis using only one trial function cannot give relevant results, even when the single core trial function is calculated with modified axial energy-dependent bucklings - which was generally done. For these cases, the $k_{eff}$ are given, but not the reactivity changes.

The $k_{eff}$ obtained by the (XY) and (XY/Z), 26 group calculations, are listed in Table III-4. These values are numerically accurate within $\pm$ 3.10-5.

- T A B L E III-4 -

$k_{eff}$ FROM (XY) AND (XY/Z) 26 GROUP CALCULATIONS.
(SNEAK - 6D, CONTROL ROD STUDY).

| Designation of the cores | | | XY | XY/Z |
|---|---|---|---|---|
| Additional edge elements | Control rod nature | SNEAK elements used for rod simulation | | |
| 4 | - | - | .98913 | .98904 |
| 4 | Na | 1 | .98725 | .98665 |
| 20 | Na | 1 | .99485 | .99444 |
| 20 | Na | 4 | .98913 | .98793 |
| 20 | Al | 4 | .98844 | .98778 |
| 24 | Al | 4 | .99038 | .98975 |
| 24 | SS | 4 | .98901 | .98802 |
| 28 | SS | 4 | .99058 | .98966 |
| 28 | SS | 4* | .98914 | .98753 |
| 32 | SS | 4* | .99092 | .98933 |
| 32 | $B_4C$ | 1* | .98806 | .98801 |
| 32 | Ta | 4* | .97626 | .97597 |
| 100 | Ta | 4* | 1.00446 | 1.00468 |
| 100 | $B_4C$ | 4* | .99081 | .99105 |

*Rod filled with $B_4C$ in upper axial blanket.

One observes the general agreement between keff obtained in (XY) with good bucklings (i.e. energy-dependent values extracted out of preceding RZ calculations) and in (XY/Z) by synthesis : all keff agree within .001, except when there is a change in the upper axial blanket region : the maximum difference then amounts to .0016.

Table III-5 below shows the calculation-over-experiment ratios for the reactivity changes, calculated in the (RZ), (XY) or (XY/Z) geometries.

All the calculated reactivity changes $\Delta k$ have been converted into $\not{c}$ using the $\beta_{eff}$ values determined in each case in (RZ) geometry.

- T A B L E III-5 -

REACTIVITY CHANGES (RZ), (XY) AND (XY/Z)
CALCULATION-OVER-EXPERIMENT RATIOS C/E
(SNEAK-6D CONTROL ROD STUDY).

| Designation of the steps | RZ | XY | XY/Z |
|---|---|---|---|
| Ref - Na. (1)★★ | 1.09 | .90 | 1.14 |
| Ref - Na | 1.06 | .95 | 1.11 |
| Ref - Al | 1.04 | .97 | 1.06 |
| Ref - SS | 1.03 | 1.01 | 1.13 |
| Ref - $B_4C$ (1)★ | 1.08 | 1.08 | 1.10 |
| Na. (1) - $B_4C$ (1)★ | 1.08 | 1.12 | 1.10 |
| Ref - Ta★ | 1.14 | 1.14 | 1.17 |
| Al - Ta★ | 1.21 | 1.26 | 1.24 |
| Ref - $B_4C$★ | 1.03 | 1.04 | 1.05 |
| Na. - $B_4C$★ | 1.02 | 1.06 | 1.03 |

★ Rod filled with $B_4C$ in upper axial blanket.

★★ (1) Means 1 SNEAK element ; in the normal case, four SNEAK elements constitute the simulated control rod.

Comparison XY-RZ :

The (XY) results are as satisfactory as it could be expected from such an approximate geometrical model (see also [2] ). In particular, the reactivity changes ref - Ta and ref - $B_4C$ due to the insertion of absorbers are well predicted (with respect to the RZ results).

For the follower materials (Na., Al, also SS), it was already noticed that the axial leakage of the neutrons within these light media was not well accounted for by the (XY) model. In the previous evaluations ([2], in this report Chapter 2) a universal $B^2$ value was used, and the model gave a large overestimate of the reactivity ref - follower. Here energy- and space-dependent axial buckling values extracted out of (RZ) calculations are used, and the model gives an underestimate of these reactivities

Comparison RZ - XY/Z :

From (RZ) to (XY/Z), one observes a general increase of the calculated reactivity changes by 1 to 5 % (except SS : 10 %).

These increases are summarized in Table III-6 for SNEAK-6D and when available also for SNEAK-6A (from Chapter 2).

They are similar for the two cores in the cases ref - B4C and Na - B4C, but for ref - Na, the effect is larger in SNEAK-6D.

- T A B L E III-6 -

EFFECTS ON REACTIVITY CHANGES (in %),
WHEN USING THE ( XY/Z) MODEL INSTEAD OF THE (RZ) MODEL.

| Designation of the changes | SNEAK-6D | SNEAK-6A |
|---|---|---|
| Ref - Na | + 4.2 | + 1.8 |
| Ref - $B_4C$ | + 1.9 | + 1.3 |
| Na - $B_4C$ | + 1.2 | + 1.1 |
| Ref - SS | + 9.6 | not available |
| Ref - Al | + 2.2 | not available |
| Ref - Ta | + 2.4 | not available |
| Al - Ta | + 2.7 | not available |

For the case ref - SS, the increase (going from RZ to XY/Z) is surprisingly high.

Of course, the RZ and XY/Z results are not expected to coincide, since the square section of the rod is cylinderized in (RZ).

Let us recall the tests of KASY synthesis method as described in the report [5] . The XY/Z synthesis results were compared with direct three-dimensional (X Y Z) results. In the first test presented (ZPR-III-48), the reactivity changes due to the insertion of a central rod were calculated. In particular, the reactivity (Na - B4C) was exactly reproduced by KASY (within the convergence accuracy), even when using only one trial function. The control rod extended axially through the complete system, core + upper and lower axial blankets.

In SNEAK-6A or -2C, the rod extended through the core + the upper blanket. In SNEAK-6D, the rod extends through the core only for Na and through the core + the upper blanket for $B_4C$.

This is probably the reason why the use of only one trial function was sufficient in the test case of [5], and not in SNEAK-6D (SNEAK-6A being an intermediate case).

In summary, the tests performed here confirm that the synthesis method is also applicable in 26 groups and yields satisfactory values for $k_{eff}$ and reactivity changes.

3.4.2. Four group study, $k_{eff}$ and reactivity worths.

The following typical cases have been selected :

1. reference core (clean critical + 5 edge elements) ;
4. four Na (clean critical + 19 edge elements) ;
5. four Al (clean critical + 19 edge elements) ;
7. four SS (clean critical + 25 edge elements) ;
13. four Ta (clean critical + 101 edge elements) ;
14. four $B_4C$ (clean critical + 101 edge elements).

The numbering of the cases corresponds to Table III-2.

The six cases were recalculated in 2-D (RZ) geometry, using four group condensed cross-sections. The cases nr 7 (SS) and nr 14 (B4C) were also recalculated in 3-D (XY/Z), using four group sections.

This program of calculations was performed using both MOXTOT and NAPPMB cross-section sets.

As far as reactivity worths of control rods are concerned, the main results of the 2-D (RZ) calculations are given in Table III-7.

- T A B L E III-7 -

SNEAK-6D REACTIVITY WORTHS, MAIN (RZ) RESULTS.
EFFECT OF NUMBER OF GROUPS AND OF CROSS-SECTION SET.

| Designation of the change | Calculation-over-experiment ratio C/E | | |
|---|---|---|---|
| | MOXTOT 26 groups | MOXTOT 4 groups | NAPPMB 4 groups |
| Na - ref | 1.06 | 1.08 | 1.12 |
| Al - ref | 1.03 | 1.00 | 1.05 |
| SS - ref | 1.03 | 1.03 | 1.07 |
| Ta - ref | 1.14 | 1.13 | 1.20 |
| $B_4C$ - ref | 1.03 | 1.03 | 1.04 |
| $B_4C$ - Na | 1.02 | 1.02 | 1.02 |
| Ta - Al | 1.20 | 1.20 | 1.25 |

They call for comments on :

- the condensation effect ;
- the effect of the cross-section set,

before being compared with the results previously obtained for SNEAK-6A (see Chapter 2) and SNEAK-2C [2] .

Effect of the condensation :

The effect of the condensation from 26 to 4 groups on $k_{eff}$ and on the reactivity worths of control rod materials is shown numerically in Table III-8. Not only the (RZ) MOXTOT results are given, but also the other results of the present study (1-D MOXTOT, 2-D (XY) MOXTOT, 3-D (XY/Z) MOXTOT). In addition, the results of a previous condensation study on a 1-D model, NAP [2], of a full plutonium reactor (SNEAK-2B) with a central rod are included.

- T A B L E III-8 -

EFFECT OF CONDENSATION ON $k_{eff}$ AND $\Delta k$
(SNEAK-6D CONTROL ROD STUDY).

A. Effect on $k_{eff}$ in $10^{-5}$ $\Delta k$ (modification from 26 to 4 groups).

| Designation of the cases | Present study ($Z_1/R_1$/blanket) | | | | Previous study [2] ($Z_1/Z_2$/Blanket) 1-D NAP |
|---|---|---|---|---|---|
| | 2-D (RZ) MOXTOT | 1-D MOXTOT | 2-D (XY) MOXTOT | 3-D(XY/Z) MOXTOT | |
| Ref | + 86 | + 36* | | | + 50 |
| Na | + 72 | + 34 | | | + 39 |
| Al | + 113 | + 33 | | | + 65 |
| SS | + 84 | + 40 | + 65 | + 202 | - |
| Ta | + 119 | + 26 | | | + 65 |
| $B_4C$ | + 86 | + 31 | + 79 | + 207 | + 38 |

* + 45 axially.

B. Effect on reactivity changes in relative %.

| Designation of the changes | Present study ($Z_1/R_1$/blanket) | | Previous study [2] ($Z_1/Z_2$/Blanket) 1-D NAP |
|---|---|---|---|
| | 2-D (RZ) MOXTOT | 1-D MOXTOT | |
| ref - Na | + 1.9 | + 0.3 | + 1.6 |
| ref - Al | - 2.8 | + 0.4 | - 2.5 |
| ref - SS | - 0.4 | - 0.3 | - |
| ref - Ta | - 0.9 | + 0.5 | - 1.2 |
| ref - $B_4C$ | + 0.2 | + 0.2 | + 0.5 |
| Na - $B_4C$ | - 0.2 | + 0.5 | 0. |
| Al - Ta | - 0.1 | + 0.2 | 0. |

The immediate conclusion which may be drawn from Table III-8 is very satisfactory : whatever the geometrical model or cross-section set may be, the largest effect due to condensation from 26 to 4 groups on reactivity changes is smaller than 3 % relative.

It is to be noticed that the figures presented in part B of Table III-8 give rather guiding values than actual corrections. This is due to the fact that the four group calculations have only been repeated for a few typical cases and not for all.

A reactivity such as ref - Na was measured as a sum of several single steps in cores of different sizes. All these steps were simulated in the calculations with 26 groups but not in the calculation with four groups.

The effect of core size on the calculated corrections (26 to 4 groups) was taken into account by converting all the $\Delta k$ in $\not{c}$, using the $\beta_{eff}$ listed in Table III-2.

The dependence of the condensation effect on the geometrical model used is of the same order of magnitude as the effect itself. For the reactivity SS - $B_4C$, this is shown by the following values :

condensation effect SS - $B_4C$ :

+ 0.2 % (2-D,RZ) ; + 0.4 % (1-D) ; - 0.3 % (2-D,XY) ;
+ 0.4 % (3-D,XY/Z).

Effect of the cross-section set (MOXTOT or NAP).

The effect of using the NAP set instead of the MOXTOT set is detailed in Table III-9 below. From the present study, two types of calculations are included : 2-D (RZ) four groups and 1-D 26 groups. Previous results, extracted from [2] for a full plutonium reactor (SNEAK-2B) with a central rod are also given in the Table.

- T A B L E III-9 -

EFFECT OF USING NAP INSTEAD OF MOXTOT ON REACTIVITY CHANGES (IN RELATIVE %). (SNEAK-6D CONTROL ROD STUDY).

| Designation of the changes | Present study ($Z_1/R_1$/blanket) | | Previous study [2] ($Z_1/Z_2$/blanket) 1-D, 26 groups |
|---|---|---|---|
| | 2-D (RZ), 4 groups | 1-D, 26 groups | |
| ref - Na | + 3.8 | + 3.3 | - 1.3 |
| ref - Al | + 4.6 | + 6.9 | 0. |
| ref - SS | + 3.7 | + 3.7 | - |
| ref - Ta | + 6.9 | + 5.5 | - 1.6 |
| ref - $B_4C$ | + 1.5 | + 2.1 | - 3.0 |
| Na - $B_4C$ | + 0.4 | + 1.7 | - 3.6 |
| Al - Ta | + 4.5 | + 4.7 | - 3.0 |

The values given in the first column (2-D, RZ 4 groups) taken as the basic ones in this study, were already used in the preceding Table III-7 for a comparison between MOXTOT and NAP.

Here again, as in the paragraph on condensation effects, it is to be noticed that due to the reduced number of cases considered the effects listed here are to be understood as guiding values rather than exact ones.

The $\Delta k$ differences calculated were also converted into $\not{c}$ and made in this manner rather independent on the core size.

One observes that for SNEAK-6D, the effects on the reactivity worths of control rods when employing NAP instead of MOXTOT, are very similar in one- or two-dimensional models.

One observes different effects in SNEAK-6D which has an U-driver and in SNEAK-2B which is calculated as an all Pu-core ; the sign is even inversed. In both cores, the control rod is placed at the center of the same zone $Z_1$.

This difference is due to the fact that the relative weight of a Pu-zone which is surrounded by a U-zone is calculated higher by NAP than by MOXTOT. Therefore in SNEAK-6D, the transition from MOXTOT to NAP brings an additional increase in the rod worth which is absent in SNEAK-2B.

3.4.3. Four group study, reaction rate distributions.

The $U^{238}$ and $Pu^{239}$ axial fission traverses for the reference core were calculated with the MOXTOT set in 26 and 4 groups, and with the NAPPMB set in 4 groups.

The resulting curves for the $Pu^{239}$ fission, normalized to unity at the core center, are given in Fig. III-5.

There are no very important discrepancies between the different results.

In the core region, the condensation from 26 to 4 groups causes the fission rates to be generally lower by about 2 % (7 % at the core-blanket interface) for $U^{238}$ and by about 1 % ($\sim$ 0 at the interface) for $Pu^{239}$.

In the blanket region, the sign of the difference is reversed.

The use of NAP instead of MOXTOT results in similar changes but generally smaller, except for $Pu^{239}$ fission in blanket.

The following table gives the influence of the condensation and of changing the cross-section set for the spectral indices (in the core center) which are used in order to derive the power distributions from the reaction rate distributions.

The indices are very weakly affected by the condensation. The change of cross-section set has of course a significant effect.

| Indices | 26 groups MOXTOT | 4 groups MOXTOT | 4 groups NAPPMB |
|---|---|---|---|
| $\sigma_{f8}/\sigma_{f9}$ | 0.0279 | 0.0279 | 0.0289 |
| $\sigma_{f5}/\sigma_{f9}$ | 1.1080 | 1.1105 | 1.1513 |
| $\sigma_{c8}/\sigma_{c9}$ | 0.1469 | 0.1461 | 0.1582 |

The radial traverses for the $U^{238}$ and $Pu^{239}$ fission and for $U^{238}$ capture were calculated using the three different methods, for the case of a central control rod made of stainless steel or $B_4C$.

Fig. III-6 shows the $Pu^{239}$ fission traverse through the Z1M fuel region with a central $B_4C$ rod.

As above the condensation has no noticeable influence on the spectral indices. With B4C, the maximum effects on the relative traverses are respectively :

for $Pu^{239}$ fission : 1 % in the central zone,
2 % within the rod ;

for $U^{238}$ fission : < 1 % in the central zone,
1 % within the rod ;

for $U^{238}$ capture : 1 % in the central zone,
3 % within the rod.

With SS, these effects are generally 1 % larger.

Using NAP instead of MOXTOT results in effects on the traverses which are of the same order.

## 3.5. COMPARISON WITH THE SNEAK-6A AND THE SNEAK-2C RESULTS.

### 3.5.1. Prediction of reactivity changes : 1°) Na - $B_4C$.

Table III-10 below compares the predictions of the reactivity changes (i.e. the calculation-over-experiment ratios C/E) associated with the Na - B4C control rod in the experiments SNEAK-6D (this Chapter), -6A (see Chapter 2) and -2C [2].

These predictions correspond to the use of the same methods : diffusion theory, 4 group condensed cross-sections, NAP set, 2-D (RZ) or 3-D (X Y Z) model. Some of these results were directly obtained, others not : appropriate corrections were then applied.

- T A B L E III-10 -

COMPARED PREDICTIONS OF REACTIVITY CHANGES
Na - $B_4C$ IN SNEAK-6D, -6A AND -2C (4 GROUPS).

| Methods and cores | Reactivity changes, calculation-over-experiment ratios C/E | | |
|---|---|---|---|
| | ref - Na | ref - $B_4C$ | Na - $B_4C$ |
| XY / Z geometry | | | |
| SNEAK-6D | 1.16 | 1.06 | 1.03 |
| SNEAK-6A (from Chapter 2) | 1.11 | 1.04 | 1.02 |
| SNEAK-2C (from [2]) | 1.25 | 1.01 | 0.93 |
| RZ geometry | | | |
| SNEAK-6D | 1.12 | 1.04 | 1.02 |
| SNEAK-6A (from Chapter 2) | 1.09 | 1.03 | 1.01 |

For the reactivity ref - Na, one observes that the C/E ratios for SNEAK-6D are 3 to 5 % larger than the -6A ones. The difference does not exceed the experimental accuracy of 5 %. Since the Na follower is made of rodlets in -6D and of platelets in -6A, the difference may be due to the different anisotropy. In addition, the cases -6A and -6D are not exactly comparable as far as the loading of the rod position in the axial blanket region is concerned : in -6A, the follower extends through core and axial blanket, in -6D through the core only. One may therefore conclude that the -6D results confirm well the -6A ones.

For the same reactivity ref - Na, the overestimate of the predictions was much larger in -2C than in -6A and -6D. Here, one must consider that the height of SNEAK-2C was only 60 cm while that of -6A and -6D was 90 cm. If one attributes as in [2] the general overestimate of diffusion calculations for the reactivity difference upon introduction of such a light medium to an inadequate treatment of the axial neutron leakage, the reduction of the overestimate with an increased core height is explainable.

For the reactivity ref - B4C, one observes a good agreement between the C/E values in -6A and -6D, the results for the latter being 1 to 2 % larger. Both results are larger than the -2C values by 3 to 5 %.

3.5.2. Prediction of reactivity changes : 2°) Al - Ta.

Table III-11 below compares similarly the predictions of reactivities associated with the Al - Ta control rod.

- T A B L E III-11 -

COMPARED PREDICTIONS OF REACTIVITY CHANGES Al - Ta IN SNEAK-6D, -6A AND -2C (4 groups).

| Methods and cores | Reactivity changes, calculation-over-experiment ratios C/E | | |
|---|---|---|---|
| | ref - Al | ref - Ta | Al - Ta |
| XY / Z geometry | | | |
| SNEAK-6D | 1.07 | 1.22 | 1.28 |
| SNEAK-6A (from Chapter 2) | 0.89 | 1.04 | 1.13 |
| SNEAK-2C (from [2]) | 1.13 | 1.08 | 1.04 |

The data are scattering widely and no comprehensive explanation can be offered for this as of now. Part of the irregularities may be due to the inconsistent loading of the upper axial blanket in the rod positions.

This loading is as follows :

| | Al follower | Ta absorber |
|---|---|---|
| SNEAK-6D | blanket material | $B_4C$ |
| SNEAK-6A | 20 cm Al, 20 cm $B_4C$ | Ta |
| SNEAK-2C | Al | Ta |

However, the effect of these different loadings in the blanket region (taken into account in the calculation model) should be well below the discrepancies found between the assemblies.

3.5.3. Further comments on the reactivity evaluations.

a. The three assemblies SNEAK-2C, -6A and -6D had neither the same volume ratios plutonium over uranium, nor the same disposition of the plutonium in the core.

For these reasons, the $\beta_{eff}$ values corresponding to the various reference cores were different :

$3.97\ 10^{-3}$ in 6D [3] ;
$4.21\ 10^{-3}$ in 6A [1] ;
$5.10\ 10^{-3}$ in 2C [2].

However, some specific uncertainties remain for the individual assemblies and cause differenciation in a global evaluation :

- in 6D and 6A the uncertainty on the delayed neutron parameters for plutonium, which is placed at the core center ;
- in 2C, the errors caused by the complicated distribution of the different fuel zones in the core.

b. Only comparable results have been gathered here. This paragraph therefore does not include important experiments which were performed in only one assembly, such as :

- partial insertion of control rods (SNEAK-6A, see Chapter 2);
- control rod placed in off-center positions (SNEAK-2C, see [2], or SNEAK-6A, see Chapter 2).

c. The cross-section set which shall give good results in the evaluation of control rod worths must predict equally well the $k_{eff}$ (or the $B_m^2$) of the different core zones. With such a set, no faulty weighting of the different core zones will perturb the evaluation.

3.5.4. Prediction of relative power distributions.

In SNEAK-2C [2], the results of the evaluation of the relative power distributions in the core (core average value normalized to unity) could be summarized in the following way. Using diffusion theory, 4 group cross-sections condensed from the NAP set and the XYZ model leads, in a core without any control rod to an overestimate of 1 % on the radial maximum-to-average power ratio, and also 1 % on the corresponding axial ratio.

In the presence of a central control rod partially or fully inserted, this prediction is worsened by additionally 1 % radially and 2 % axially.

The total effect of using transport instead of diffusion, MOXTOT instead of NAP, and heterogeneity corrected cross-sections instead of homogeneous ones, is about 1 % maximum.

The SNEAK-6A results, extracted from Chapter 2, confirm the 2C results, as well in the radial direction as in the axial one, as far as the maximum-to-average power ratios are concerned.

Similar results are found for SNEAK-6D when the measured fission traverses are combined to a power traverse and compared to calculations.

3.6. COMPARISONS WITH RESULTS OF THE FRENCH EVALUATION.

3.6.1. The SNEAK-6D program, with its central zone made of MASURCA rodlets, has been performed mainly according to the wishes of the French group of MASURCA (Cadarache).

The French group has evaluated the 6D experiments with its own methods. The report [3] details this evaluation and compares French and GfK results. As far as control rod experiments are concerned, the GfK results are precisely those given in sections 3.2. and 3.3. of the present report (diffusion theory, 26 groups, MOXTOT set, RZ geometry).

A summary of these comparisons, extracted out of [3], is given below.

3.6.2. Concerning reactivities, the main results obtained with both approaches are given in Table III-12 below.

- T A B L E III-12 -

SNEAK-6D CONTROL ROD EXPERIMENTS

COMPARISON OF FRENCH AND GfK RESULTS : $\frac{E - C}{C}$ (%)

2-D (RZ) GEOMETRICAL MODEL

| Reactivity against reference (fuel) | French results SETR set, Version 2 25 groups | GfK results MOXTOT set, 26 groups (see Table III-3) |
|---|---|---|
| 1 Na | - 32 | - 9 |
| 1 $B_4C$ | - 15 | - 8 |
| Na | - 21 | - 6 |
| Al | - 19 | - 3 |
| SS | - 16 | - 3 |
| Ta | - 19 | - 12 |
| $B_4C$ | - 12 | - 3 |

The French results call for an explanation of the approach, which is completely different. In the calculational model (2-D,RZ), one reaches criticality by radius iterations. For each insertion of control rod, the critical core radius is therefore determined and the variation of critical radius $\Delta R_C$ due to the insertion of the rod is considered as the basic result, in order to be compared with the experimentally determined variation $\Delta R_E$. The latter is corrected for the presence of partially inserted SNEAK shim rods.

Such an evaluation is made independent of any $\beta_{eff}$ influence, contrary to the evaluations based on rod reactivities in ¢, which are directly affected by the uncertainty of the calculated $\beta_{eff}$ value. One may therefore assume when comparing both evaluations from a basic point of view (improvement of cross-section sets and of method of calculation), that the GfK results are affected by about 16 % uncertainty, resulting from a quadratic combination of 5 % (experimental errors) and 15 % (uncertainty on $\beta_{eff}$) ; on the other hand, the French results should be affected by a very small uncertainty, of 1 to 2 %, resulting from errors in some corrections.

3.6.3. One observes immediately that the French ratios $\frac{E - C}{C}$ are systematically larger than the GfK data (more negative), by the order of 10 - 15 %. The influence of $\beta_{eff}$ on the GfK results could be suspected to be the reason.

If one corrects however the calculated GfK ratios E/C by + 12 %, as suggested from preliminary results of $\beta_{eff}$ investigations recently made in SNEAK-7 [8], the discrepancy C E A - GfK is even increased instead of being reduced.

3.6.4. Besides this general disagreement both evaluations overestimate strongly the tantalum reactivity with respect to the $B_4C$ reactivity. In addition, there is a common main conclusion : for the Na - $B_4C$ rod, the reactivity prediction is better in case of 4 SNEAK elements (size of a SNR control rod) than in case of one.

* * *

# 4.- CONCLUSIONS WITH RESPECT TO THE DESIGN OF THE SNR CONTROL RODS.

## 4.1. SCOPE OF THE PRESENT CONCLUSIONS.

Clearly the results of the evaluations presented in this report are useful for testing cross-section sets and methods of calculation. As most typical examples, the capture cross-sections and probably also the inelastic scattering cross-sections of tantalum should be revised and the neutron diffusion in the axial direction within a sodium follower (or any other diluent) should be thoroughly investigated.

Besides of such long term goals, the evaluations have also a short term purpose : the results obtained with the presently running SNR calculations are to be checked and corrected.

The present chapter considers the application to the SNR of the standard methods used for its calculation and gives :

- correction factors to be applied to the calculated SNR reactivity worths and power distributions, based on the calculation-to-experiment ratios found in SNEAK ;
- the accuracy of the SNR predictions, derived from a combination of uncertainties from experiment and extrapolation.

The standard methods of calculation considered correspond to the use of diffusion theory, four group cross-sections condensed from the NAPPMB set and the 3-D (XYZ) synthesis method.

Let us note that it was preferred here to consider only the application of a standard method, relatively simple in some aspects (e.g. diffusion theory), which was consistently applied to the whole evaluation and not to consider more elaborate methods (e.g. transport programs) : the latter, although also used in the evaluation, were not employed for all the cases and the corresponding uncertainty therefore becomes higher.

## 4.2. REACTIVITY WORTH OF A CENTRAL CONTROL ROD.

### 4.2.1. Correction factors.

A detailed comparison of the SNEAK-6A and -6D results has been given in Chapter 3, and particularly, in Table III-10.

The two assemblies represent a rather good mock-up of the SNR conditions for the case of a central control rod. The simulation is good except for the heterogeneous structure of the materials. The SNR fuel is made of pins which are similar to the MASURCA rodlets, but have a smaller diameter.

It seemed adequate from the comparison 6A / 6D (Chapter 3), to take as correction factors for the SNR calculations the average of the calculation-to-experiment ratios obtained in 6A and 6D, under the condition that the latter ratios agree within the experimental accuracy.

This condition is fulfilled for the reactivities associated with the control rod Na - $B_4C$, which is precisely the present rod concept for the SNR.

The condition is not fulfilled for the other control rod investigated, Al - Ta, and for this rod, no correction factors can be derived as long as the observed discrepancies have not found an explanation.

The correction factors for the rod Na - $B_4C$, placed in the central position, are given in the following Table IV-1. Only the reactivities fuel-Na, fuel-$B_4C$ and Na - $B_4C$, corresponding to the full rod insertion, are considered here. For partial rod insertion, one should refer to the detailed text, and particularly to Chapter 2, Table II-3 (four insertion depths in SNEAK-6A) and Fig. II-3 (characteristic curves).

- T A B L E IV-1 -

CENTRAL CONTROL ROD
CORRECTION FACTORS AND UNCERTAINTIES.

| Reactivity changes | Correction factors : Calculation-to-experiment ratios | Uncertainties (in ±) |
|---|---|---|
| Fuel - Na | 1.14 | .08 |
| Fuel-$B_4C$ | 1.05 | .06 |
| Na - $B_4C$ | 1.03 | .06 |

4.2.2. Accuracy of the predictions.

Four distinct sources of uncertainties are distinguished : the experiments, the influence of $\beta_{eff}$, the different conditions in SNEAK and SNR, the calculation methods.

The experiments are affected by a general uncertainty of ± 5 % as detailed in Chapter 1. This uncertainty applies to any reactivity step measured, and should also be used for reactivities obtained by summing a number of steps. But, since SNEAK-6A and -6D represent two independent series of measurements, one may justify dividing the figure of 5 % by $\sqrt{2}$, which gives 3.5 %.

When the actual deviations observed between the 6A and 6D results are considered, they indicate an agreement within this error of 3.5 %. The deviation is somewhat higher for Na-fuel (2.5 %) than for fuel-$B_4C$ and Na-$B_4C$, which may be due to the heterogeneity effects.

The effective delayed neutron fraction $\beta_{eff}$ has a direct influence on a comparison of experimental and calculated reactivities, since the calculated $\Delta k$ are converted into ¢ by means of a calculated value of $\beta_{eff}$.

The calculation-to-experiment ratios are therefore affected by the $\beta_{eff}$ uncertainty, which may be as high as 5 % for uranium cores and 20 % for plutonium cores.

However in an extrapolation from SNEAK to SNR, uncertainty is introduced only by the need to calculate the difference of the $\beta_{eff}$ in both reactors. The calculated $\beta_{eff}$ values are :

0.30 % $\Delta k$ in SNR

0.42 % $\Delta k$ in SNEAK-6A
0.40 % $\Delta k$ in SNEAK-6D

the differences being due to the different contents of plutonium and uranium fuels.

One realizes that only a small fraction of the uncertainty in $\beta_{eff}$ enters the uncertainty of the extrapolation. A more detailed examination shows that it amounts to about 2 % of the rod worth to be determined. This is an uncertainty of systematic nature (to be added separately).

The relative variations of $\beta_{eff}$ throughout the experiments have been considered in the evaluation and do not contribute significantly to the uncertainty.

Let us note that as soon as definite conclusions from $\beta_{eff}$ investigations such as those made in SNEAK-7 [8] will be available, all results will be corrected accordingly and only a very small uncertainty due to $\beta_{eff}$ will remain.

The different conditions in SNEAK and SNR or the accurate knowledge thereof (e.g. for the absorber masses), are assumed to contribute in general at most 0.5 %.

However, a particular feature of SNEAK-6 is that the central Pu zone is surrounded by a uranium zone. The calculated rod worths depend on the calculated weight distribution between the different fuel zones which, in turn, depends on the cross-section set used (see also paragraph 3.4.2.). A 1 % systematic uncertainty is associated with this imperfect simulation of the SNR conditions.

The difference in heterogeneity was already underlined for the case of SNEAK-6A and -6D, in particular for the sodium rod. For the reactivity fuel-sodium, a 2 % uncertainty of systematic nature is retained as due to the different heterogeneous structure or a sodium follower in SNEAK-6 and in SNR. Not considered here are any factors due to the technological nature of the SNR itself (for example, inhomogeneous repartition of the fuel).

The influence of the calculation methods could, in principle, be set equal to zero, as long as one considers the application of the same well defined methods to SNEAK and SNR. However, a 0.5 % uncertainty is retained, which represents an allowance for small deviations from the standard method, for example in the choice of the mesh intervals or of the trial functions in the synthesis. (Such effects have already been carefully investigated).

All partial uncertainties are quadratically combined, except the $\beta_{eff}$ error, the weight distribution error and the heterogeneity effect for the sodium follower, which are separately treated as systematic. The resulting figures are quoted in Table IV-1 hereabove.

## 4.3. REACTIVITY WORTH OF ONE OR SEVERAL OFF-CENTER CONTROL RODS.

4.3.1. The preceding conclusions all are extracted from the SNEAK-6 evaluations, which concerned only the case of a central control rod.

In the previous SNEAK-2C evaluations [2], rods positioned at different radii were considered and investigations were also made about the effects of neighbouring rods (shadowing effects).

4.3.2. In SNEAK-2C, when a single control rod was displaced from the core center (plutonium zone $Z_1$) radially towards the boundary of the core zones $Z_1$ and $Z_2$ - which was the location generally planned for a bank of compensating rods in the SNR - the predictions (C/E) of the reactivities were changed as follows :

Fuel-Na : reduction of the overestimate to about one third ;
Fuel-$B_4C$ : + 2 %.

The correction factors obtained for the central rod, of respectively 1.14 (fuel-Na) and 1.05 (fuel-$B_4C$), are therefore to be modified to 1.05 (fuel-Na) and 1.07 (fuel-$B_4C$) ; for Na-$B_4C$ the correction factor becomes 1.07.

One has to take into account an additional source of error, associated with the measurements performed in SNEAK-2C. These sector-type experiments were characterized by a dissymmetry of the flux and importance distribution in the core : the insertion of rods in the plutonium fuelled sector created a significant increase of the relative weight of the uranium fuelled zone. As stated already before the calculated weight distribution between the different fuel zones depends on the cross-section set used. The estimated increase of the uncertainty due to this mechanism is reflected in the single rod data of Table IV-2 below.

- T A B L E IV-2 -

OFF - CENTER CONTROL RODS
CORRECTION FACTORS AND UNCERTAINTIES

| Reactivity changes | Correction factors C/E | | Uncertainties (in ±) |
|---|---|---|---|
| | A single rod | Several rods (shadowing effects) | |
| Fuel - Na | 1.05 | 1.05 | .10 |
| Fuel-$B_4C$ | 1.07 | 1.04 | .08 |
| Na - $B_4C$ | 1.07 | 1.03 | .08 |

4.3.3. Some indications on the shadowing effects of control rods placed at certain distance from each other at the boundary of the two core zones may also be extracted from the SNEAK-2C evaluations [2].

In case one rod is half-inserted (Na → Na-$B_4C$) in the presence of two neighbouring rods half-inserted, and then is further fully inserted (Na-$B_4C$ → $B_4C$) with the two neighbouring rods fully inserted, the calculation-to-experiment ratio is each time 3-4 % smaller than for the case of a single rod. These cases simulate most closely the insertion of a rod bank and indicate that in such a situation, the correction factors derived from single rod experiments may be somewhat reduced.

For other cases, such as differential insertions, different results were found. A clear concept how the correction factors depend on the insertion policy could not yet be derived from the evaluations.

4.4. RELATIVE POWER DISTRIBUTION IN A CORE CONTAINING CONTROL RODS.

4.4.1. Correction factors.

The evaluations made in SNEAK-6A and -6D (one central control rod) as well in SNEAK-2C (one central rod, one or several off-center rods) lead finally to the same common conclusions, gathered in Chapter 3 (paragraph 3.5.4.). The relative power distribution is considered in the core regions, normalized to the core average value.

The standard methods of calculation used give, in a core without any control rod :

- an overestimate of 1 % on the radial maximum to average power ratio ;
- an overestimate of 1 % also on the axial corresponding ratio.

In the presence of one or several control rods partially or fully inserted, these predictions are worsened by additionally :

1 % radially ;
2 % axially,

as indicated in the following Table IV-3.

- T A B L E IV-3 -

RELATIVE POWER DISTRIBUTIONS IN THE CORE
CORRECTION FACTORS AND UNCERTAINTIES

| Maximum to average power ratios | Correction factors C/E | Uncertainties (in ±) |
|---|---|---|
| Radially, core without rod | 1.01 | .02 |
| Radially, core with rods | 1.02 | .03 |
| Axially , core without rod | 1.01 | .02 |
| Axially,, core with rods | 1.03 | .03 |

4.4.2. Accuracy of the predictions.

Four sources of uncertainties are distinguished : the measurements, different conditions in SNEAK and S N R, the calculation methods and the influence of the control rods.

The measurements with fission chambers are characterized in the core region by a statistical accuracy of 0.3 % for $Pu^{239}$. Using spectral indices measured within 2 % in order to add to the power density due to $Pu^{239}$ the part due to the other isotopes contributes in the core another 0.3 %. When adding 0.2 % for streaming effects in the measuring channel, evidenced in the outer part of the core by comparison with foils and 0.1 % for small positioning errors of the chambers, one arrives at 0.9 %.

The different conditions in SNEAK and SNR, such as heterogeneity effects in SNEAK, different fuel compositions or the presence of fission products, are assumed to contribute 1 % of uncertainty in the extrapolation of SNEAK results to SNR design ; this also includes the simulation of SNR hexagonal boundaries in the square SNEAK lattice.

Assuming the same basic calculation methods are applied to SNEAK and SNR, a 0.3 % uncertainty is retained as maximum allowance for method deviations.

When cases with control rods are considered additional errors are introduced by uncertainties in position and composition of the rods and in position of chambers or foils in the vicinity of the rods ; these errors are assumed to amount to about 1 %.

The linear combination of all partial uncertainties leads to the figures quoted in Table IV-3 hereabove.

## 4.5. TRENDS FOR FUTURE CONTROL ROD EXPERIMENTS.

Performing the control rod experiments SNEAK-2C, -6A and -6D has reduced the uncertainties in SNR control rod worths and power distributions by more than a factor of 2. In particular, the prediction of power distribution in the core with control rods is well ascertained.

Concerning rod reactivities, two major causes of uncertainties remain. The first one is related to $\beta_{eff}$ ; it is hoped that the situation will be improved after the final evaluation of the $\beta_{eff}$ investigations made in SNEAK-7.

On the other hand, since only a limited number of rod positions was until now investigated in SNEAK, the case of the full system of control rods of the SNR lay-out was not yet really simulated ; in particular, the shadowing effects are still insufficiently known.

It is therefore foreseen to perform a new program of control rod experiments in the next assembly SNEAK-9, core A. This core will be of a cylindrical symmetry and contain the whole control rod system of the SNR. In order to avoid any geometrical disturbances and due to the plutonium shortage, this mock-up will contain only uranium fuel, distributed in two enrichment regions. In view of an adequate SNR simulation, the uranium cell compositions are chosen so that the $B^{10}$ material worth and the core diffusion coefficient match closely those of the plutonium fuelled SNR.

Various rod configurations and insertion policies will be considered. This will provide support for the SNR optimization studies [9]. One main aim of the experiments will be indeed to measure the influence of the control rod policy on the power distribution, or in other words, to find an optimal control rod policy as far as power flattening is concerned.

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* *

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## - L I S T  O F  F I G U R E S -



* * *

## - A P P E N D I X -

SNEAK-6A AND -6D ATOMIC COMPOSITIONS ($10^{22}$ at/cm$^3$).

| | Elements | 1 $Z_{1A}$ SNEAK | 2 $Z_1$ MASURCA | 3 $R_{1A}$ SNEAK | 4 Breeder blanket (axial) | 5 SNEAK blanket (radial) |
|---|---|---|---|---|---|---|
| 1 | $U^{235}$ | .0046 | .0016 | .1877 | .0050 | .0162 |
| 2 | $U^{238}$ | .6743 | .5930 | .6516 | .6879 | 3.9940 |
| 3 | $Pu^{239}$ | .1215 | .1210 | - | - | - |
| 4 | $Pu^{240}$ | .0109 | .0112 | - | - | - |
| 5 | $Pu^{241}$ | .0010 | .0011 | - | - | - |
| 6 | $Pu^{242}$ | .00005 | .00013 | - | - | - |
| 7 | Cr [1] | .3236 | .3442 | .3695 | .2677 | - |
| 8 | Fe | 1.1916 | 1.2548 | 1.2386 | .9030 | - |
| 9 | Ni [2] | .2268 | .1764 | .1921 | .1432 | .0412 |
| 10 | O | 1.1998 | 1.1597 | .4841 | 1.3848 | - |
| 11 | C | .0049 | .0013 | .6523 | .0047 | - |
| 12 | Na | .8552 | .8856 | .2770 | .6445 | - |
| 13 | Al [3] | .0004 | - | .8530 | .2951 | .5155 |
| 14 | Si | .0153 | .0046 | .0205 | .0149 | .0012 |
| 15 | Ti | .0009 | - | .0040 | .0001 | .0001 |
| 16 | $B^{10}$ | - | - | - | - | - |
| 17 | $B^{11}$ | - | - | - | - | - |
| 18 | Ta | - | - | - | - | - |

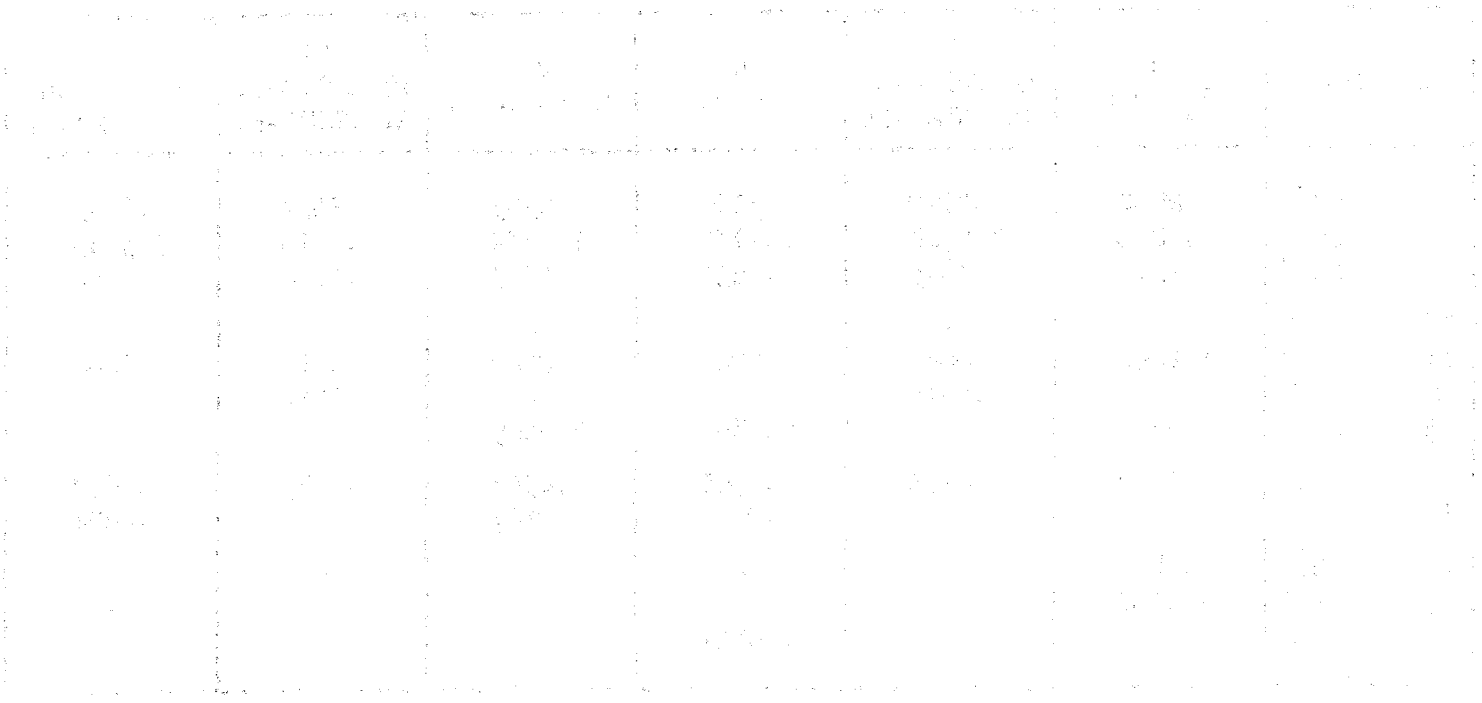
1) Mn has been assimilated to Cr
2) Co has been assimilated to Ni
3) Mg has been assimilated to Al

TABLE (continued)

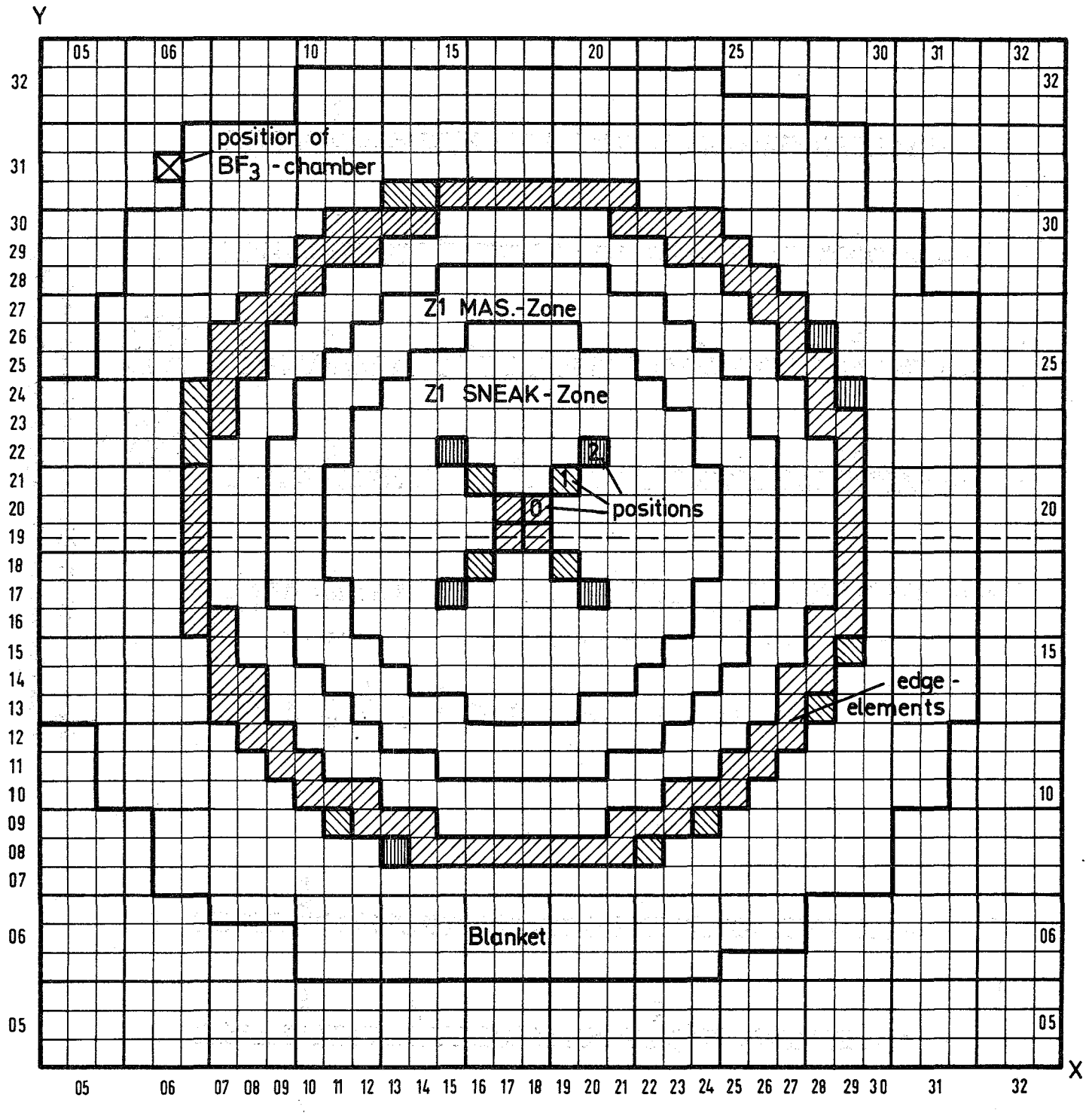
| Elements | | 6 $B_4C$ rod | 7 Na follower in SNEAK-6A | 8 Ta rod | 9 Al follower | 10 Na follower in SNEAK-6D | 11 steel rod in SNEAK-6D |
|---|---|---|---|---|---|---|---|
| 7 | Cr[1] | .2552 | .3170 | .3505 | .3505 | .2367 | 1.6359 |
| 8 | Fe | .8293 | 1.0728 | 1.1789 | 1.1789 | .8316 | 5.5353 |
| 9 | Ni[2] | .1176 | .1845 | .1633 | .1633 | .1148 | .0758 |
| 10 | O | - | - | - | - | - | - |
| 11 | C | 1.2096 | .0043 | .0029 | .0029 | .0014 | .0202 |
| 12 | Na | - | 1.6611 | - | - | 1.7714 | - |
| 13 | Al[3] | .0504 | - | 2.5868 | 4.3865 | - | - |
| 14 | Si | .0172 | .0173 | .0365 | .0505 | .0045 | .0632 |
| 15 | Ti | - | - | .0117 | .0167 | - | .0284 |
| 16 | $B^{10}$ | .9449 | - | - | - | - | - |
| 17 | $B^{11}$ | 3.8762 | - | - | - | - | - |
| 18 | Ta | - | - | 1.6537 | - | - | - |

1) Mn has been assimilated to Cr
2) Co has been assimilated to Ni
3) Mg has been assimilated to Al

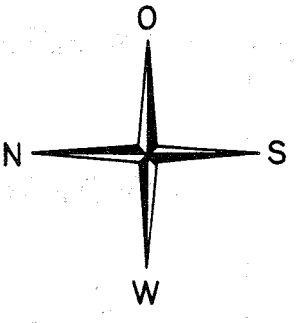
* * *



# - F I G U R E S -



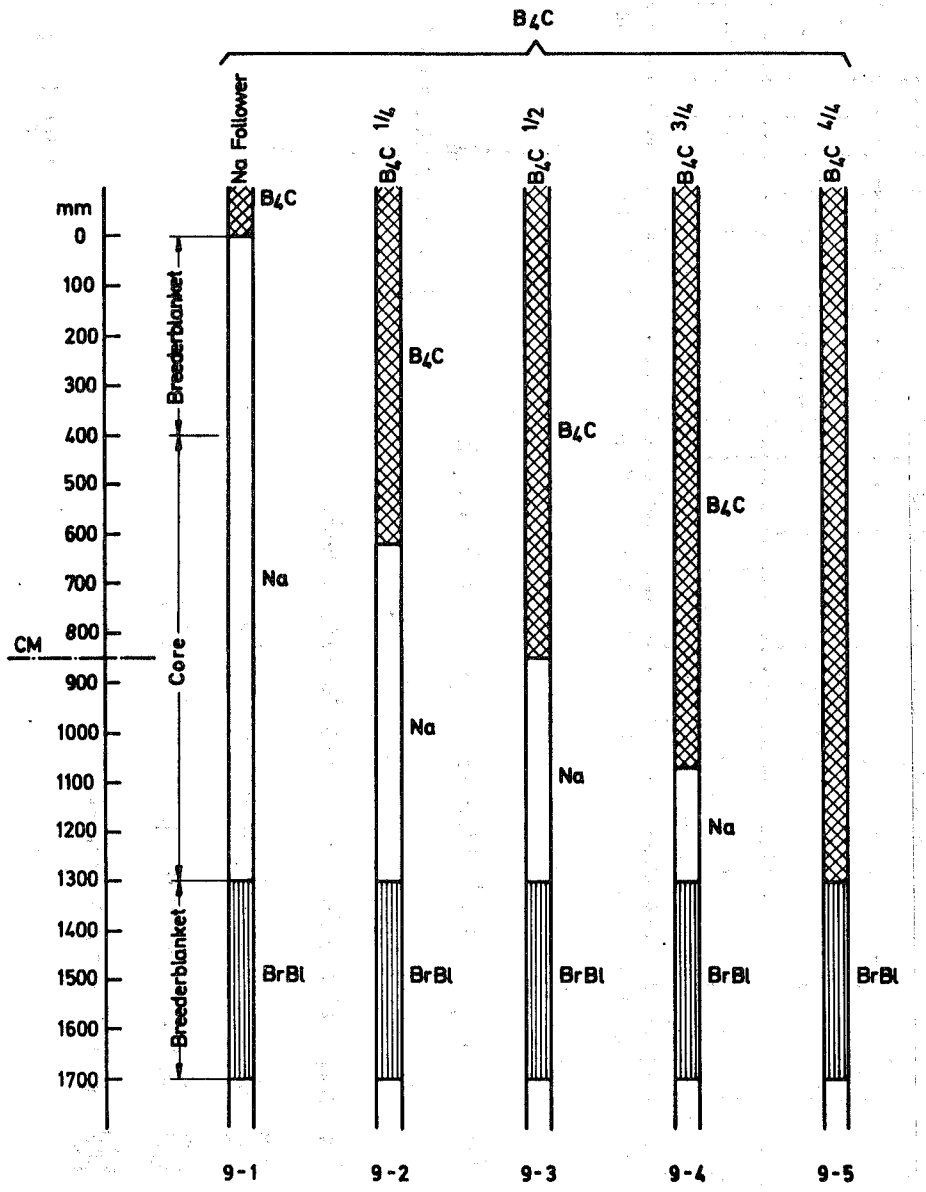

SNEAK-6A $B_4C$-Rod splitted up in two steps




The edge elements correspond to respectively :

- $B_4C$, reference position (0)
- $B_4C$, first split-up (1)
- B C, second split-up (2)

Fig. II-1



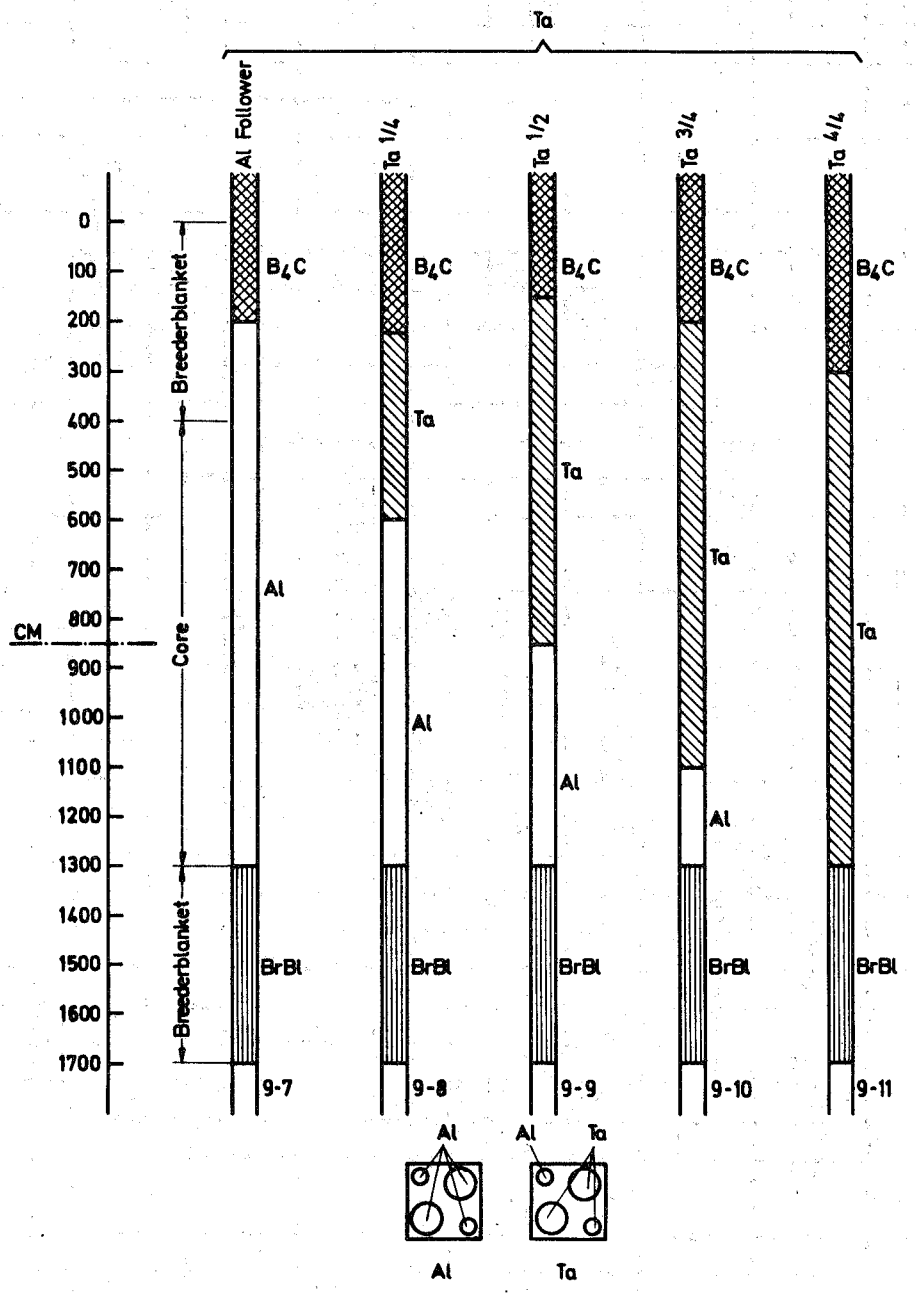




**Fig. II - 2 SNEAK - 6A Control Rod Experiment**
**Axial Distribution of the Materials Na - $B_4C$ and Al - Ta**

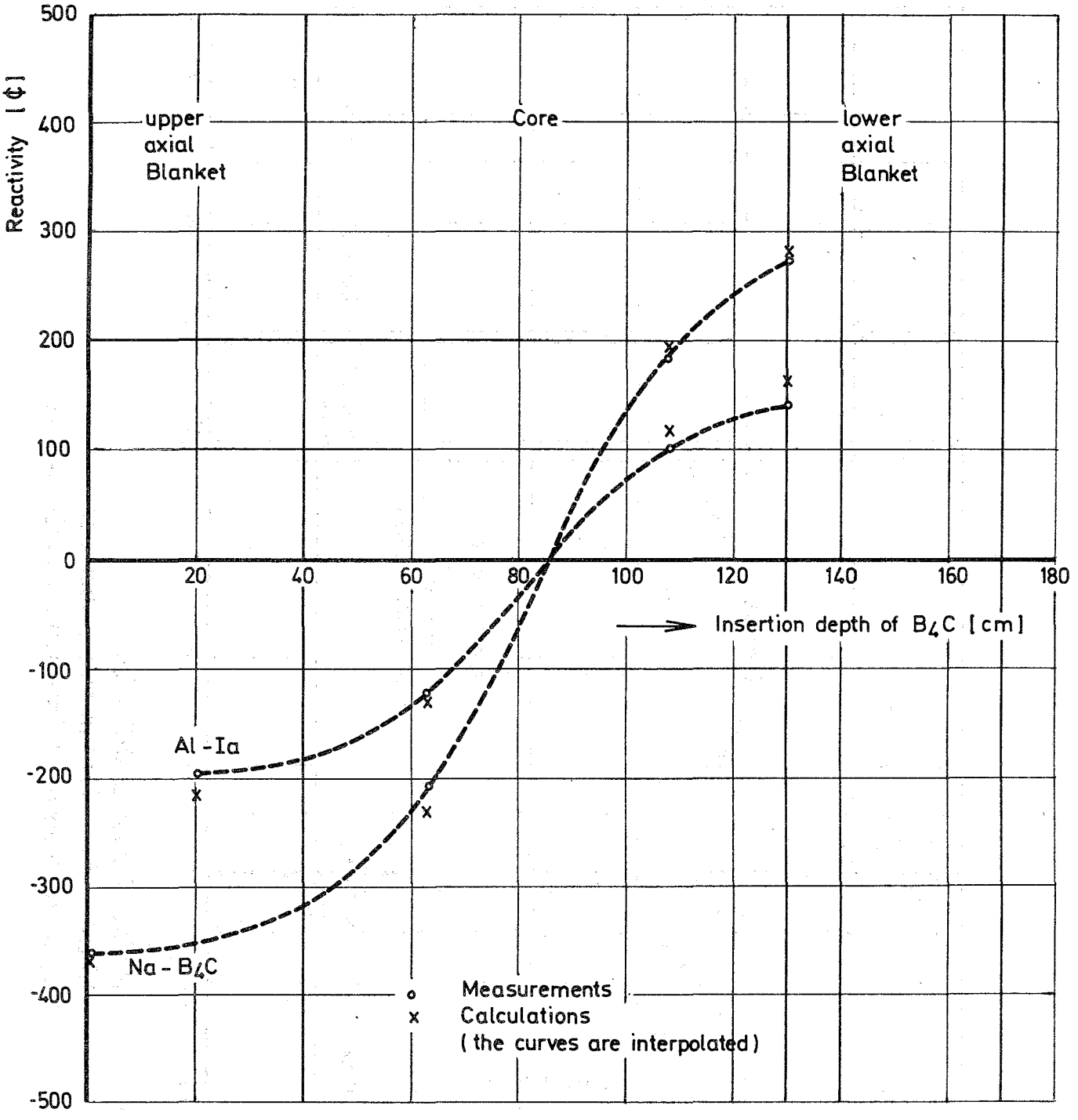



Fig. II - 3 SNEAK - 6A

Characteristic Curves Na - $B_4C$ and Al - Ta

( 4 SNEAK Elements )

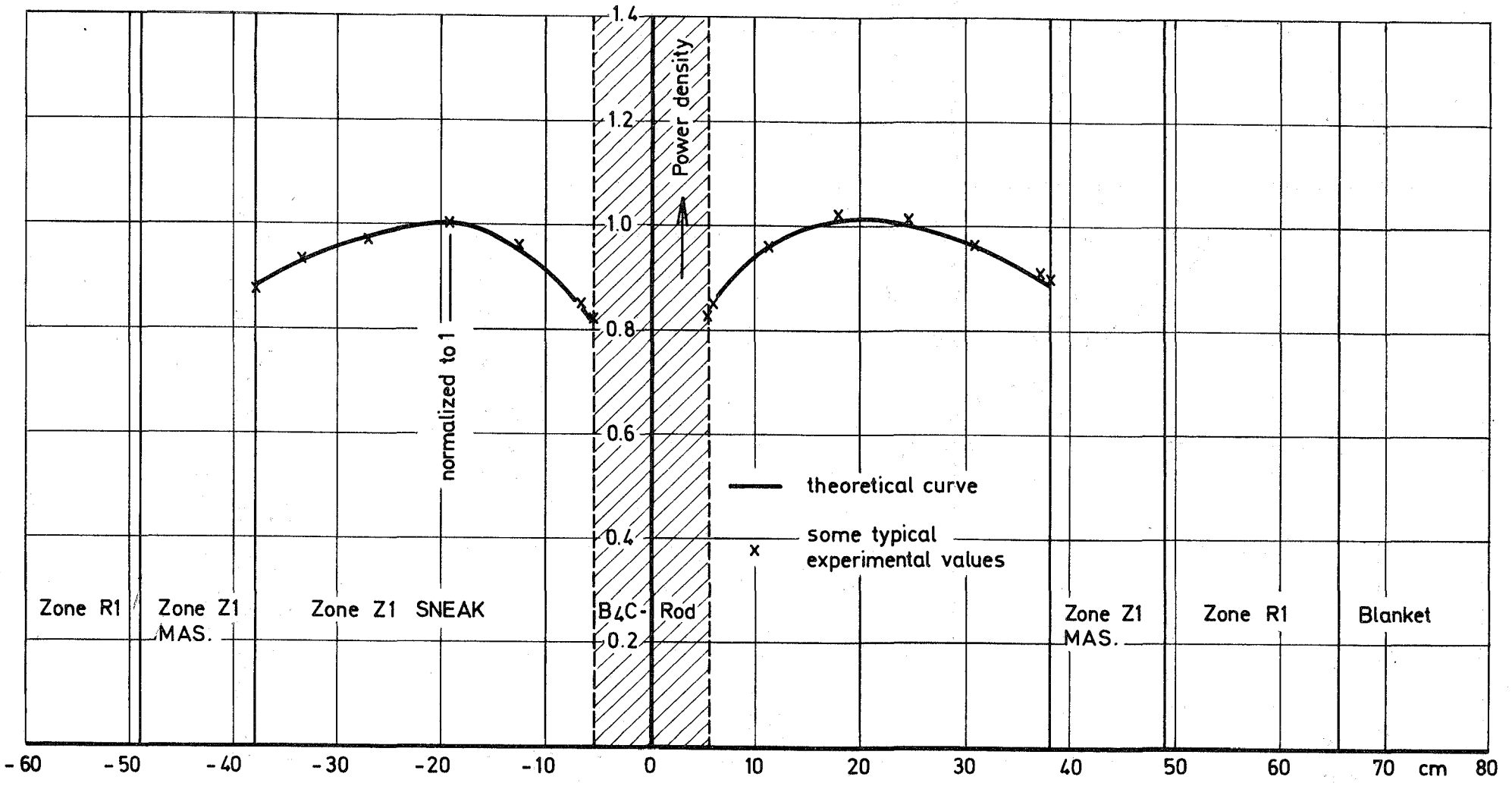



Fig. II - 4 Radial Power Distribution
SNEAK - 6A Control Rod Experiment
$B_4C$ full inserted ( Core with 468 Elements )

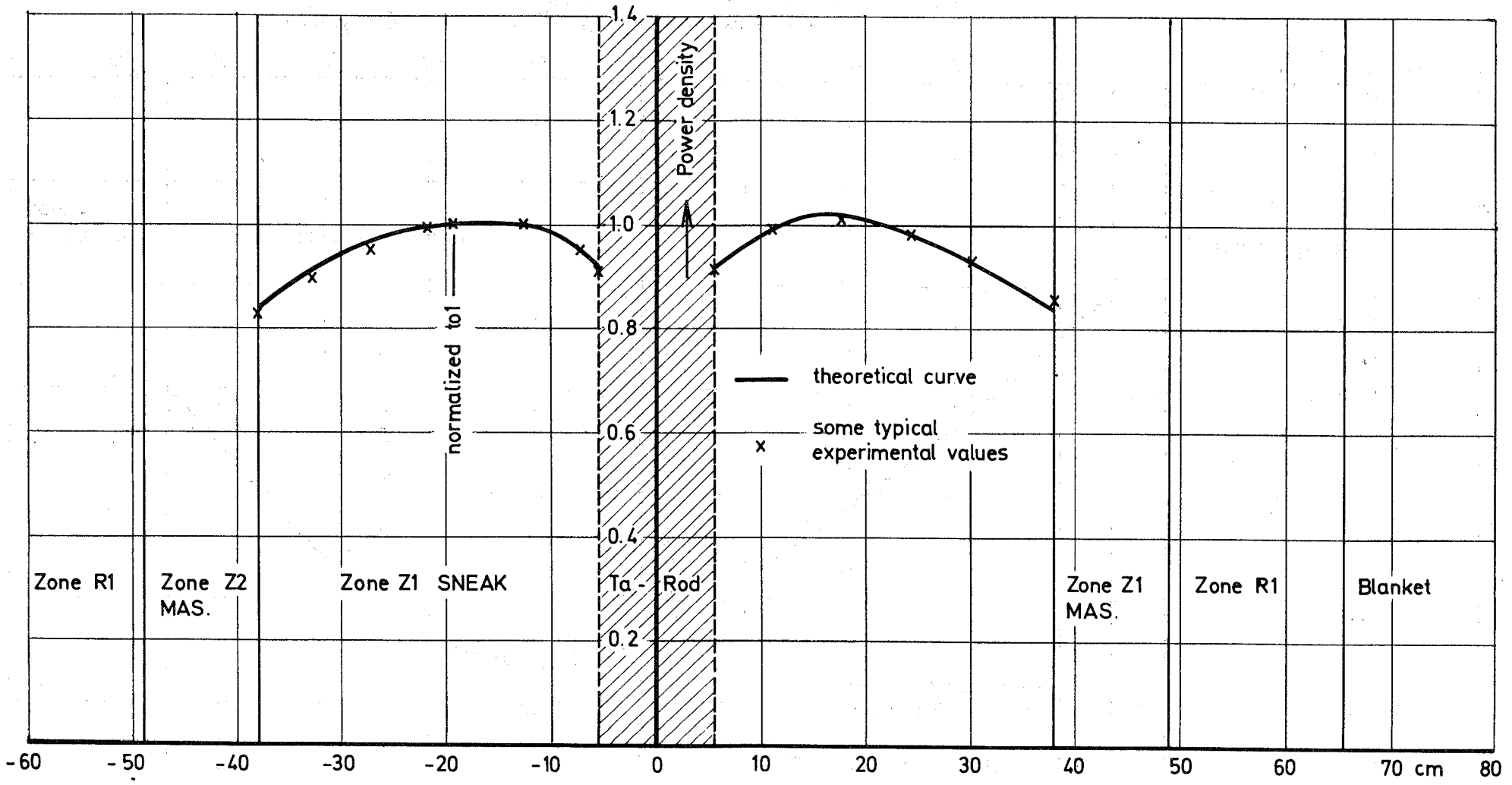



Fig. II - 5 Radial Power Distribution
SNEAK - 6A Control Rod Experiment
Ta full inserted ( Core with 432 Elements)

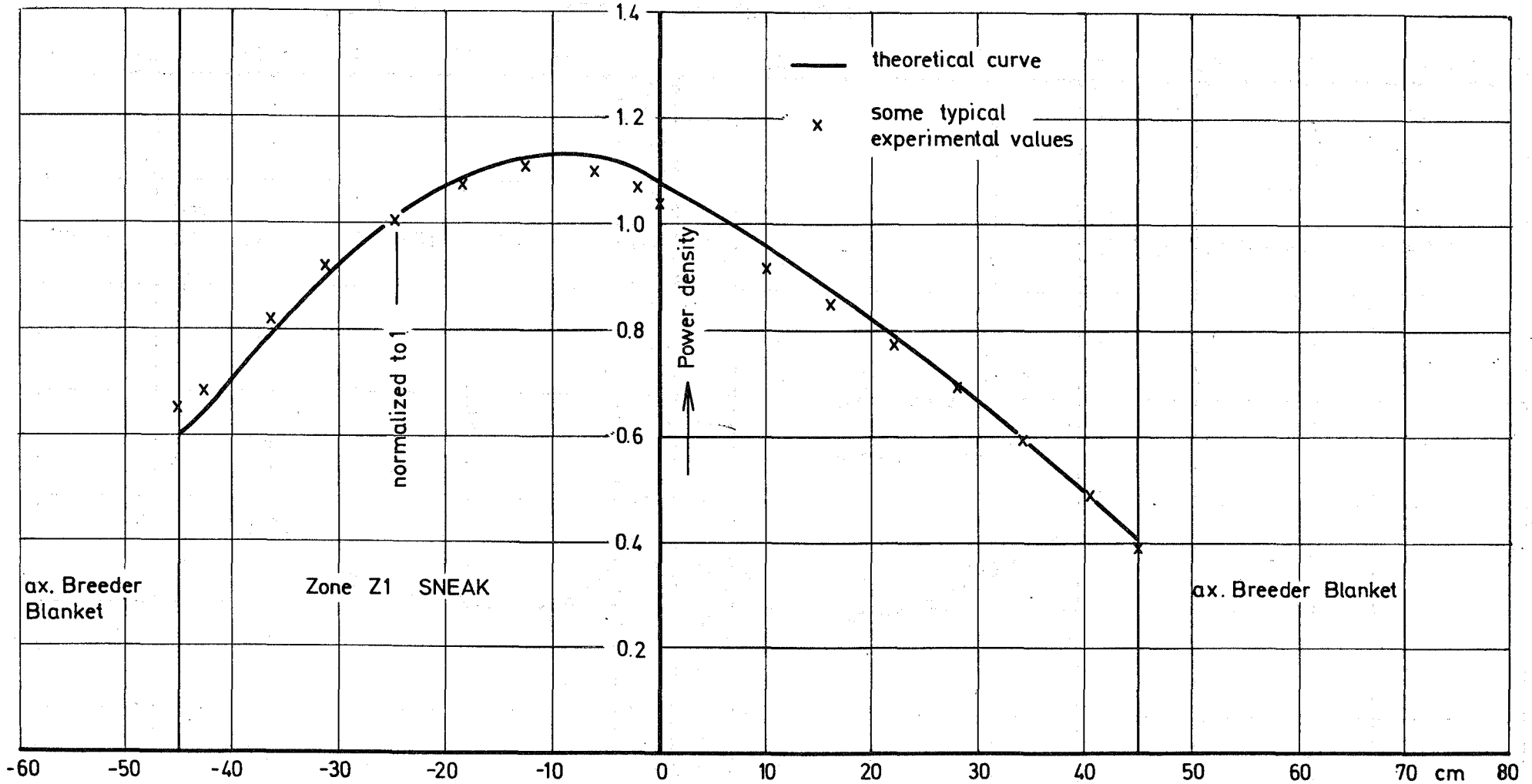



Fig. II -6 Axial Power Distribution
SNEAK-6A Na-$B_4C$ : $B_4C$ half inserted
(Core with 458 Elements)

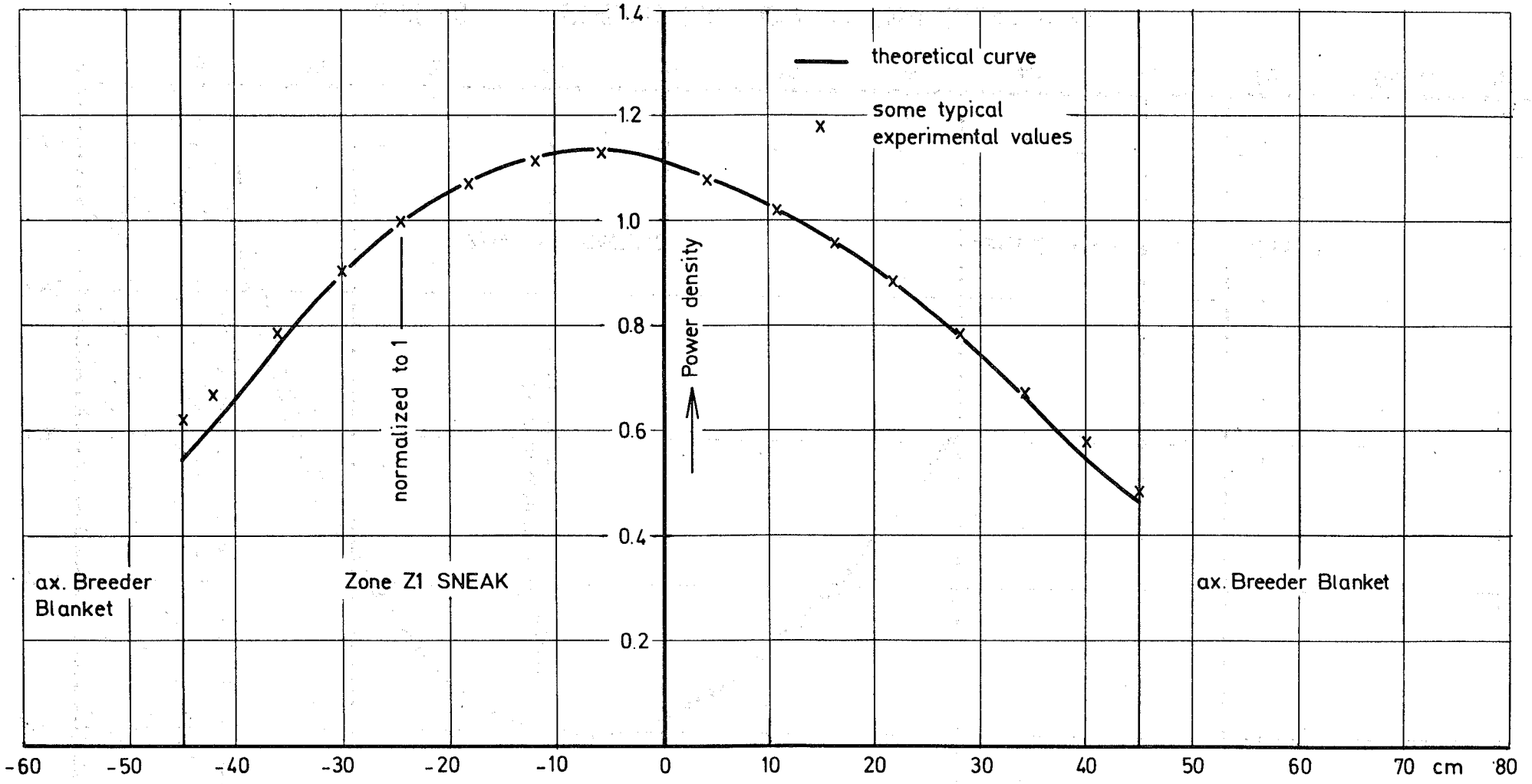



Fig. II-7 Axial Power Distribution
SNEAK-6A Al-Ta ( Ta half inserted )
(Core with 412 Elements )

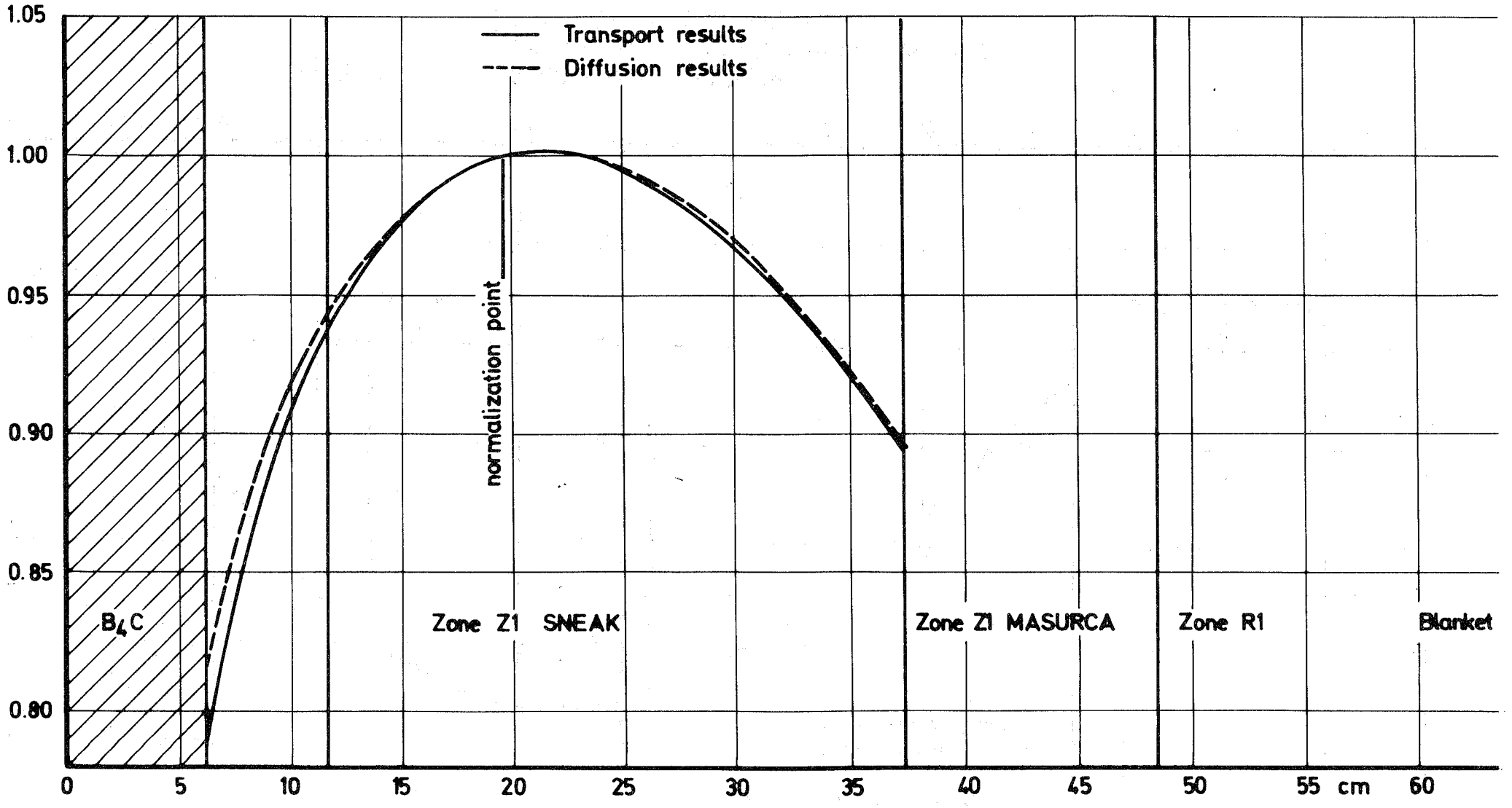



Fig. II - 8 SNEAK-6A Control Rod Experiment
Traverse : Radial $B_4C$
Pu 239 Fission

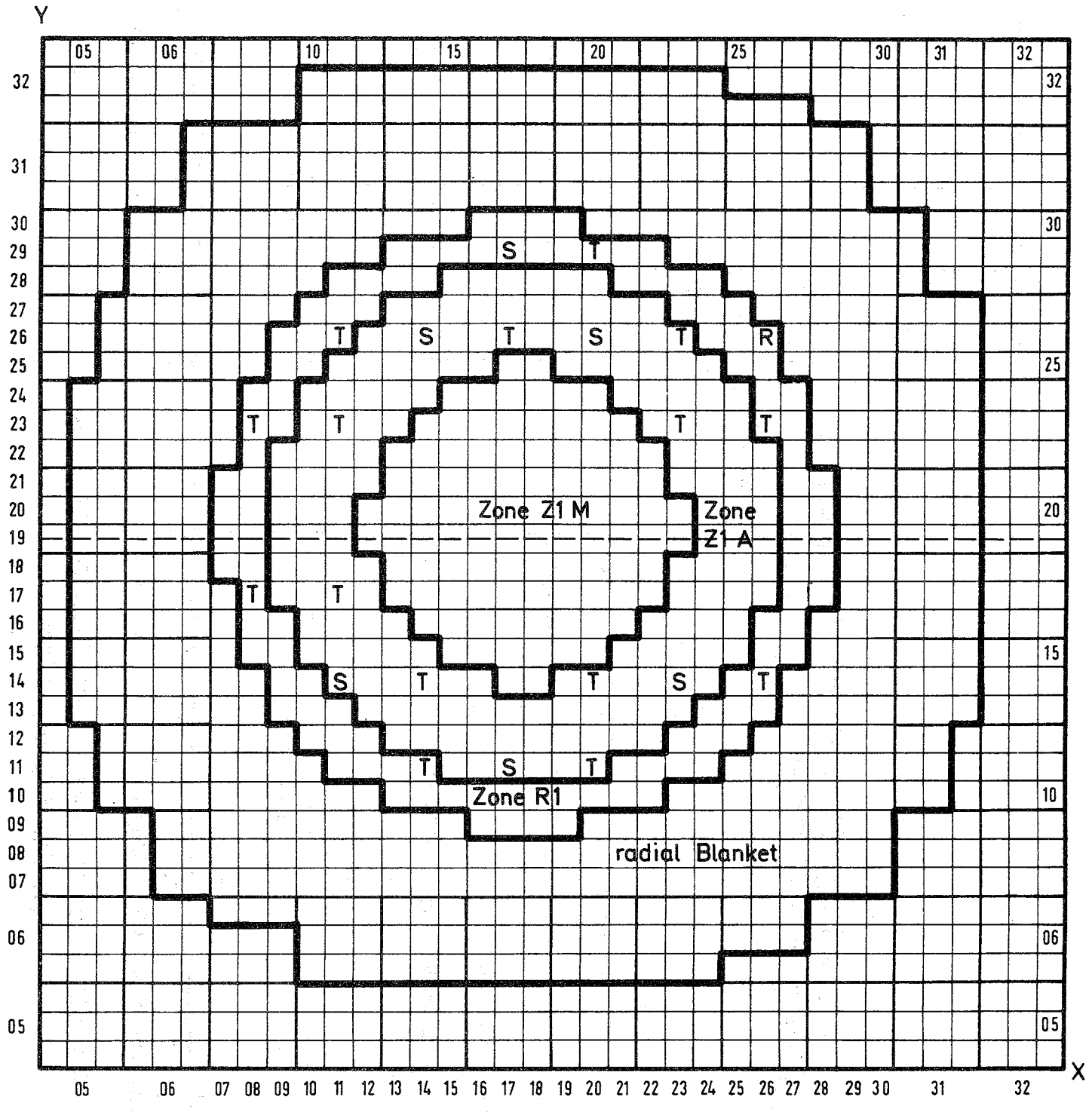



Fig. III - 1

SNEAK - 6D Cross Section of the Clean Critical Core

Zone Z1 M (96 fuel elements )
Zone Z1 A (147 fuel elements, 8T-rods, 5S-rods )
Zone R1 (104 fuel elements, 7T-rods, 1S-rod, 1R-rod )

S = safety rod
T = shim rod
R = regulating rod
--- radial chamber traverse

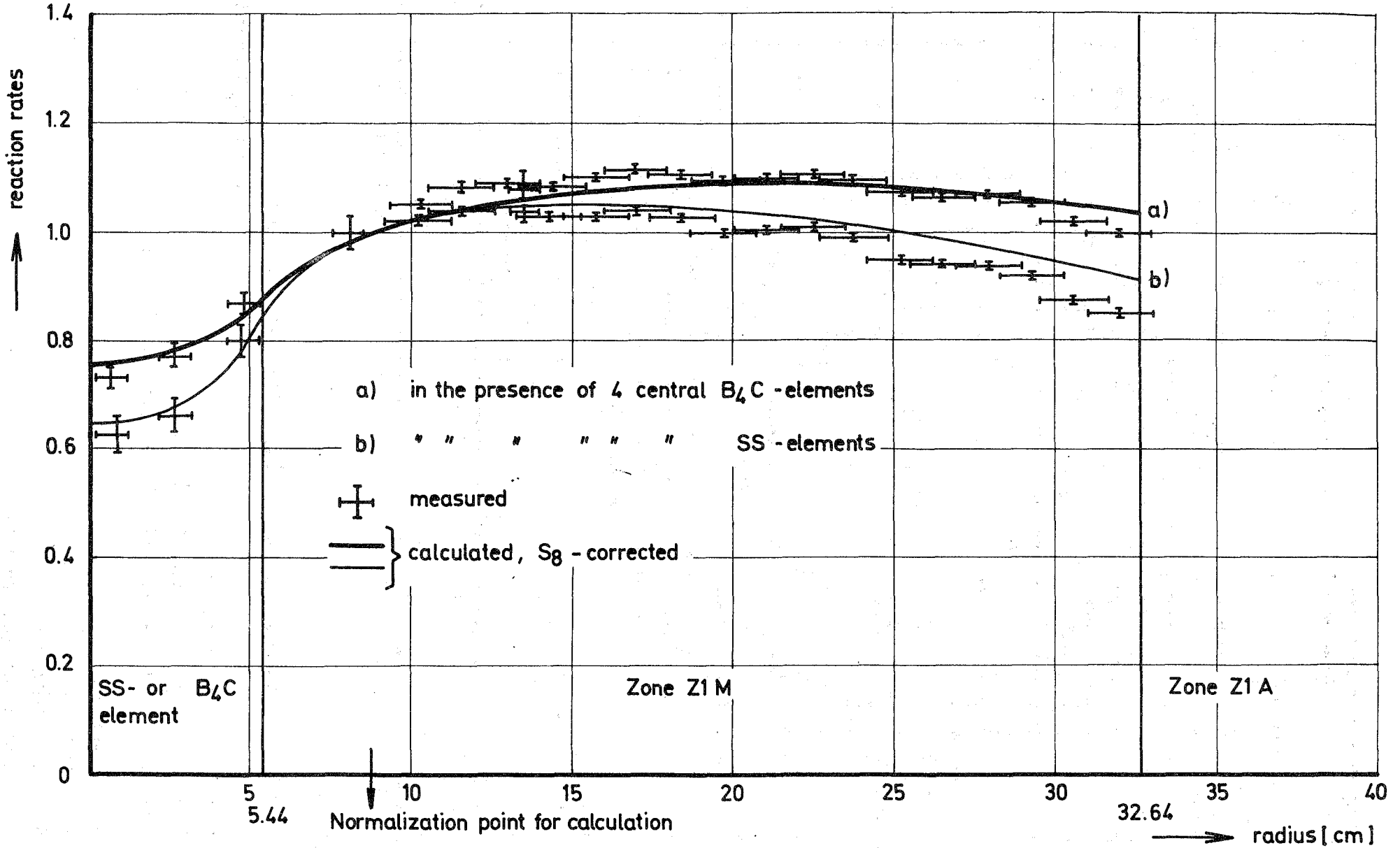



Fig. III - 2 SNEAK-6D Radial Fission Rate Traverse for U 238 normalized to one Atom

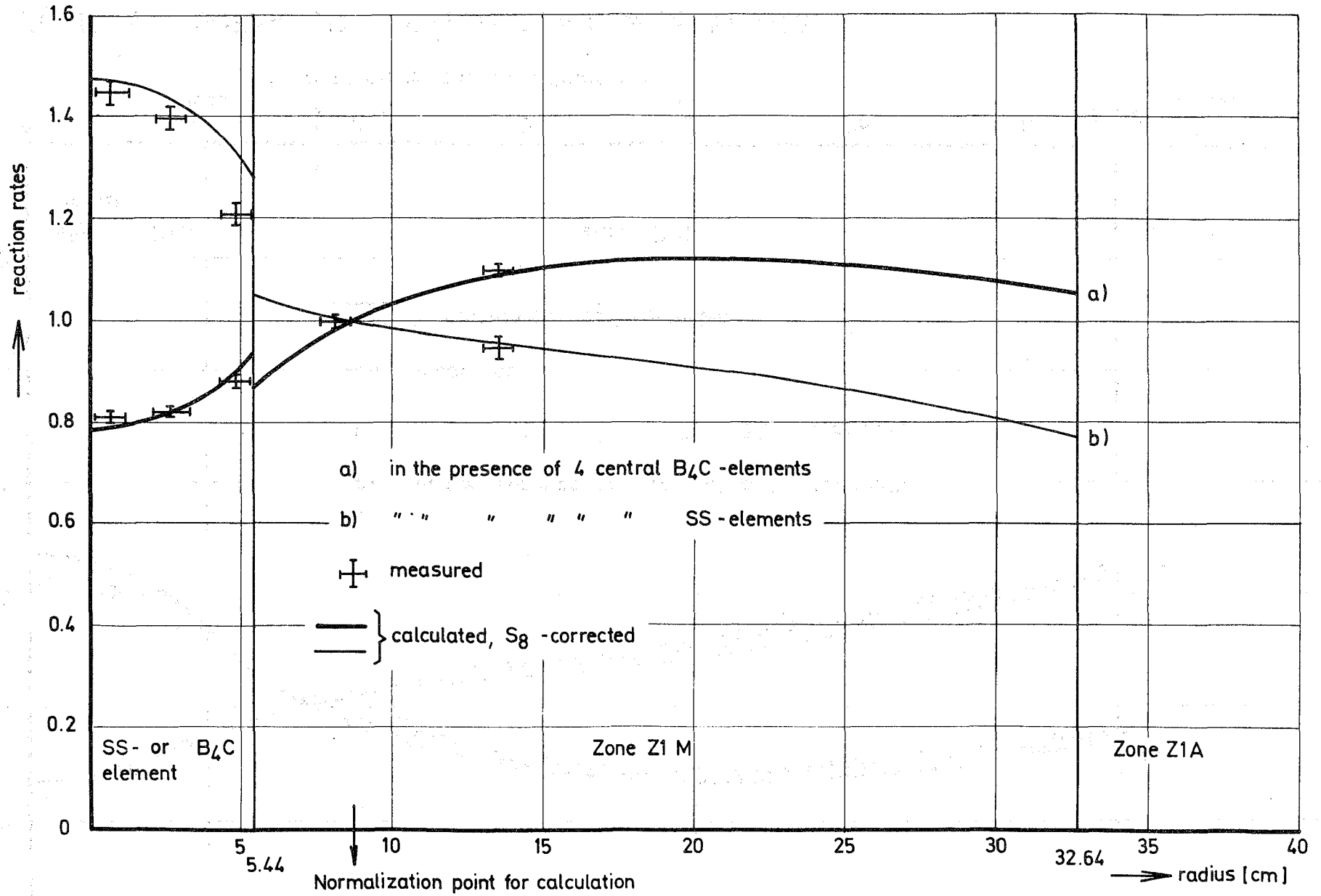



Fig. III-3 SNEAK-6D Radial Capture Rate Traverse for U238 normalized to one Atom

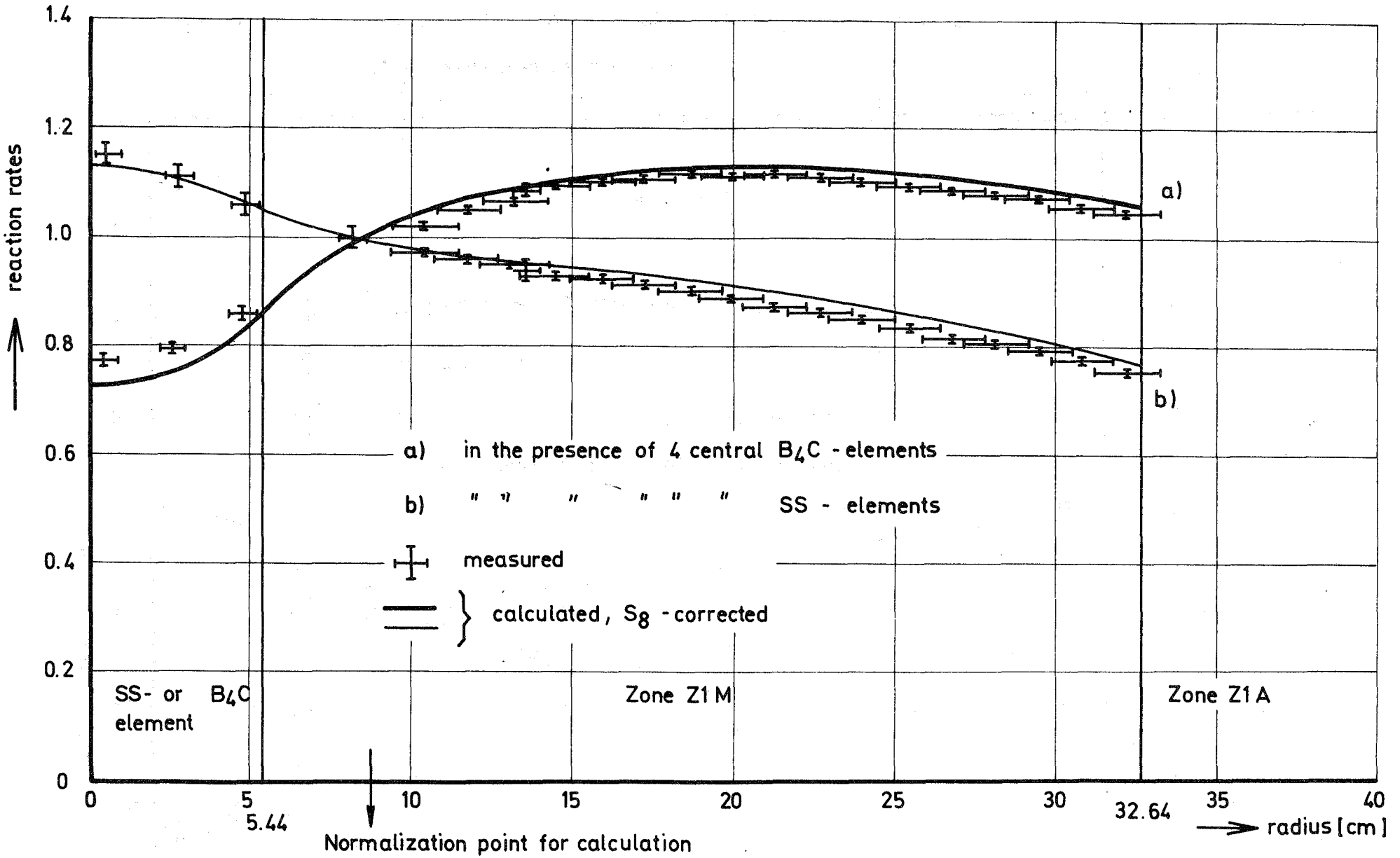



Fig. III - 4 SNEAK - 6D Fission Rate Traverse for U 235 normalized to one Atom

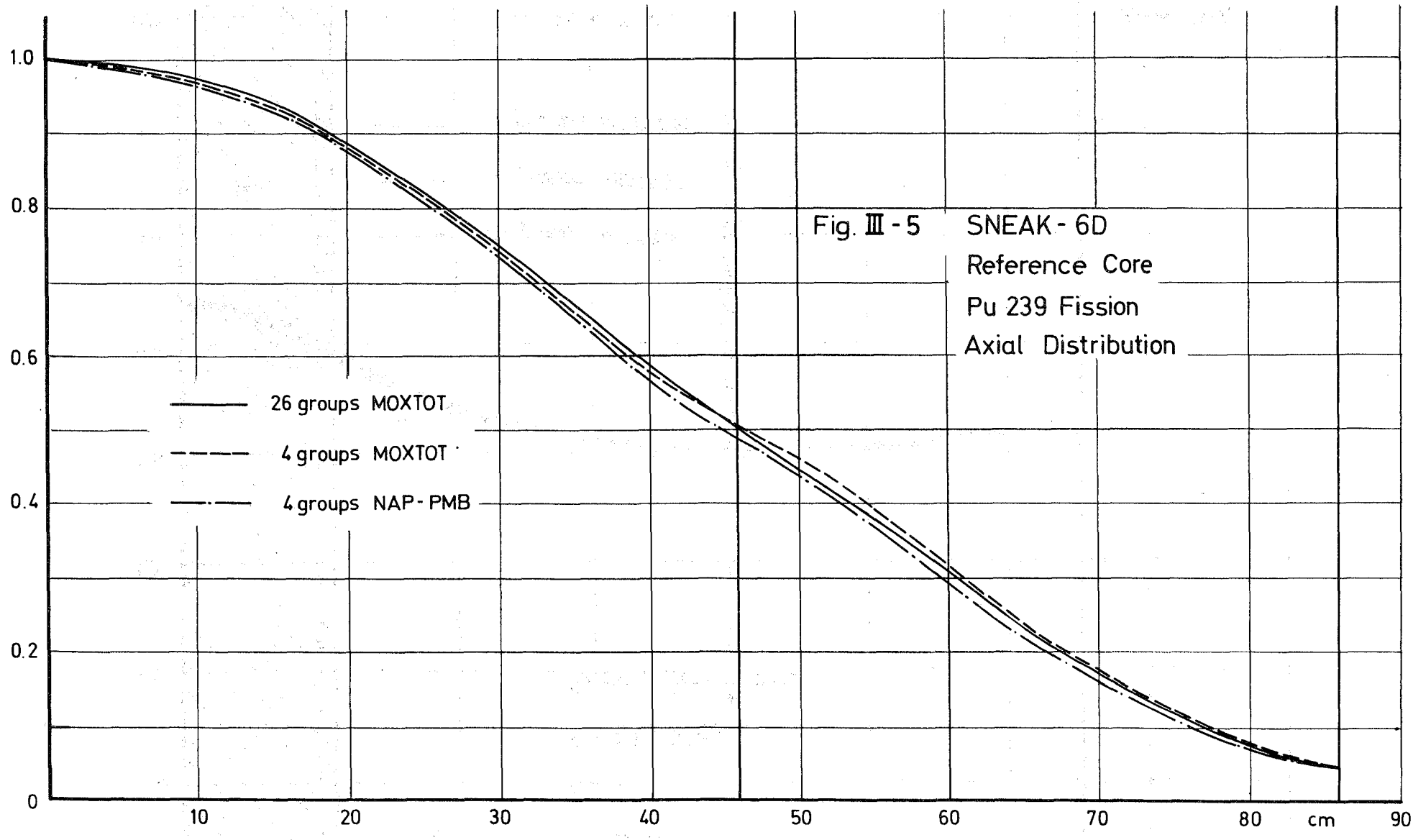



Fig. III-5 SNEAK-6D
Reference Core
Pu 239 Fission
Axial Distribution

Fig. III - 6 SNEAK - 6D