KERNFORSCHUNGSZENTRUM

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Institut für Angewandte Systemanalyse

Argentine-German Cooperation on Fuel Cycle Optimization
For the Nuclear Power Plant Atucha
Burn-up Calculations for the Atucha Nuclear Power Reactor

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Burn-up Calculations for the Atucha Nuclear Power Reactor

by

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P r e f a c e

A basic agreement on cooperation in scientific research and technological development was signed between the governments of the Federal Republic of Germany and of the Republic of Argentina on March 31 st, 1969.

Within the frame work of this basic agreement a special cooperation agreement for the peaceful uses of nuclear energy between the Comision Nacional de Energia Atomica (CNEA) and the Gesellschaft für Kernforschung (GfK) came into effect on July 29 th, 1971.

Under these cooperation agreements it was decided in 1974 to study, among other items, the possibilities to increase the discharge burn-up values of the fuel elements of the nuclear power plant Atucha. Work related to this project which was performed in 1974 is contained in a progress report KFK-2133 issued in July 1975.

Due to technical reasons, documentation of the burn-up calculations performed by Mr. Pieroni in 1974 was delayed and therefore could not be included in KFK-2133. Results of this work are now presented in this report.

Hans-Jürgen Zech

January 1976

Abbrandrechnungen für das Kernkraftwerk Atucha.

Z u s a m m e n f a s s u n g

Die physikalischen Eigenschaften des Kerns während des Abbrandes des Schwerwasser-Kernkraftwerkes Atucha werden untersucht. Das Zellrechnungsprogramm HAMMER und das Abbrandprogramm CITATION werden angewendet.
Zwei Fälle werden studiert: 1) Ein Naturuurankern, der im Gleichgewicht einen Entladeabbrand von ca. 7.0 MWD/kg erreicht; 2) Ein 1% homogen angereicherter Urankern, der im Gleichgewicht einen Entladeabbrand von ca. 15.0 MWD/kg erreicht. Die Ergebnisse zeigen die Anwendbarkeit des Programmsystems für die Studie zur Optimierung des Brennstoffzyklus des Atucha-Reaktors.

A b s t r a c t

The physical characteristics during the burnup of the core of the Atucha heavy water power reactor are investigated. The lattice program HAMMER and the depletion program CITATION are applied.
Two cases are studied: 1) a natural uranium core, which provides a discharge burnup of approximately 7.0 MWD/kg; and 2) a 1% homogeneously enriched uranium core, which provides a discharge burnup of approximately 15.0 MWD/kg. The results support the use of the program system for the study of the fuel cycle optimization of the Atucha reactor.

I. INTRODUCTION

In order to increase the burnup of the natural uranium heavy-water Atucha power reactor several possibilities are envisaged: homogeneous and heterogeneous uranium enrichment, homogeneous and heterogeneous plutonium recycling, and possibly others.

The possibilities mentioned require to be investigated as regards their technical feasibility as well as their economic aspects. The advantage of adopting one or the other possibility can be determined in accordance with the requirements of the nuclear power program.

A reliable calculation procedure is needed so as to perform an investigation of the physical characteristics for the different possibilities. A generalized fuel cycle program system, available at Karlsruhe, has been applied to the case of the Atucha reactor. This system includes the HAMMER program /1/ for the lattice calculations, and the CITATION program/2/ for the composite reactor calculations. The mass balances obtained from CITATION can be used in the CINCAS-II program /3/ to calculate the fuel contribution to the total electrical generating costs.

In the present study the program system has been applied to the case of the Atucha reactor with natural uranium core -which is the reference case- and to the case of a 1 % homogeneously enriched uranium core.

II. LATTICE CALCULATIONS

II.1 Atucha lattice cell

The heterogeneous Atucha reactor lattice is described by the multigroup transport theory lattice program HAMMER /1/. The multigroup homogeneized cross sections from the lattice program are condensed to produce a set of four-group cross sections with upper energies: 10 MeV - 1.05 MeV - 9.12 KeV and 0.625 eV. The four-group microscopic cross sections are transferred to an input library for the composite reactor calculation.

A cross-sectional view of the Atucha fuel bundle is shown in Fig. 1. Details of compositions and dimensions of the Atucha reactor, provided by the Kernkraftwerk Union (KWU) /4/ are given in the appendix. In Table I the lattice geometry and compositions used in the present calculations are given.

II.2 Cell calculation results

The results obtained with the HAMMER program for the natural uranium cell are shown in Table II. For comparison, the corresponding values obtained by KWU with their own program CIRTHE are also given /4/. It can be seen that good agreement between both calculations is obtained.

The concentrations corresponding to the different degrees of burnup are applied in the HAMMER program to obtain the variation of the infinite multiplication factor as a function of burnup. It can be seen in Fig. 2 that for the natural uranium case, the infinite multiplication factor is smaller than unity for burnup values greater than 4.2 MwD/Kg. This value can be considered as the maximum burnup obtainable from the first fuel elements to be discharged from the initial core.

The parameters corresponding to the 1 % enriched uranium cell are shown in Table III, and in Fig. 3 the variation of the infinite multiplication factor as a function of burnup is given. In this case this factor is smaller than unity for burnup values greater than 7.1 MwD/Kg.

III. REACTOR BURNUP CALCULATIONS

III.1 Atucha geometric mesh

The finite-difference diffusion theory program CITATION /2/ is applied to solve the depletion problem with elaborate refueling treatment for multi-cycle analysis.

A cross-sectional view of the Atucha reactor geometric mesh is shown in Fig. 4. The calculations are performed for a three-dimensional hexagonal geometry. Individual fuel elements are represented by corresponding zones. This detailed representation of the reactor geometry requires correspondingly long computation times. A typical time step in the multicycle analysis requires approximately 2.4 min CPU time.

In order to investigate the accuracy obtainable with a two-dimensional hexagonal geometry representation, a paralell calculation has been intended. But the version of the CITATION program available at Karlsruhe, called:

CITATION ISSUE DATE 07/01/71

did not work correctly for this geometry /5/. The latest version of CITATION, called: CITATION - REVISION 2 (JULY 1971) - SUPPLEMENT 2 (MARCH 1972) is presently being incorporated and tested, and it is expected that the difficulties mentioned will be overcome.

III.2 Influence of isotope concentrations

Figs. 2 and 3 give the effect on the infinite multiplication factor due to the different isotope concentrations. The corresponding cross sections are also affected by the variations of the isotope concentrations. For the present study it is relevant to know to what extent the reactor burnup calculations are influenced by the variations in the microscopic input data.

To investigate this effect, a series of reactor burnup calculations are performed with different microscopic cross-section data. For the natural uranium case, the cross-section data are obtained from calculations with the HAMMER program of cells with isotope concentrations corresponding to burnup values ranging from 0 to 7.2 MwD/Kg. In all cases the maximum variation in the reactor burnup value was smaller than 2 %. Therefore for the natural uranium case the microscopic data corresponding to the fresh cell has been adopted.

For the enriched uranium case a similar investigation is made in the burnup range from 0 to 18.0 MwD/Kg. The microscopic data set corresponding to a cell with a burnup of 9.6 MwD/Kg has been selected. With this data set the maximum variation in the reactor burnup value is smaller than 5 %.

III.3 Results for the natural uranium case

Fig. 5 shows the variation of the effective multiplication factor as a function of full power days (fpd) of operation after starting with a fresh core without fuel management manipulations.

The variation of the concentrations of the plutonium isotopes and of the ^{235}U isotope in a fuel element as a function of burnup is shown in Fig. 6.

A fuel management program is developed in order to obtain the equilibrium core condition. Due to the detailed reactor mesh adopted, the burnup of every individual fuel element can be determined and the corresponding thermodynamics requirements (see appendix) can be verified. Accordingly, a convenient fuel management manipulation can be found to optimize the obtainable burnup. A fuel management scheme results with fresh fuel elements being introduced in an intermediate region in the core. After reaching a burnup of approximately 2.0 MwD/Kg the fuel elements are moved to the central region of the core. There they obtain a burnup of approximately 6.0 MwD/Kg before being moved to the outer region of the core, from where they are later discharged. With this -shortly described- fuel management scheme, which requires replacement or shifting of approximately one fuel element every full power day, the equilibrium core condition is obtained after approximately 950 fpd.

Fig. 7 shows the power shape factor for every fuel element after 950 fpd. In Fig. 8 the corresponding burnup distribution, normalized to the central zone, is shown. The discharge burnup is approximately 7.0 MwD/Kg, and the average burnup of the core is approximately 4.0 MwD/Kg. These values agree with the corresponding ones obtained by KWU.

III.4 Results for the 1 % homogeneously enriched uranium case

Fig. 9 shows the variation of the effective multiplication factor as a function of time after starting with a fresh core without fuel management operations. The concentrations of the plutonium isotopes and of the ^{235}U isotope in the 1 % enriched uranium element as a function of burnup are shown in Fig. 10.

A similar fuel management scheme as in the natural uranium case is applied to obtain the equilibrium core condition. In the enriched uranium case the fuel elements must have a burnup of approximately 6.0 MwD/Kg before they are moved to the central region of the core. One fuel element must be replaced or shifted every 2.2 days, approximately. The equilibrium core condition is obtained after approximately 600 fpd of operation.

Fig. 11 shows the power shape factor and Fig. 12 the burnup distribution in the core after 1000 fpd. In the equilibrium core condition the discharge burnup is approximately 15.0 MwD/Kg, and the average core burnup is approximately 9.0 MwD/Kg.

IV. CONCLUSIONS

The results of the physical calculations performed with the generalized fuel cycle program system available at Karlsruhe agree with the known values corresponding to the Atucha reactor with a natural uranium core.

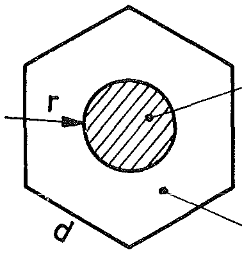
The results for the 1 % homogeneously enriched uranium core also agree in the range expected according to previous approachs.

This confirms the reliability of the program system for the burnup analysis of a heavy water reactor.

In particular, the present study supports the confidence in the use of this program system to investigate other possibilities of improving the fuel cycle of the Atucha reactor.

Table I

Atucha lattice cell



Isotope		Average concentration (at/b.cm) Natural uranium case	1 % enriched uranium case
Region 1 T=770°C	^{238}U	7.368 E-03	7.347 E-03
	^{235}U	5.284 E-05	7.421 E-05
	O	3.200 E-02	3.200 E-02
	"Zry"	6.263 E-03	6.263 E-03
	D	3.424 E-02	3.424 E-02
	H	8.582 E-05	8.582 E-05
Region 2 T=227°C	D	6.613 E-02	6.613 E-02
	H	1.658 E-04	1.658 E-04
	O	3.315 E-02	3.315 E-02

Table II

NATURAL URANIUM case

Cell Parameters / Clean Hot Critical

PARAMETER	CIRTHE	HAMMER
ETA_{TH}	1.2495	1.2573
F	0.9563	0.9855
$ETA_{TH} * F$	1.1922	1.2391
L^2	2.5826 E+02	2.4416 E+02
D_{TH}	1.0044	0.8928
Σa_{TH}	0.3889 E-02	0.3657 E-02
$\nu \Sigma_{fTH}$	0.4859 E-02	0.4531 E-02
P	0.8924	0.8456
AGE	1.3541 E+02	1.0436 E+02
K_∞	1.0919	1.0947
K_{EFF}	1.0457	1.0526
$\frac{ABS\ (^{238}U)}{PROD\ (^{235}U)}$	0.4050	0.4116
$\frac{ABS\ (^{235}U)}{PROD\ (^{235}U)}$	0.4886	0.4851
ε	1.0218	1.0448

Table III

1 % ENRICHED URANIUM case

Cell Parameters / Clean Hot Critical

ETA_{TH}	1.4180
F	0.9874
$ETA_{TH} * F$	1.4000
L^2	2.1269 E + 02
D_{TH}	0.8928
Σ_{aTH}	0.4196 E - 02
$\nu\ \Sigma_{fTH}$	0.5877 E - 02
P	0.8404
AGE	1.0417 E + 02
K_{∞}	1.2337
$\frac{ABS\ (^{238}U)}{PROD\ (^{235}U)}$	0.3224
$\frac{PROD\ (^{238}U)}{PROD\ (^{235}U)}$	0.01721
$\frac{ABS\ (^{235}U)}{PROD\ (^{235}U)}$	0.4862
ε	1.0571

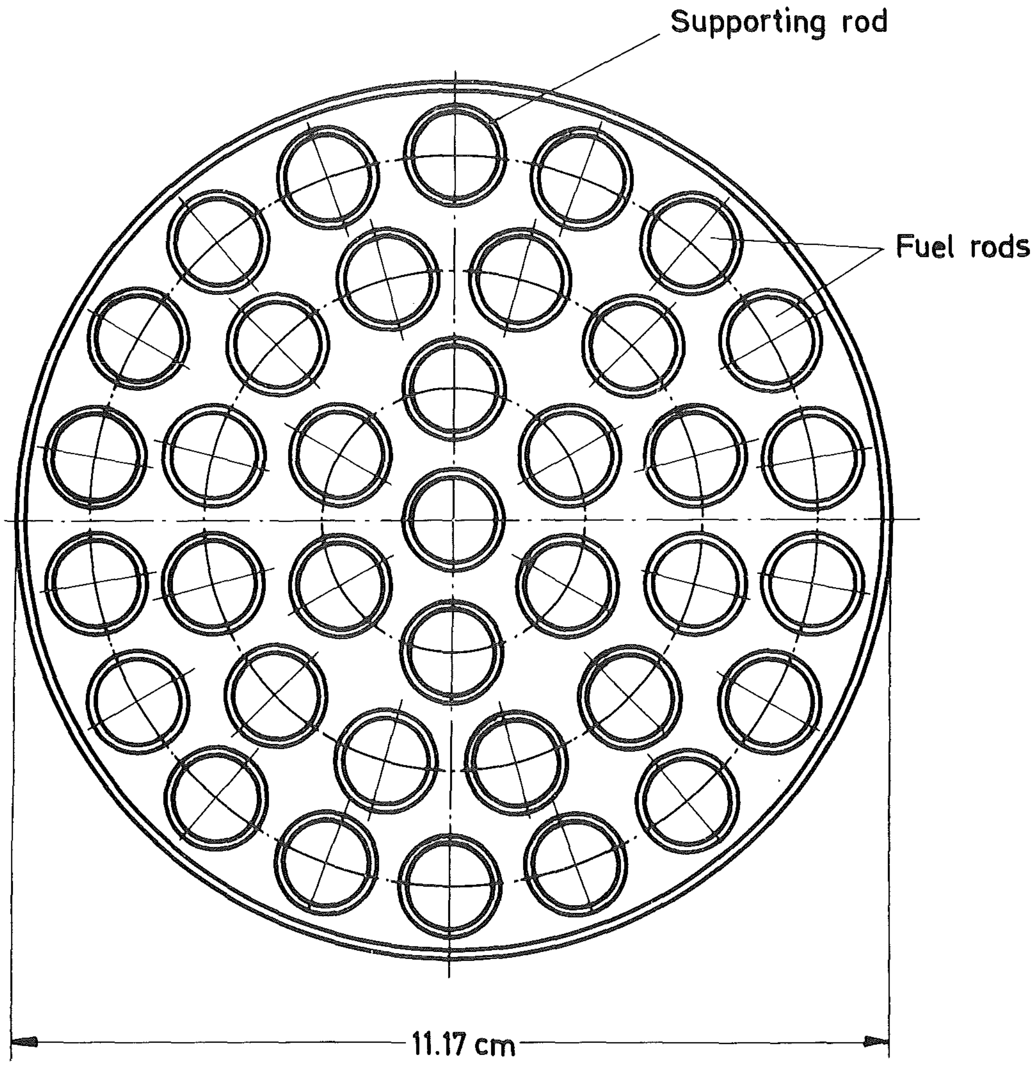


Fig. 1 Cross Sectional View of Fuel Rod Bundle

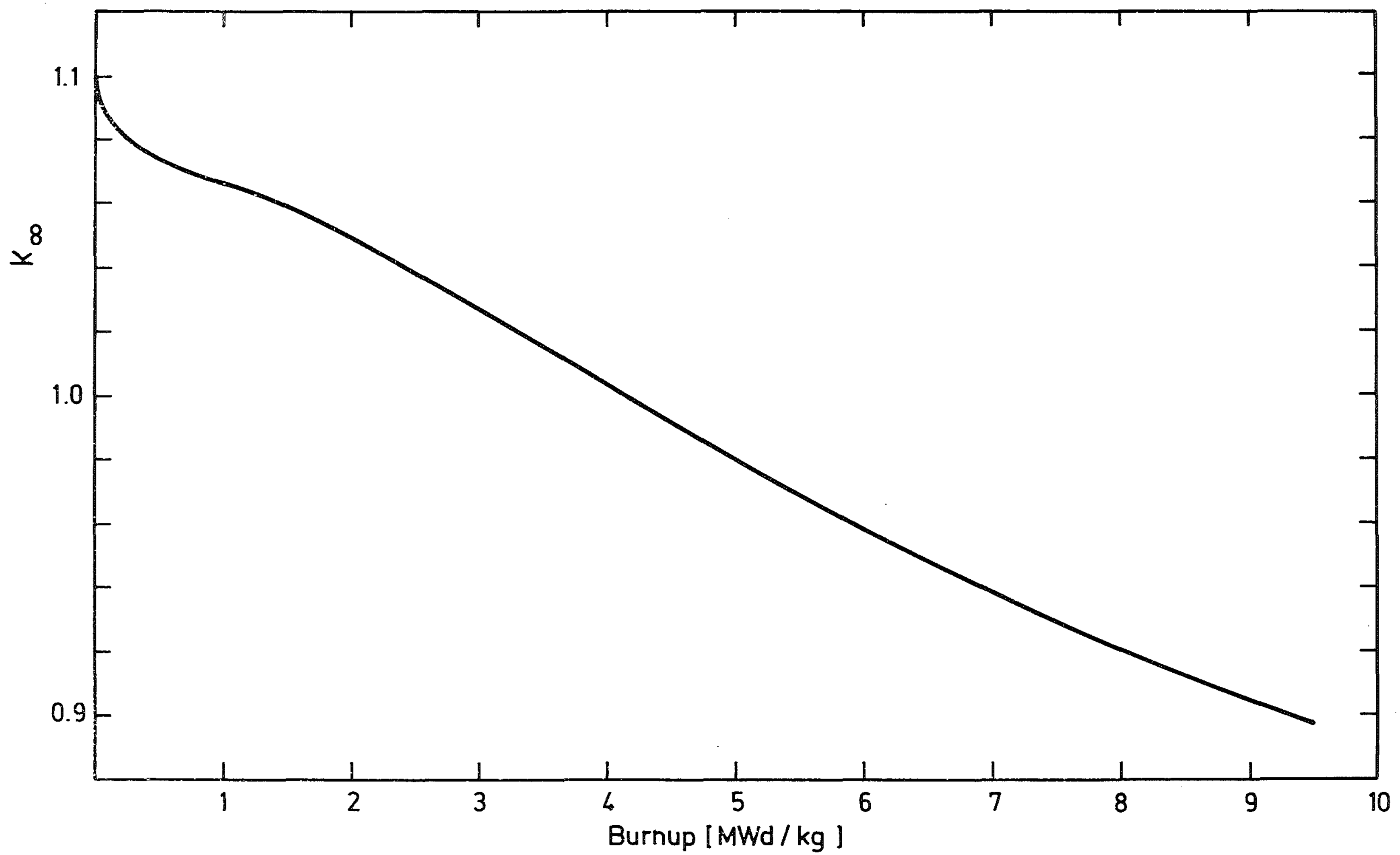


Fig. 2 ATUCHA Natural Uranium Cell : Infinite Multiplication Factor as a Function of Burnup

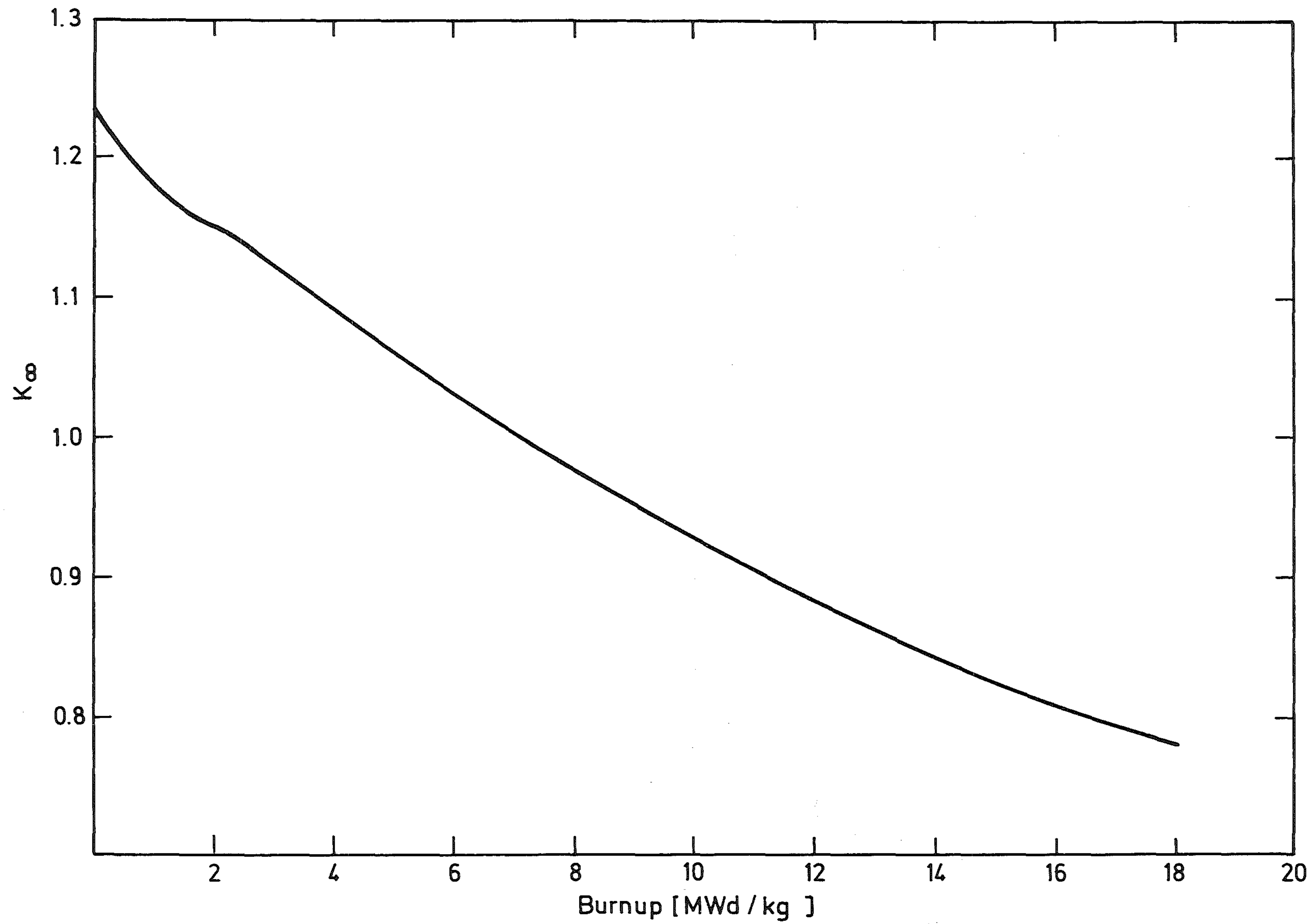
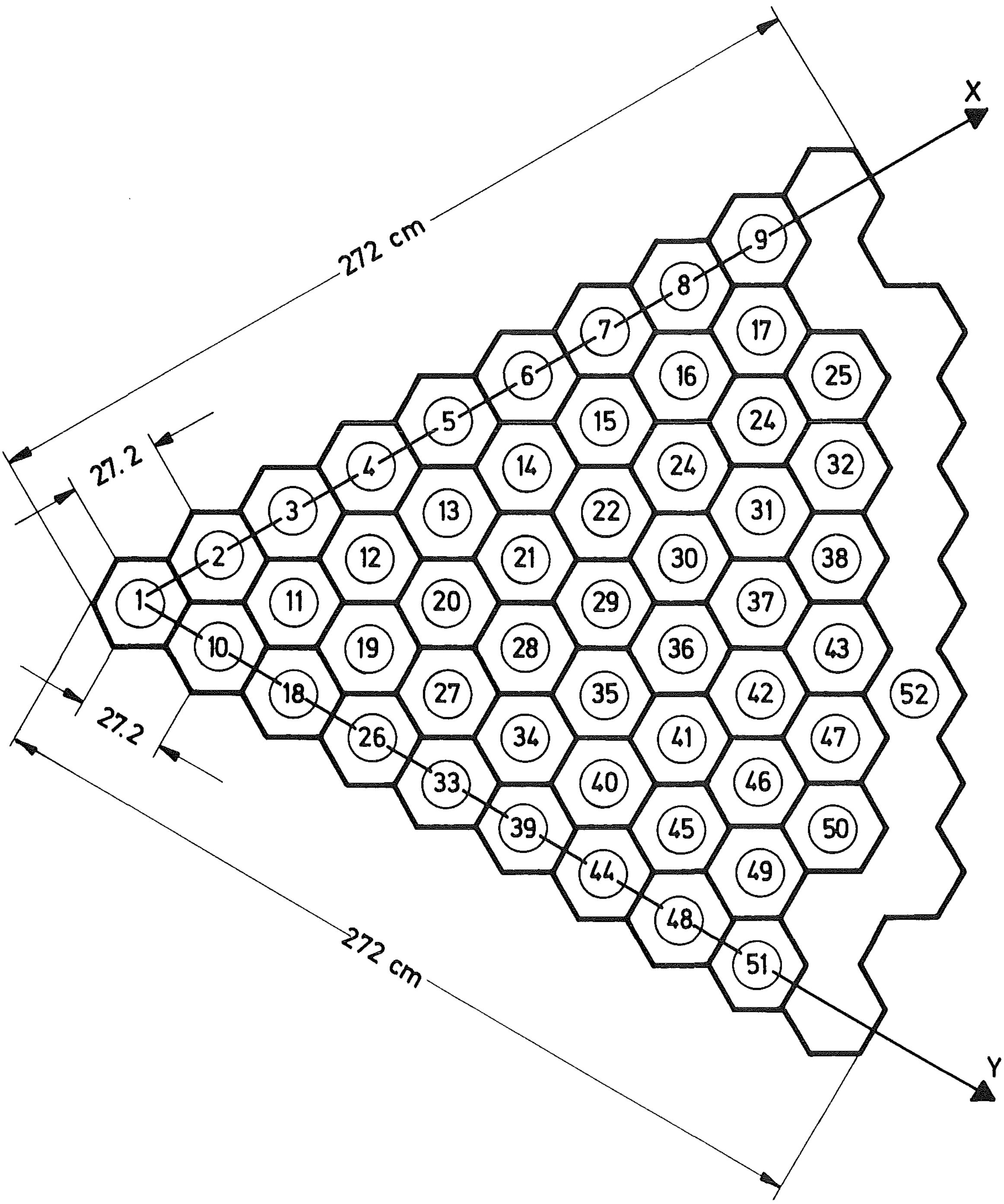


Fig. 3 ATUCHA Enriched Uranium Cell :
Infinite Multiplication Factor as a Function of Burnup



Zones 1 to 51 : Fuel

Zone 52 : Reflector

Fig. 4 Cross Sectional View of the Geometric Mesh for the ATUCHA Reactor Calculation

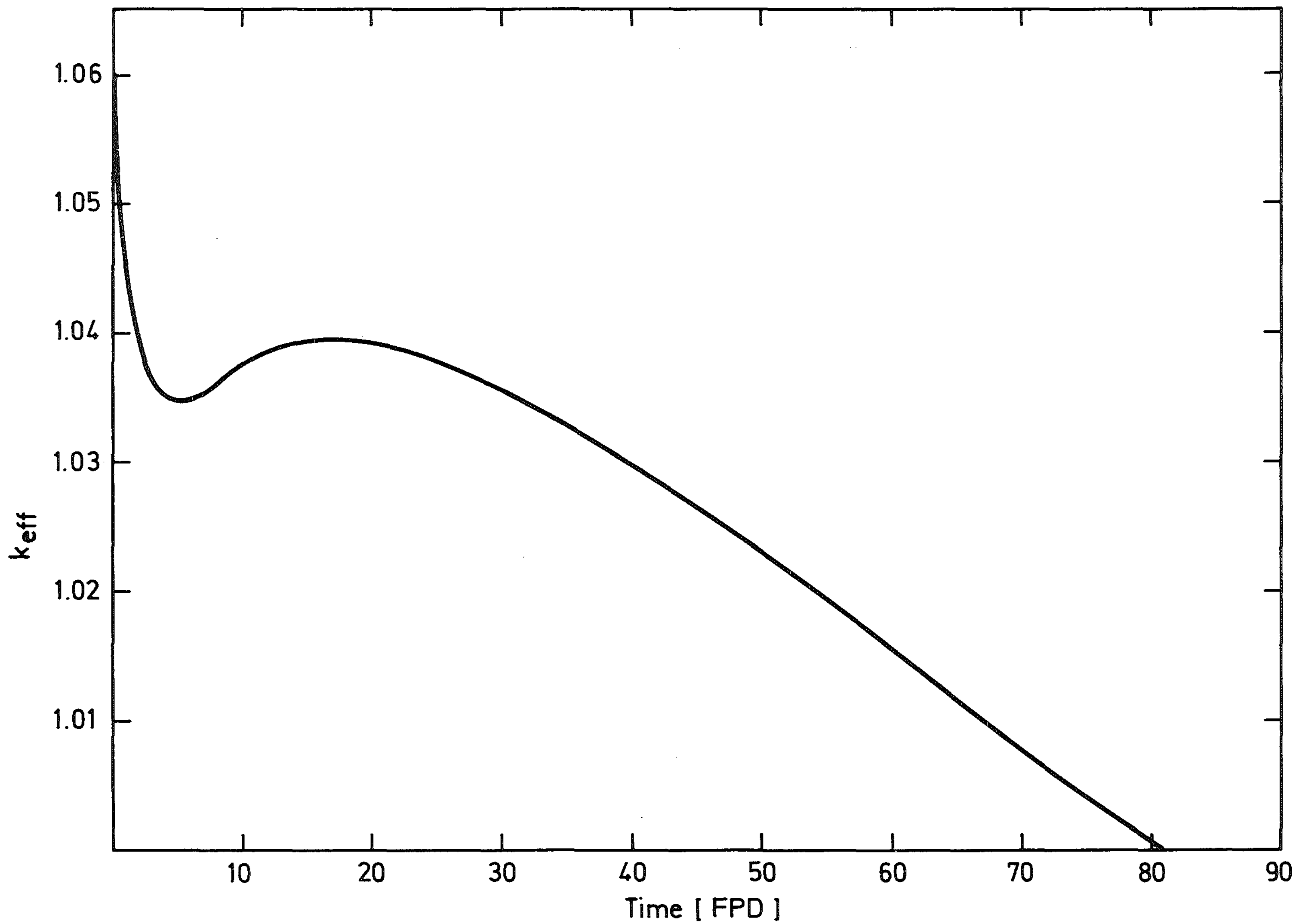


Fig. 5 Natural Uranium Case :
Effective Multiplication Factor as a Function of Time

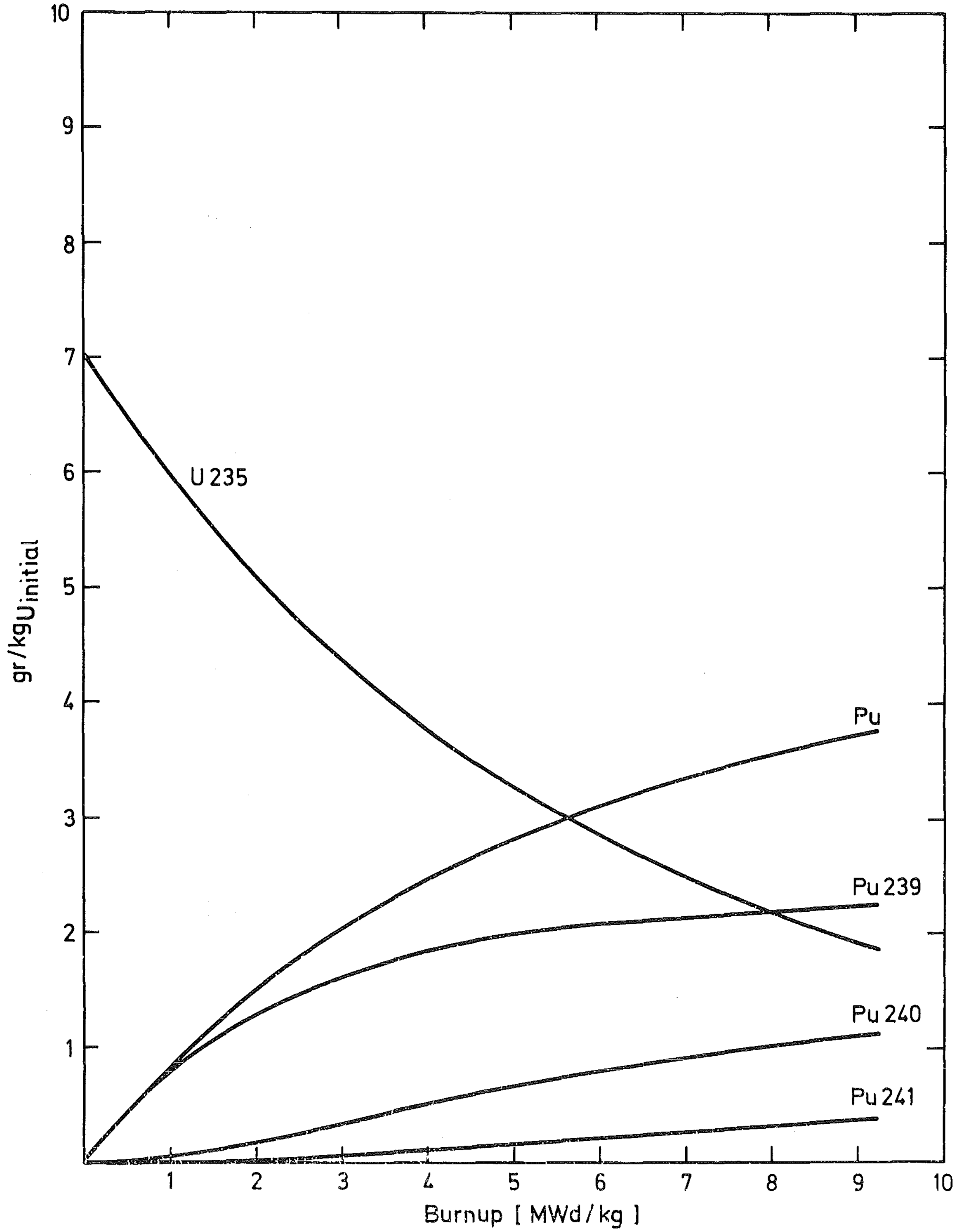


Fig. 6 ATUCHA Natural Uranium Fuel :
Isotope Concentration as a Function of Burnup

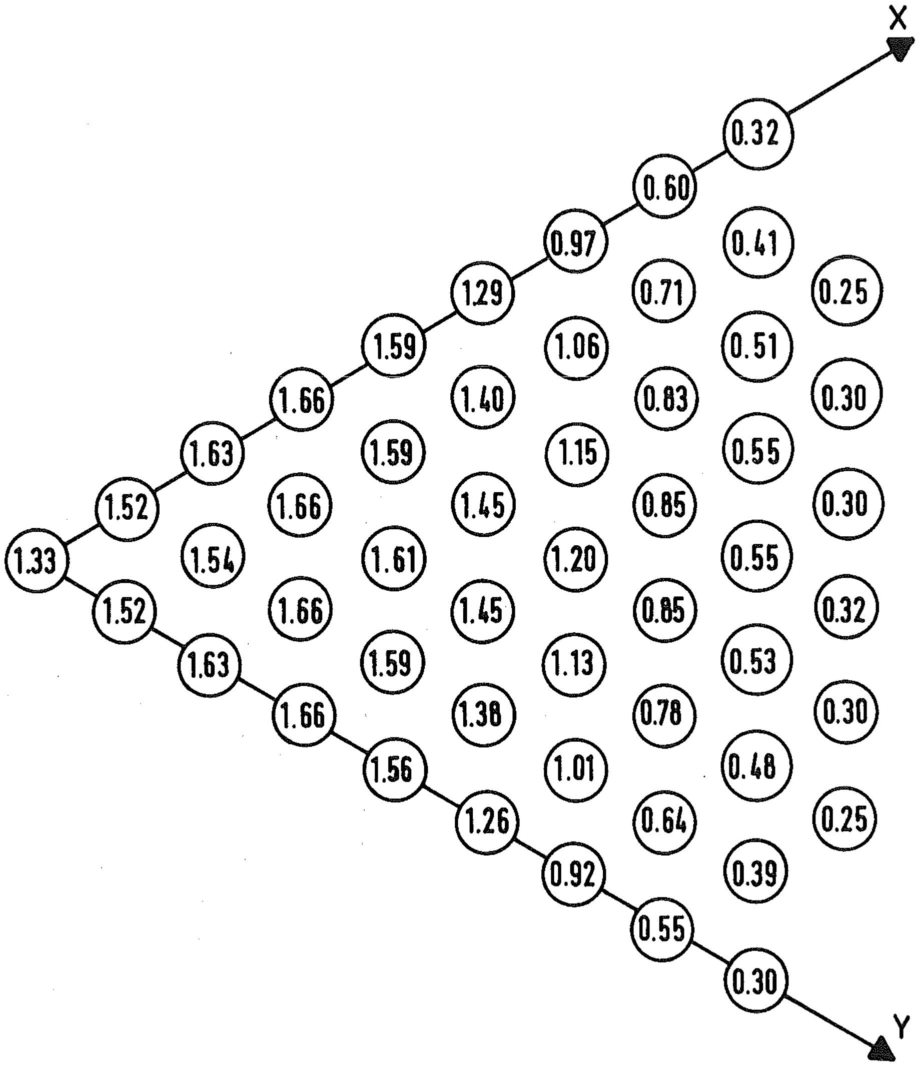


Fig. 7 Natural Uranium Case :
Power Shape Factor (Equilibrium Core)

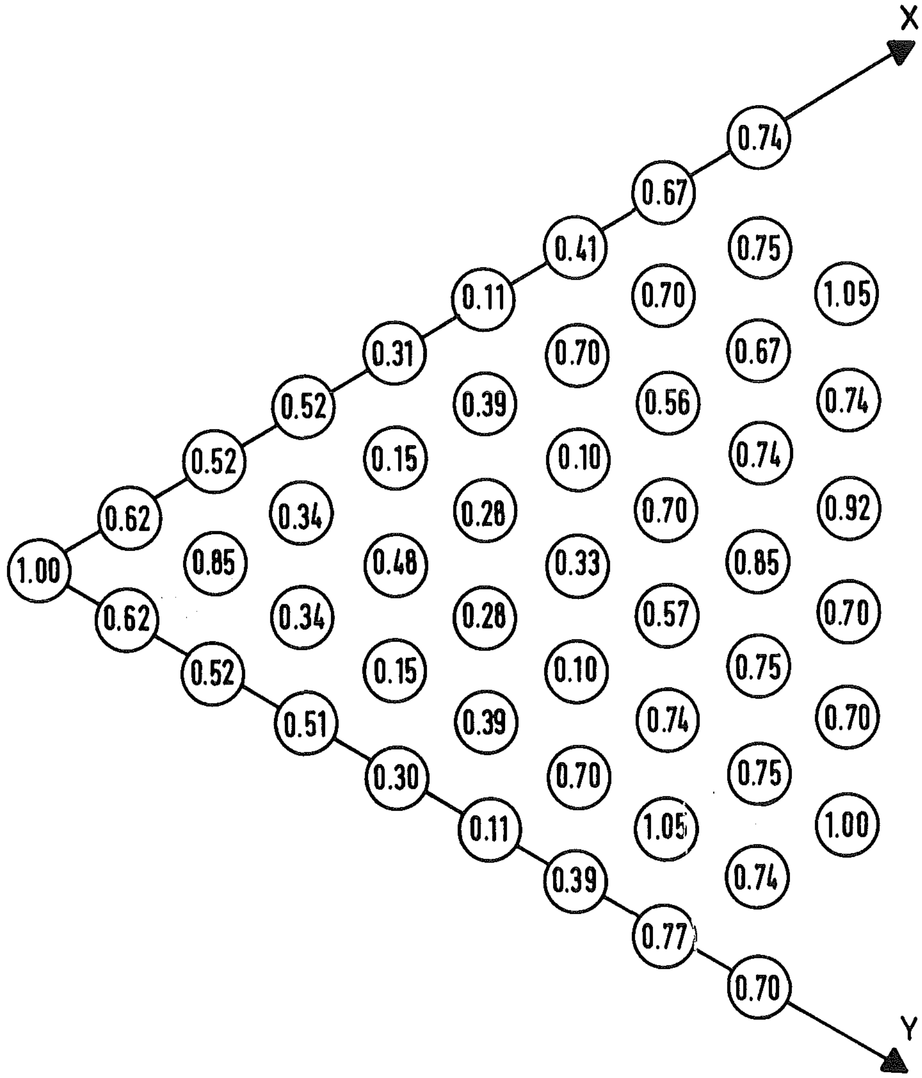


Fig. 8 Natural Uranium Case :
Burnup Distribution Normalized to the
Central Zone (Equilibrium Core)

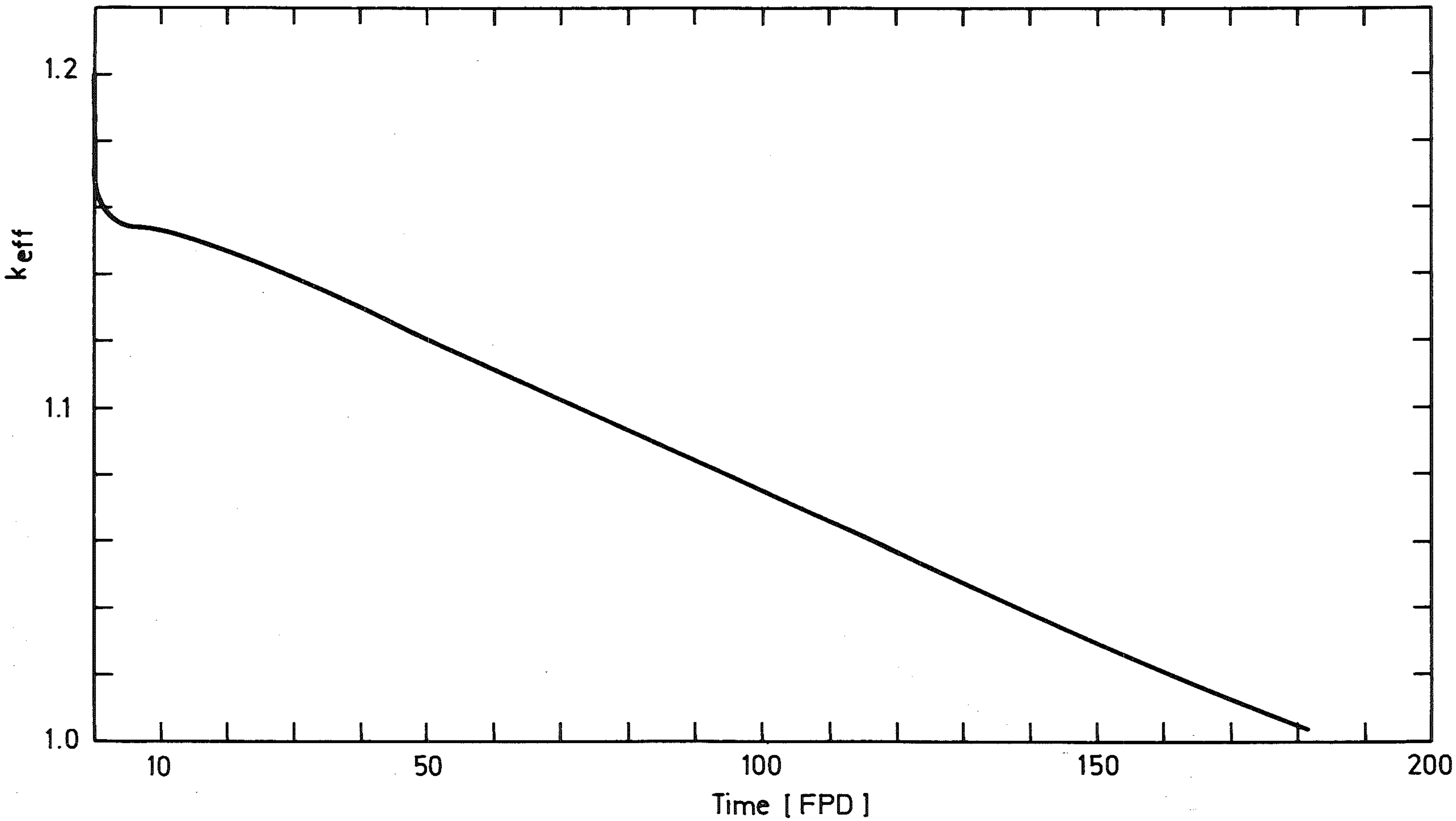


Fig. 9 1% Homogeneous Enriched Uranium Case :
Effective Multiplication Factor as a Function of Time

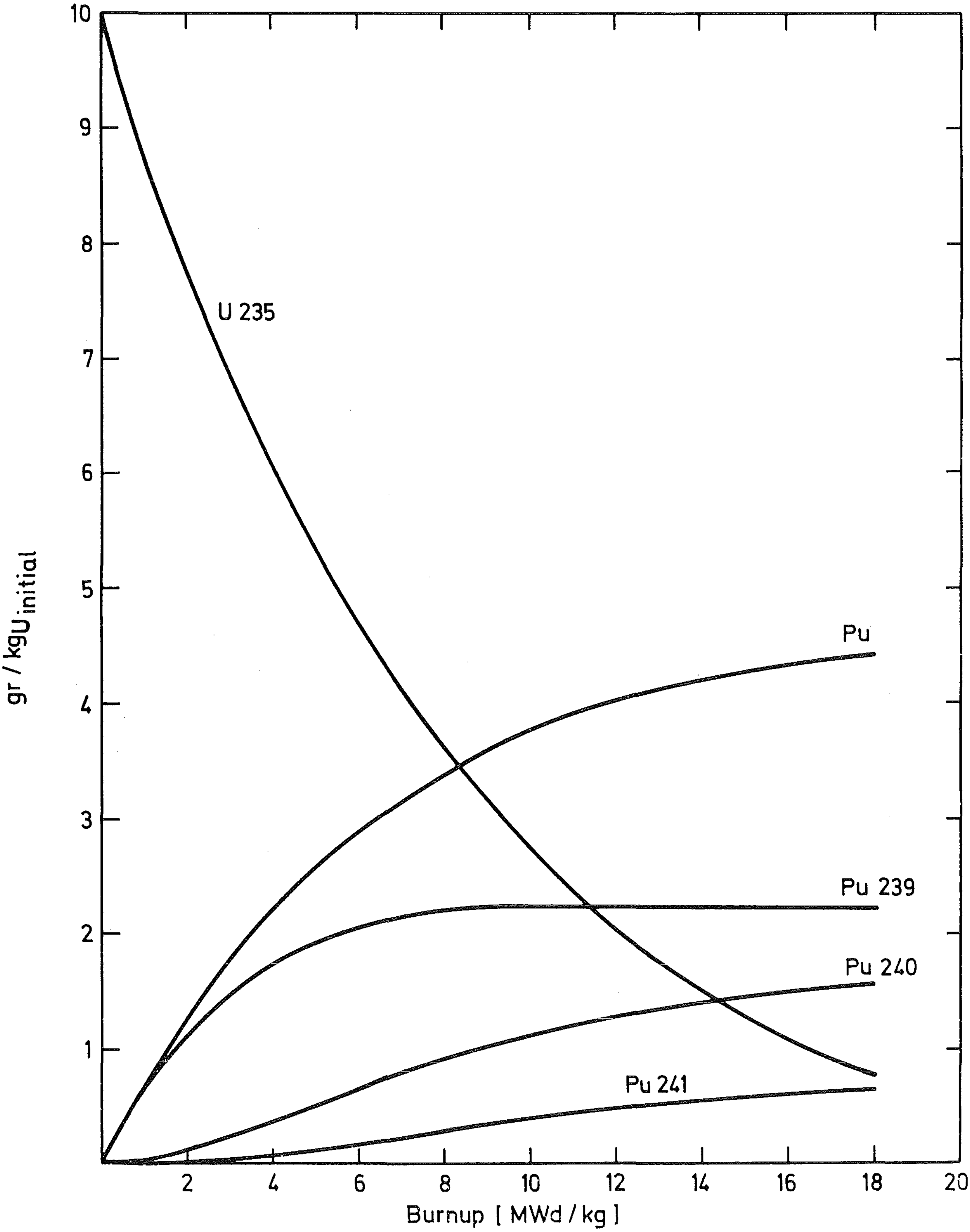


Fig. 10 ATUCHA 1% Enriched Uranium Fuel : Isotope Concentration as a Function of Burnup

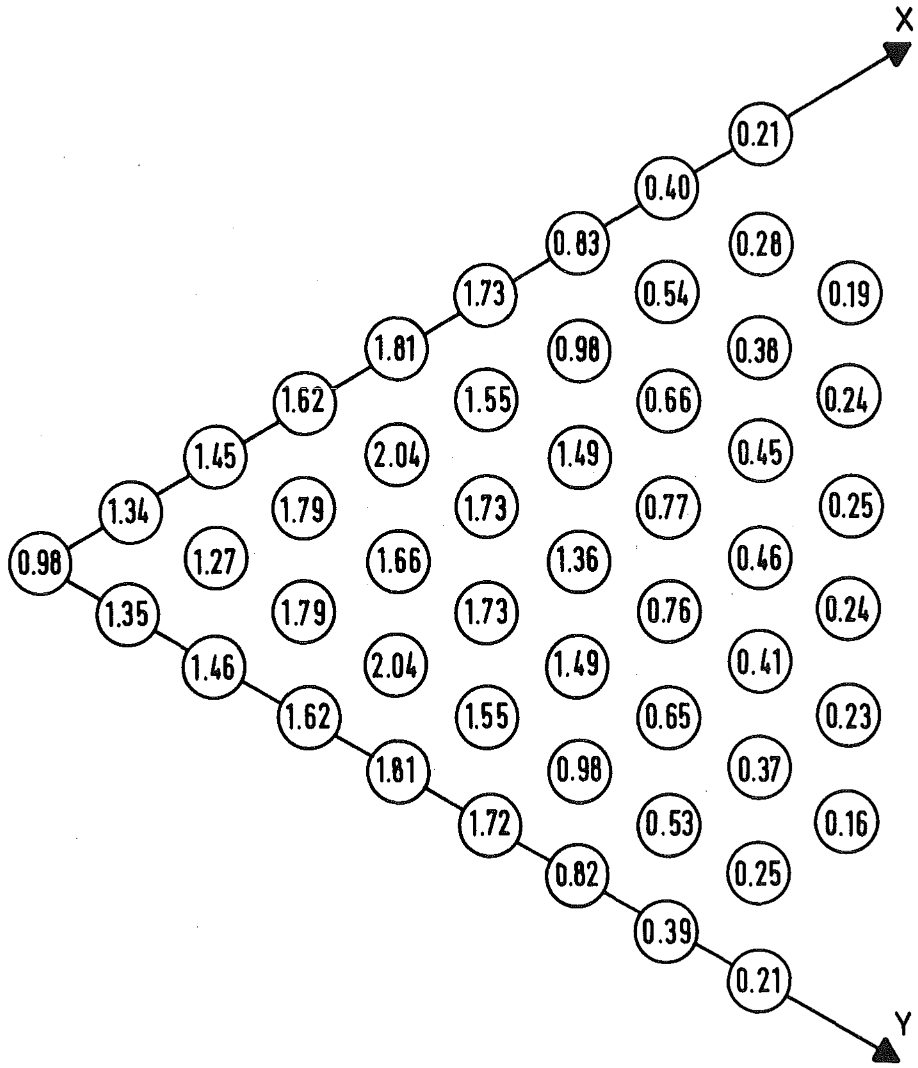


Fig. 11 1% Homogeneous Enriched Uranium Case : Power Shape Factor (Equilibrium Core)

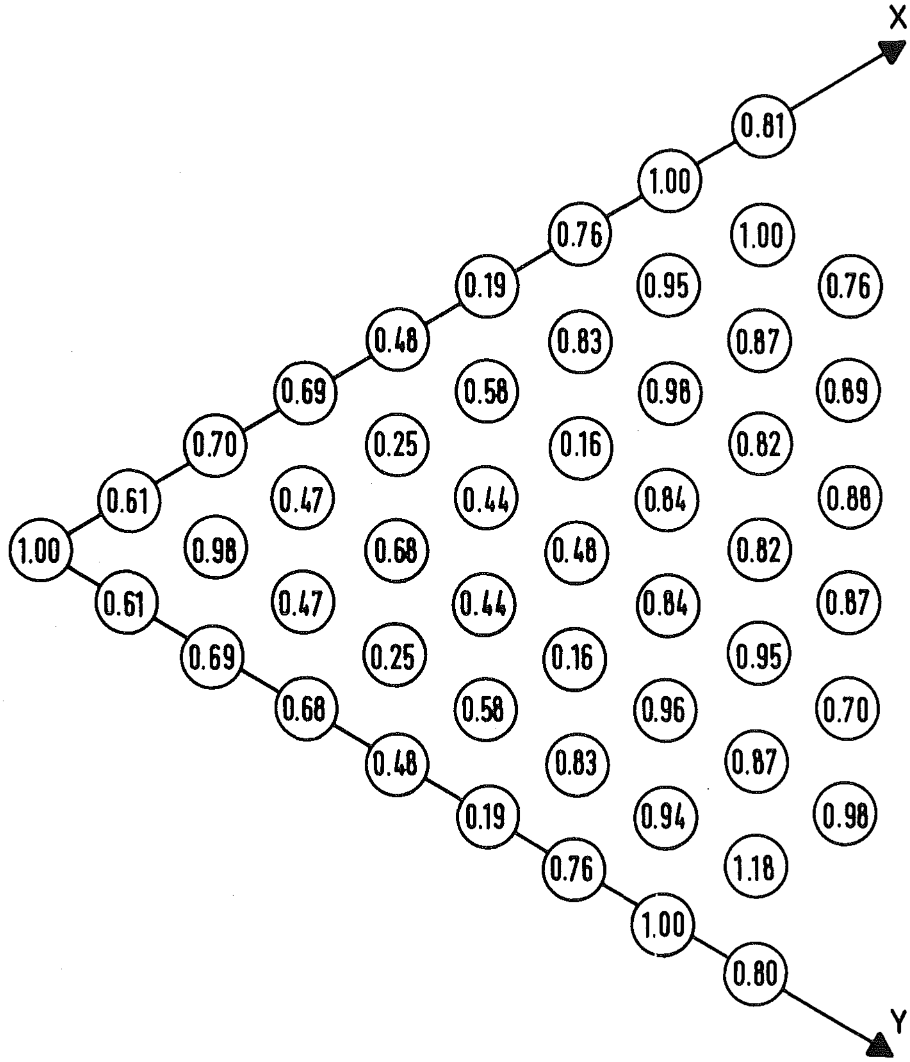


Fig. 12 1% Homogeneous Enriched Uranium Case : Burnup Distribution Normalized to the Central Zone (Equilibrium Core)

Appendix

Data list Atucha

FUEL Natural uranium dioxide

Effective density	9.9 g/cm^3
Pellet radius	0.54 cm
Mean temperature	770 ^{o}C

ARRANGEMENT OF FUEL RODS

1 central rod

6 rods in a circle, radius 1.60 cm

12 " " " " " 3.04 cm

17 " " " " " 4.52 cm

The outer circle also includes a supporting tube made of Zry-4 with an internal radius of 0.475 cm, wall thickness of 0.12 cm and 6.55 g/cm^3 density.

CLADDING TUBE Zry-4

Density	6.55 g/cm^3
Inner radius	0.54 cm
Effective wall thickness	0.06 cm

COOLANT D_2O with 0.25 % weight H_2O

Mean temperature 288 ^{o}C

CALANDRIA Zry-4

Density	6.55 g/cm^3
Inner radius	5.41 cm
Effective wall thickness	0.175 cm

MODERATOR D_2O with 0.25 % weight H_2O

Mean temperature 180 °C

Lattice pitch 27.2 cm (trigonal arrangement)

Effective core height 530 cm

Effective core radius 227.15 cm

Reflector thickness: axial upper part 45.5 cm

axial lower part 46.5 cm

radial 24.05 cm

Power limitations: maximum power per channel $Q \leq 7.2$ Mw

maximum linear power $q' \leq 500$ w/cm

mean linear power $\overline{q'} \leq 216.5$ w/cm

macroscopic form factor $F = F_{rad} * F_{axial} \leq 2.07$

References

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4. H. Raum, Kernkraftwerk Union AG, Erlangen, private communication.

5. H. Bogensberger and H.-J. Zech, Gesellschaft fuer Kernforschung m.b.H., Karlsruhe, private communication.

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