مجلهٔ علوم دانشگاه تهران جلد ۱۷ شماره‌های ۱ و ۲ سال ۱۳۶۷ ازصفحهٔ ۲۱ تا ۳۰

محاسبات نوترونی قلب راکتور تحقیقاتی سازمان انرژی اتمی ایران

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چکیده

دیر زمانی نیست که سوخت اورانیوم باغنای بالا (بیش از ۹۰ درصد) برای ادامه کار راکتورهای تحقیقاتی براحتی در اختیار قرار نمیگیرد. آژانس بین‌المللی انرژی اتمی دراین راه قدم های اولیه را برداشته و ازکشورهای مختلفی دعوت به‌همکاری برای محاسبات نوترونی و ترموهیدرولیکی، ساخت و آزمون سوخت های باغنای ۵ر۶، ۲۰ و ٤٥ درصد را نموده است. کشورهای دارنده اینگونه راکتورها خود نیز جهت دستیابی و آشنایی باروش های محاسباتی و اشکالات تعویض سوخت اقدام به اینگونه محاسبات نموده‌اند.

در رابطه با تعویض سوخت راکتور تحقیقاتی ۵ مگاواتی سازمان انرژی اتمی ایران، نگارنده خود محاسبات نوترونی و ترموهیدرولیکی با سوخت باغنای بالا و ۲۰ درصد را انجام‌داده‌است. از آنجاکه درتعویض سوخت از ۹۰ به ۲۰ درصد تغییراتی در توزیع فلو ودر نتیجه تغییراتی در توزیع قدرت تولیدی وراکتیویته وکمیت‌های ویژه بوجود میآید لازم است محاسبات نوترونی و حرارتی صورت‌گیرد تا بتوان آرایش قلب جدید را به‌نحوی ترتیب داد تا تغییرات ناشی از تعویض سوخت به‌حداقل ممکن برسد. گواینکه بعضی ازکشورها ضمن تغییر سوخت تغییراتی درآرایش قلب میدهند تا نیازهای تحقیقاتی خود را فراهم نمایند .

در این مقاله نتیجه محاسبات نوترونی قلب فعلی باغنای ۹۳ در صد انجام‌گرفته است و ازمقایسه این نتایج با نتایج سوخت ۲۰ درصد غنی، دیده می‌شود که تعویض سوخت از نقطه نظر نوترونی کاملا - امکان‌پذیر است .

این تغییرات نوترونی بیشتر ناشی از وجود ۸۰ درصد اورانیوم ۲۳۸ از نظر جذب نوترون دررزنانس ها و پدیده دوپلر و ابعاد قلب جدید است، که موجب تغییرات طیف نوترون‌گشته میزان واکنش های هسته‌ای را تغییر میدهد. محاسبات بیشتر نوترونی و حرارتی و سینتیک نیز در جریان است .

J. of Science, Univ. of Tehran (1988) **17**, 21-30.

Neutronic Calculations of NRC HEU Core

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Abstract

Research reactors as a source of neutrons and gamma rays have a great potential for doing original applied physics studies such as solid state, nuclear properties, neutron crystallography, neutron dosimetry, neutron radiography, neutron activation analysis - (NAA), metallurgy and reactor physics.

The HEU fuel of nearly all research reactors have to be replaced by lower enriched uranium fuel, namely 20%, in near future. In the conversion of HEU to LEU fuel

analytical and experimental studies should be made to assist decision makers on fuel loading and on changes necessary in the core configuration to be implemented to meet present and future researcher's demands .

In this study, neutronic calculations of the present HEU fueled core was performed. Reactivity, group flux distribution, burnup pattern and power distribution in the core were obtained. In the continuation of the calculations neutronic, thermal hydraulic and kinetics of the HEU and LEU fueled cores will be investigated in order to achieve a consistent results for fuel replacement .

1. Introduction

Research reactors from the day of their birth have always been valuable source of neutrons and gamma rays of fairly wide range of energy to implement research projects and practical studies in many fields of scientific and technological interests. It is worthwhile to pay more attention to its development and capabilities for future planning of scientific research and technolo - gical developments in Iran .

Research reactors are now passing a new era of core design and capabilities. Compact cores with high density of uranium fuel (Bonig and Van Der Hardt, 1988) with high neutron flux and equipped with cold neutron facilities give wider range of applicabilities for neutron studies. In many countries have already ini tiated new core design.

AEOI is in the process of changing HEU fuel of its reactor core to LEU fuel of 20% enriched. The present calculation is only a part of neutronic analysis of the HEU core for fuel conversion to LEU fuel .

2. Core specifications

The core configuration is shown in Fig - 1. The core consists of 20 standard fuel elements (SFE) (AMF, 1959) 2 partial fuel elements and 6 control fuel elements. In this study in place of partial fuel elements two standard fuel elements were considered. Each SFE is composed of 18 curved plates which 16 of them are fueled with 209. 37 g of uranium of 93% enriched in U - 235.

3. Method of calculation

3. 1 Cell Calculation

To perform cell calculations a unit cell shown in Fig 2 (a,b)was considered. The extra zone accounts for extraAluminium and water in the SFE. In the cell calcula tion the modular code System RSYST (Ruhle, 1975, Ruhle, 1983) was used. By calculation neutron spectrum in zero burnup and no Xe - 135 buildup, five group constants were generated from 45 energy group libraty see Table 1 for group structure. As fuel irradiation

Table 1 Five and Three Energy Group Structure used in RSYST

Energy group	Energy Ranges	
1	15 Mev	0. 111 Mev
2	111. 1 Kev	1. 855 Kev
3	1. 855 ev	0. 625 ev
4	0. 625 ev	0. 11157 ev
5	0. 11157 ev	0. 000001 ev
	A	B
1	1 + 2	1
2	2	2 + 3
3	4 + 5	4 + 5

proceeds fuel burns and as a result neutron energy varies, as such for further calculation new cross section data becomes necessary. Therefore, a new set of cross sections in five groups for different materials were det - ermined. In this sequence of calculations burnup, Kinf

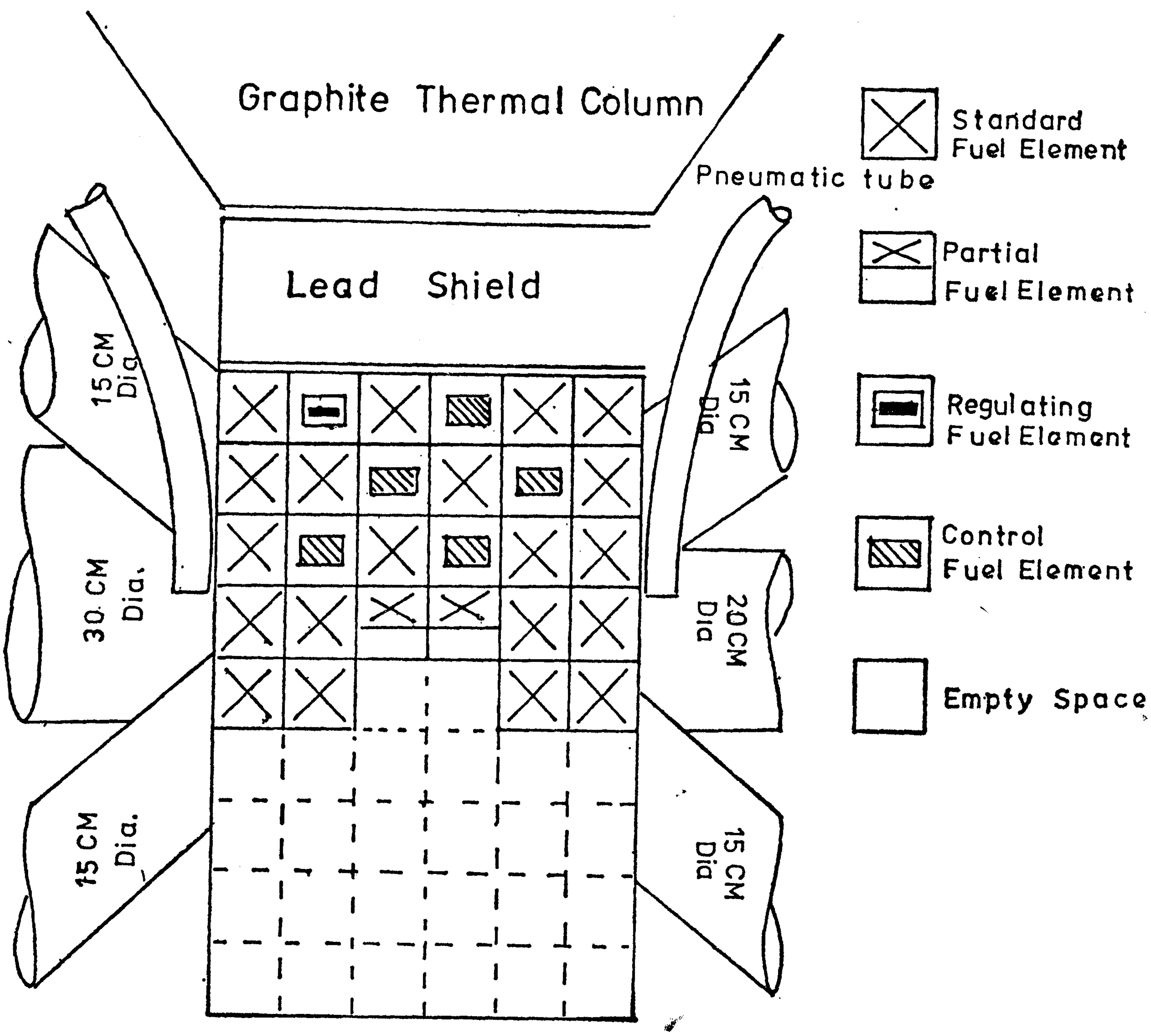

Fig. 1. NRC - Core Arrangement

and group constants were calculated as a function of irradiation time in unit of full power day (FPD). The cross section calculation was performed in a unit cell after homogenization of the cell and macroscopic cross sections in 45 groups were collapsed into five broad energy groups.

The solution of burnup equations yielded atomic densities of 16 actinides and 64 fission products as a function of irradiation time (FPD).

3.2 Few Group Diffusion Calculation

Solution of the multigroup diffusion equation yielded the eigenvalue, mesh point group flux and consequently power distribution in each fuel element. Burnup pattern in fresh core and in equilibrium core for a specified fuel cycle length, 10 FPD, were obtained. Table 2 gives results of the core calculations.

To study fuel shuffling scheme in an equlibrium core a fresh SFE was inserted in position 3 followed by sequential shuffling of the other SFEs and the last SFE was discharged from position 22, see Fig - 1. The fuel cycle length is a period of time during which a fresh SFE is irradiated in each position. Therefore the total length of irradiation for each SFE is 220 FPDs. The fuel cycle length is determined in such a way that

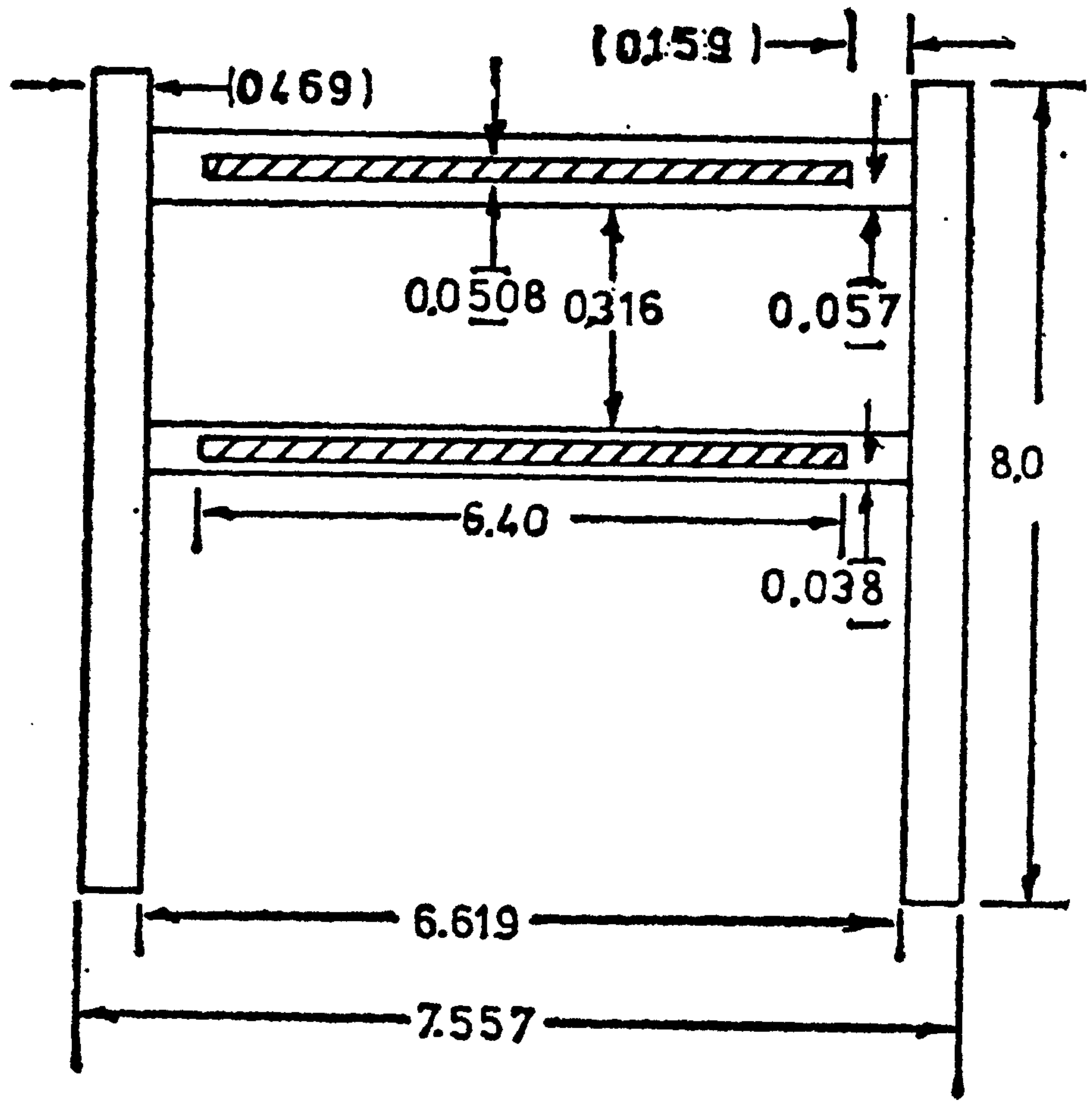

Fig. 2 a. Representation of physical geometry of fuel element - 5MW Core
Dimensions in Cm
18 plates St. FE & 8 plates Con. FE
End ptates are dummy in case of 93%fuel

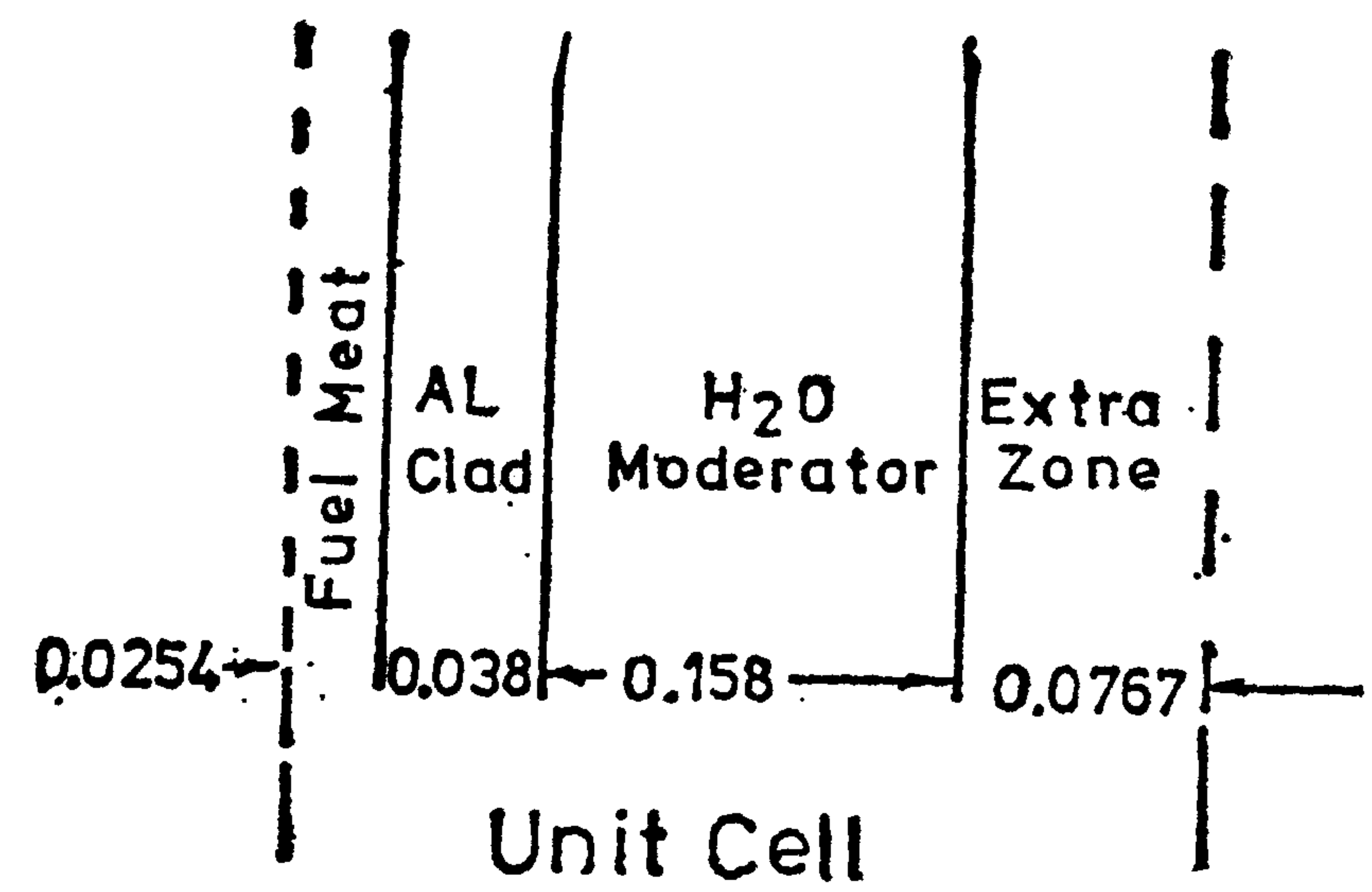

Fig. 2b. Unit Cell for cross section calculation - in RSYST
Dimenstions in Cm
18 Ptates St. FE

the core reactivity remains in controlable margin. As irradiation progresses core reactivity drops and there - fore Keff should always remain above unity. In this calculation control fuel elements were apparently unch - angeable but in fact they were replaced by new CFEs after one - half of the core life .

4. **Results and discussion**

Most of the results of calculations are shown in form of figures and tables. Table 2 gives Keff, burnup and U - 235 - loss as a function of irradiation time (FPD) and the same results are shown in Fig 4.

Fig 5 shows the integrated power generation per SFE. It is uderstood that the fuel burnup in each SFE can not easily be determined, that is because of sharp rise in flux distribution in water traps, see Figs 6. and 7 In normal operation about 60% of the control rods are withdrawn and therefore great part of power is generated in lower portion of the core. As fuel burns many fission products are produced and accumulated and also many actinides are born through successive neutron cap- ture. Majority of fission products are neutron absorbers among them Xe-135 and Sm-149 are most prominnt in absorbing thermal neutrons. Accumulation of fission products especially two above mentioned nuclides create rather high negative reactivity. Fig 8 shows variation of

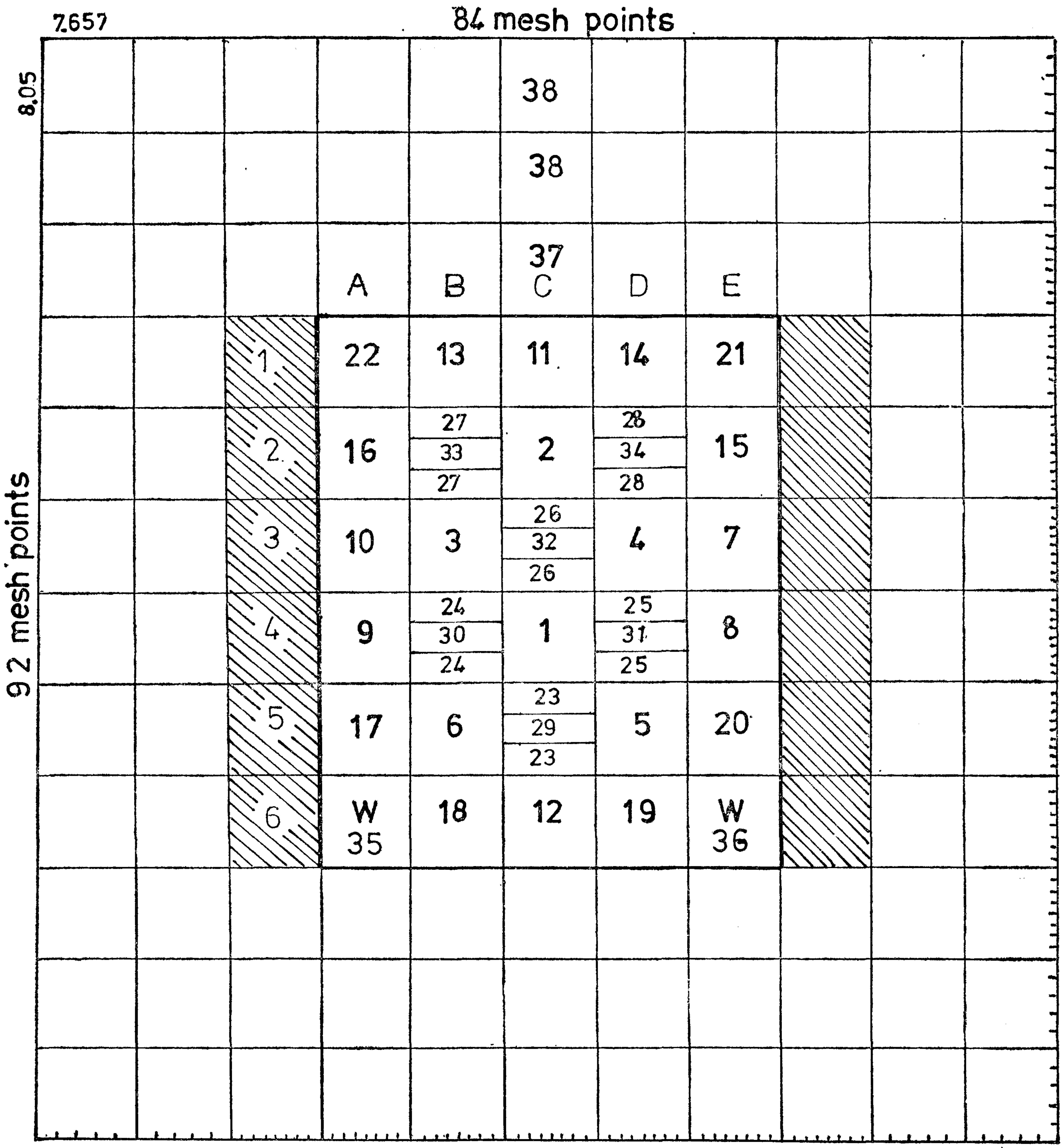

Fig 3. Core Shuffling pattern

Table 2 Summary of HEU Fuel Core Calculation

U235 SFE 93% plts SFE	BOC Keff EOC	Cycle Length (FPD)	Burnup at Exit MWD/Kg	U235— Loss %	U235 Loading in core
1904. 72	1. 09649	10	———	———	4868g
16A + 2 D					
194. 72	1. 01462	10			
16A + 2 D	1. 00540		19. 90	26. 36	4868g

A = Active D = Dummy

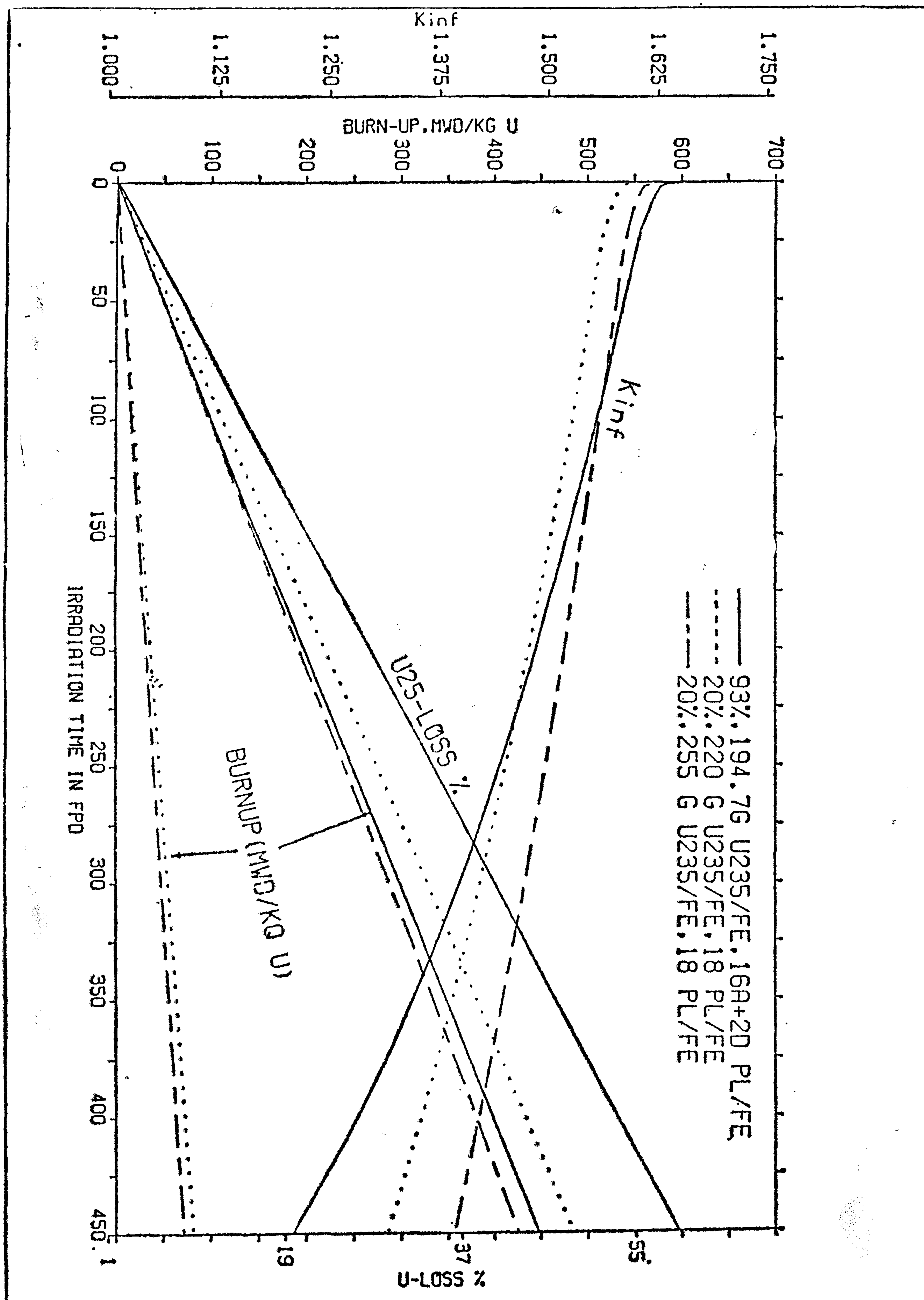

Fig. 4. Kinf, U235 - Loss and Burnup vs Irrad. Time

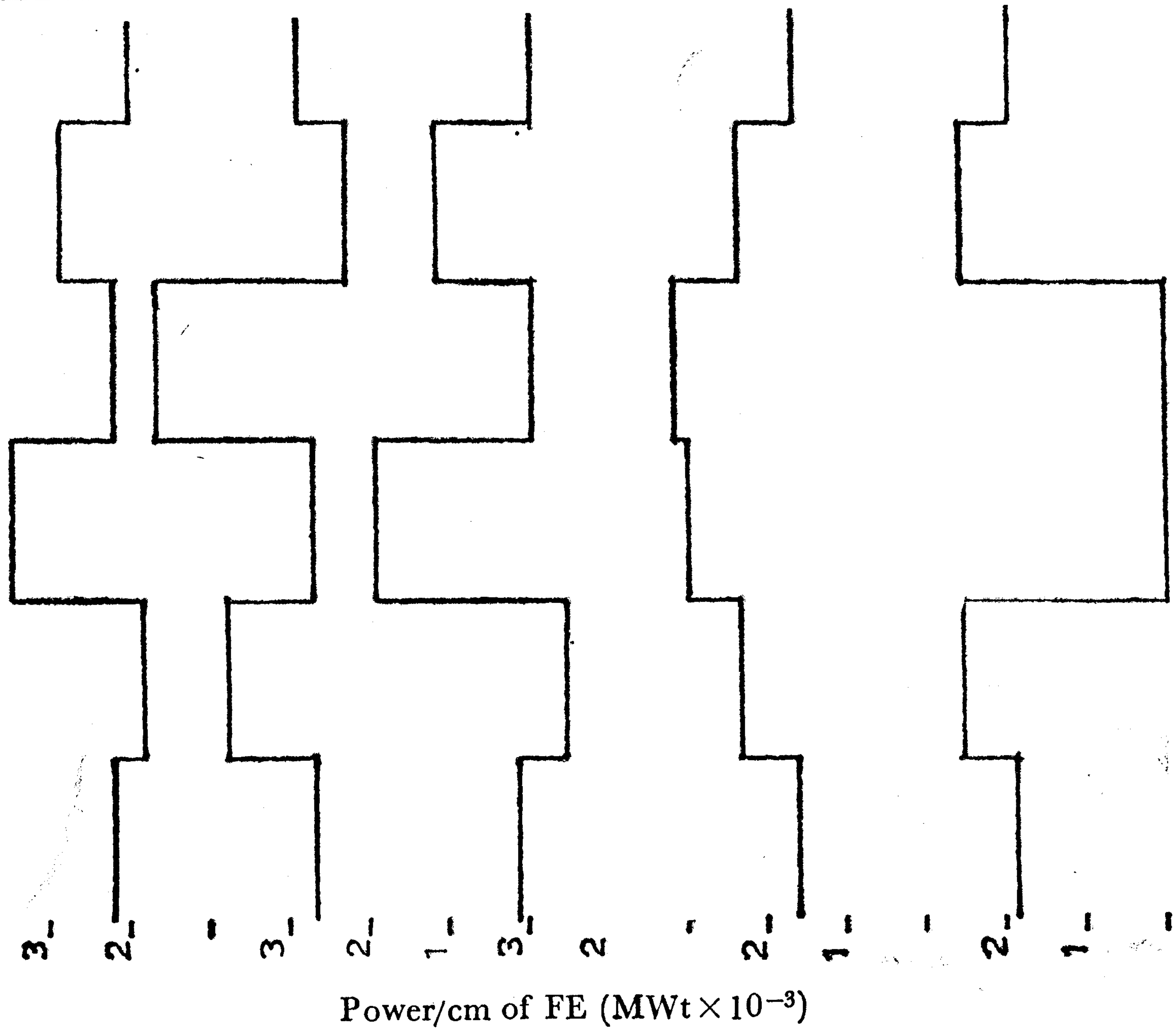

Fig. 5. Integrated Power FE in NRC Core

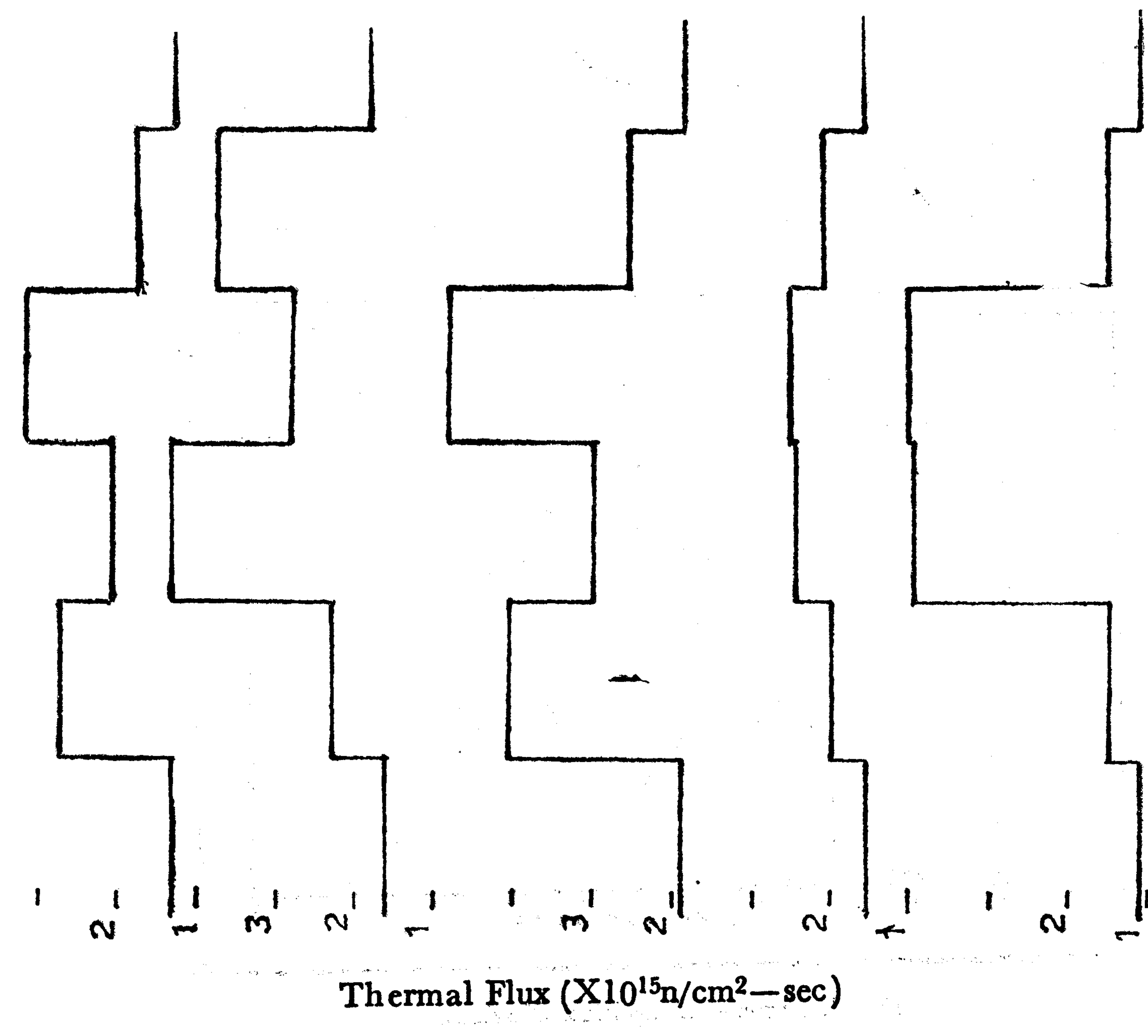

Fig. 6. Integrated Flux FE in HEU NRC Core

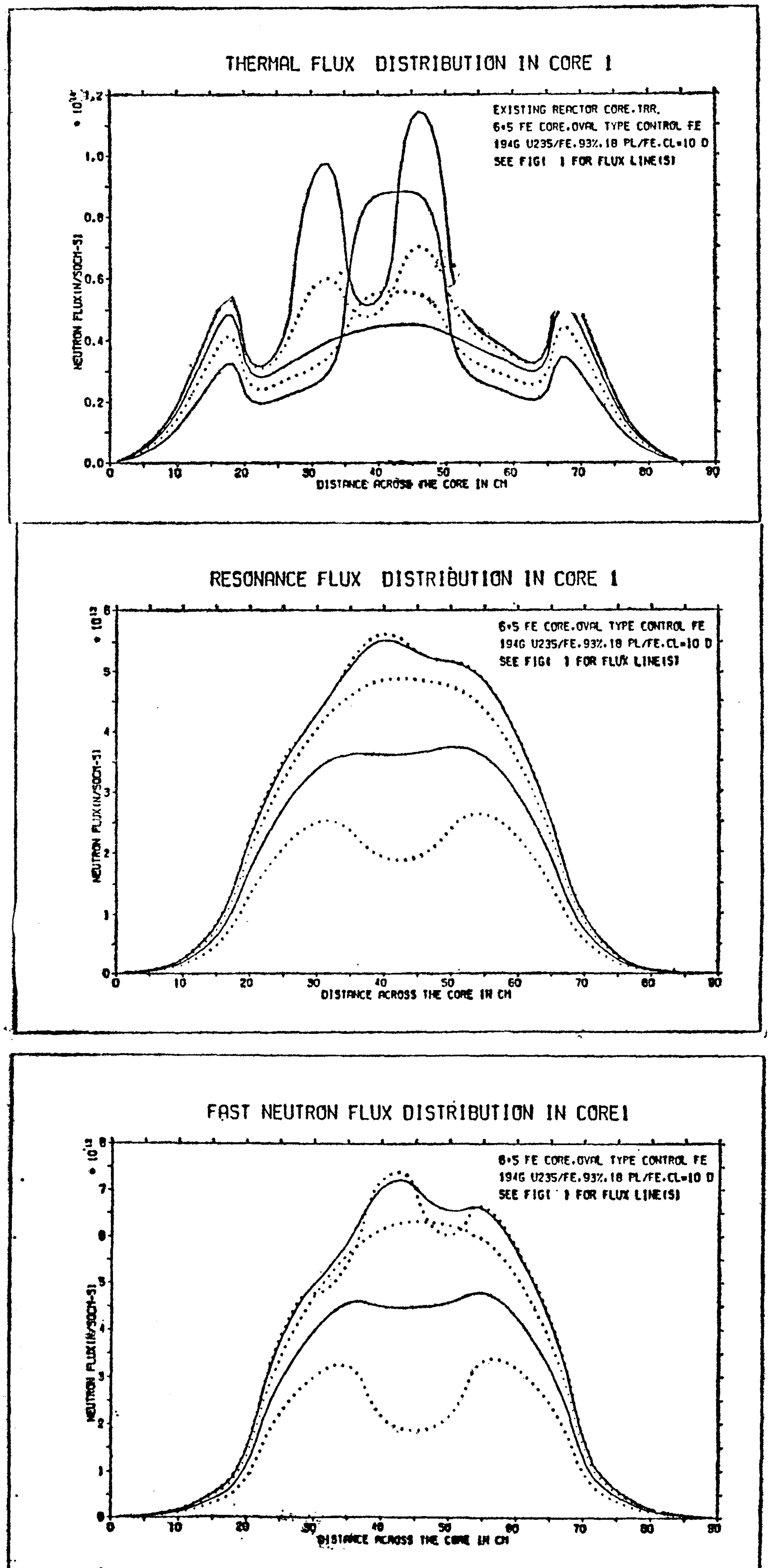

Fig. 7. Three - Group Flux Distribution

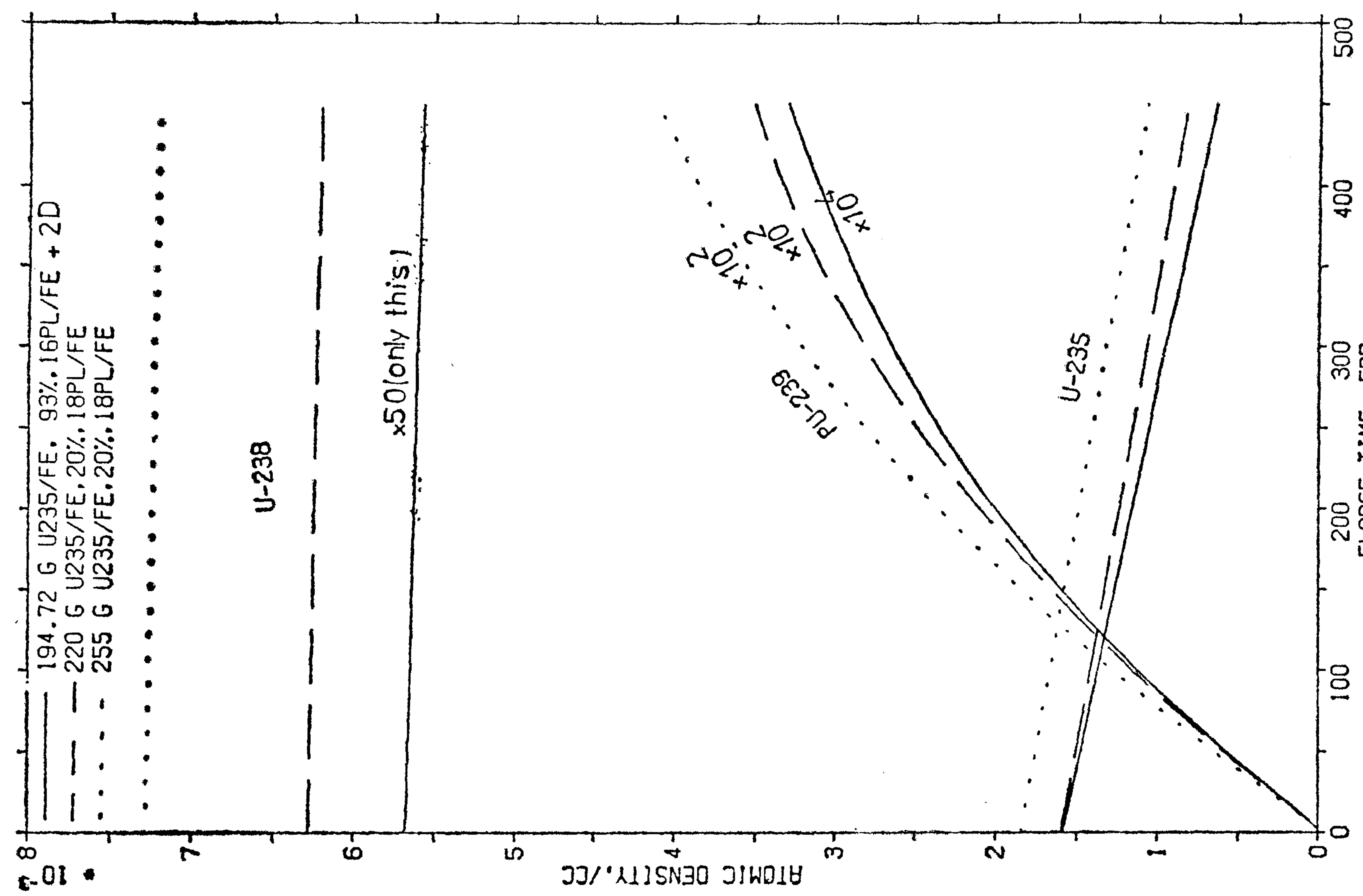

Fig. 8. Atomic Densities in Function of FPD

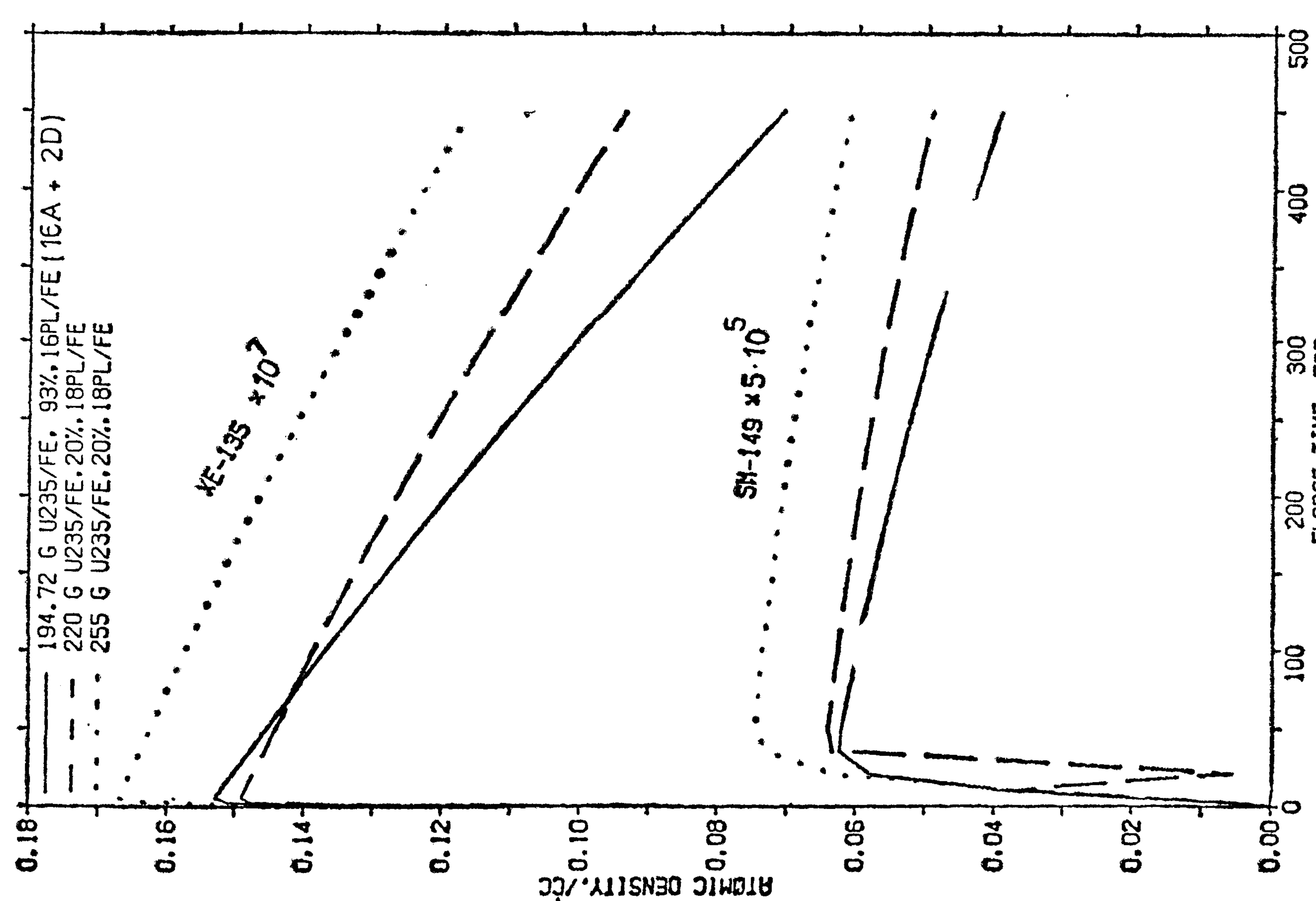

Fig 9. Atomic Densities in Funtion of FPD

atomic densities of Xe - 135 and Sm - 149 as a function of irradiation time. Some of the thermal neutrons suffer non - fissionable capture in uranium atoms genérating transuranic elements such as Np - 239, Pu - 239, Pu-240, Pu - 241 etc. which some of them are fissile and act similar to that of U - 235. In Fig 9 changes of atomic densities of U-235, U - 238 and Pu - 239 are indicated.

One should bear in mind that in fuel conversion from HEU to LEU thermohydraulic study should be performed since the neutronic behaviour and heat transfer conditions in the reactor core are intimately interrelated. The neutronic calculations show that the fuel replacement from HEU to LEU quite feasible even in upgrading power of the reactor .

Ahbreviations

NRC Nuclear Research Center

HEU Highly Enriched Uranium

LEU Low Enriched Uranium

FPD Full - Power - Day

AEOI Atomic Energy Organization of Iran

References

AMF (1959) Core Physics of Tehran University Research Reactor, Prepared by the AMF Atomic

Bonig K. and Van Der Hardt P. (1987) Physics and Safety of Advanced Research Reactors. *Nuc. Inst, and Methods in Phys. Research* **A260** 239 246 .

IAEA (1980) Research Reactor Core Conversion from the use of HEU to the use of LEU fuels. IAEA-TECDOC - 233.

Pazirandeh A., Bartsch G. and Lutz D. (1987) Neutronic Analysis of HEU to LEU fuels conversion calculations for AEOI 5 MW Pool Type - Research Reactor Core, TUBIK - 62, Institut fuer Kerntechnik (IKT), Technische Universitaet Berlin, W. Germany.

Ruhle R. (1975) RSYST I - III Experience and Features developments, *Atomkernenergie* (*ATKE*) **Bd 26** .

Ruhle R. (1983) RSYST Kurzberschreibung der Moduln, IKE - 4 - 5 January 1983, Institut fuer Kernenergetik und Enertgiesysteme.