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Institut für Angewandte Reaktorphysik
Institut für Material- und Festkörperforschung

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IRRADIATION PERFORMANCE OF FAST REACTOR FUELS *)

D. Geithoff, G. Karsten and K. Kummerer

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Gesellschaft für Kernforschung mbH., Karlsruhe

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ABSTRACT

The requirements for uranium-plutonium dioxide fuels are the basis for a program of irradiation tests which is outlined in the introduction. Some hypothetical considerations supply a rational background for the design of specimens and the chosen operation conditions. The experimental facilities and irradiation devices are shortly described and discussed with respect to the irradiation characteristics. The available hot cell equipment provides for the standard examination technique including X-raying by betatron and fission gas chromatography.

One group of experiments refers to the short term behaviour of the oxide type fuel. In another group the long time behaviour of oxide fuel pins is studied. Different pins have been irradiated in capsules and in a helium loop. The present results indicate, that the basic assumptions for the pin design are principally confirmed, the specimen failures only arising from malfunctions of the irradiation devices. The now available irradiation experience is used for the fabrication of further irradiation specimens of more realistic fast reactor dimensions and also for the concept of fast breeder pin specifications. Under these aspects the forthcoming main experiments are discussed. Finally the status of knowledge and experimental experience is composed in a set of conclusive remarks.

INTRODUCTION

The operational requirements for oxide type fuel, which are described in full details in the related paper[1] in the first session of this symposium, are the principal guidelines for all irradiation test work in the Karlsruhe fast breeder project. The most important technical features are the proper irradiation conditions with respect to linear rod power, to cladding temperature and to the expected total burnup performance. The typical external and internal geometry is characterized by a small fuel diameter (compared to thermal reactor fuel) and a length distribution which includes an

active fuel zone, at least one axial blanket region and a fission gas plenum. The typical ranges of requirements both for sodium and steam cooled pins are compiled in Table I.

The whole irradiation program is oriented towards these target data. As in the first period only a thermal reactor with a limited neutron flux, the Karlsruhe research reactor FR2, was available, significantly larger fuel diameters were necessary in order to reach the linear rod power wanted. Above the intention of a mere performance test up to high burnups - a goal which will be achieved by tests currently underway - the here described work refers partly to parameter studies. The short term tests try to evaluate a systematic scheme of parameter variations. The quoted long term tests are only a first step, which has to demonstrate the feasibility of the irradiation equipment and furthermore to experience the post irradiation examination routine. The next step with much higher target burnups and parameter constellations more similar to fast reactor conditions are already in operation, on the one side with improved irradiation devices in the FR2 again, on the other side with fast flux irradiations in the Dounreay Fast Reactor.

The present paper describes the status of our experimental potential and experience and also some introductory theoretical considerations. We feel it necessary, to establish simple model assumptions concerning the expected irradiation behaviour as a rough guide.

IRRADIATION HYPOTHESES

The target burnup for a fast breeder fuel pin is 100 000 MWd/t. In order to make predictions about the pin behaviour within an equivalent irradiation time, the pin design is based on a theoretical fuel pin behaviour analysis. In this context two limitating assumptions are made, namely

- the fuel operation time will come to an and, after the available void volume of the most critical fuel cross section has been filled by fission-induced fuel swelling to the highest possible extent,

- the fuel operation might end earlier by mechanical contacts between cladding and non-plastic fuel.

As a consequence of these assumptions three separate questions can be discussed with theoretical methods:

1. What is the maximum achievable burnup locally, if only the volume swelling due to fission products is considered?
2. Applying the answer of question 1, where is the most critical fuel cross section along the fuel pin located?
3. Does a socalled "non-plastic" fuel behaviour arise under the planned operation conditions, leading to a mechanical attack on the cladding?

In order to produce an answer to the first question a burnup model is established using some schematic considerations:

- The internal volume of the pin consists of the fuel volume V_F, the (hot) gap volume between fuel and cladding V_G, and the dishing volume (if applicable) V_D.
- The fuel volume itself can be divided into a plastic zone V_{pl} with temperatures above 1700°C, a creep zone V_{cr} with temperatures between 1300 and 1700°C, where creep and diffusion are of remarkable velocity, and a low temperature zone V_{lt} below 1300°C with visco-elastic behaviour mainly[2,3,4].
- Furthermore there are defined availability factors ℓ, m and n in the 3 fuel zones, which give the volume portions of the porosity being actually available for volume expansion due to swelling processes. The porous volume should be available[5] to any expansion with about 80 v/o above 1700°C, with 50 v/o at 1300-1700°C and - according to a numerical evaluation - with not more than 30 v/o below 1300°C. The last figure may rise somewhat during long time operation as the fuel becomes more plastic at low temperatures. Hence we apply for numerical evaluations $\ell = 0.8$, $m = 0.5$, $n = 0.3$.
- Finally two swelling processes are envisaged which need a continuously increasing amount of space with burnup. The basic swelling effect is defined here to be caused by solid fission products and a contribution of fission gases, working in all

three zones. According to a literature evaluation[5] this basic swelling rate h_s seems to be 1.6 % volume change in 100 % dense oxide per 10 MWd/kg heavy metal burnup. That means a figure $h_s = 1.6 \times 10^{-3}$ kg/MWd. An additional swelling effect caused by fission gases mainly is added for the creep zone. Its rate is assumed to be $h_g = 0.4 \times 10^{-3}$ kg/MWd.

With these explanations the maximum possible burnup B (in MWd per kg heavy metal content), after which all available volume is used up by the swelling fuel, is:

$$B = \frac{(1-\varrho_F)\,(\ell V_{pl} + m\,V_{cr} + n\,V_{lt}) + V_G + V_D}{\varrho_F\, h_s\, V_F + h_g\, V_{cr}} \qquad (1)$$

where ϱ_F means the fuel density in the pellet fuel. A graphic evaluation of this burnup formula is given in Fig.1 for a linear rod power of 500 W/cm.

The second question for the most critical cross section is handled by applying the burnup formula, which represents the fuel swelling capacity. Hence locally the maximum allowable burnup can be evaluated for the axial power distribution of a pin. These local maximum burnup figures have to be compared with the really arising burnup at each axial part of a pin in equal times. For certain parts of a pin, which are not necessarily the absolute maximum burnup ranges, the capacity for burnup is implemented firstly. These constitute the critical fuel cross sections. Such an analysis can be expanded of course on a total reactor core by application of specific thermal design data. Generally the critical zones are in the half of the core with the coolant inlet.

To deal with the third question for the existence of non-plastic fuel, the expansion rate of the fuel, which is given by the swelling rate, is compared to a compression rate, which is produced by an evaluation of fuel-cladding interaction, Fig.2. Here irradiation damages and thermal stresses with the cladding are taken into account by a pessimistic assumption for the 0.2 %-creep stress for 10 000 hours of the considered stainless steel. Furthermore, by application of mechanical oxide data[2,3,4] the force of the cladding on the swelling fuel ring with the volume V_{lt}

is calculated. The result is, that, under the assumption of 950°C average ring temperature, only those parts of the fuel with linear rod powers lower than 250 W/cm behave non-plastic. They are situated beyond the critical zones mentioned above and hence do not represent limitations for operation time in such designs.

EXPERIMENTAL FACILITIES

The irradiation program in the Karlsruhe research reactor FR2 - a heavy water-moderated reactor with 44 MW thermal power - is performed with two types of irradiation equipment. There is a helium loop operating in the central channel and furthermore, there can be introduced irradiation capsules into normal fuel subassembly positions. The helium loop can be loaded with two different irradiation inserts. For long term irradiation an insert is used containing 4 fuel pin specimens in a cluster. The maximum achievable linear rod power is about 800 W/cm with cladding temperatures up to 600°C. The second loop insert is movable in axial direction, also during reactor operation. Only one specimen can be irradiated at the same time. Here the maximum linear rod power can be up to 1000 W/cm, and both central temperatures and specimen inside pressure can be measured. This "short term loop insert" is primarily provided for irradiations in the time scale between a few minutes and a few days.

The irradiation capsules for fuel specimens use liquid metal layers consisting of lead-bismuth and sodium as heat transfer media. There are three variants available, two with lead-bismuth of eutectic composition - 26 and 19 mm in diameter - for specimens of 12 and 10 mm diameter. The third one has two separate layers of lead-bismuth and sodium at 26 mm capsules diameter, the specimen diameter being 7.4 mm in this case. The maximum linear rod power which may be produced is 400, 500 and 800 W/cm in these three designs. Each capsule can be loaded with four specimens some thermocouples being welded to the can or - in the third design - fixed in the sodium gap.

The post-irradition examination of fuel test specimen is

carried out in a group of hot cells. This group consists of five cells with heavy concrete shielding and an attached ceramographic line with 15 cm lead protection. With the shielding installed, it is possible to handle up to 10^7 MeV Curie of γ-rays in the cells and some 50 MeV Curie in the ceramographic line. All the facilities are completely α-tight and are operated by master slave manipulators.

POST IRRADIATION EXAMINATION METHODS

The majority of irradiated specimens is examined according to a standard procedure, which has been developed and exercised during the past years with only a few samples subjected to a more detailed examination. The most important examination steps are shortly described in the following paragraphs.

Isolation of specimens. Post irradiation examination starts with the isolation of a specimen. This means a simple disassembling procedure for the helium loop specimens, but is of a more complex nature for the capsule specimens. The complete removal of the lead-bismuth alloy, adhering to the can is accomplished by a combination of chemical and mechanical methods. Care must be taken not to do any sort of damage to the cladding, as this would impair precise dimensional measurements.

Visual examination. All fuel specimens are visually examined through the lead-glass windows in front of the cells, since no appropriate periscope is available at the present time. Slight magnification of the inspection zone is achieved by using a monocular. In the same way photographs are taken, which nevertheless show a remarkably good quality.

The leak testing method accepted for the fuel specimens is considered to be severe and has the virtue of simplicity. The procedure is to dip a specimen into liquid nitrogen until the thermal equilibrium is reached. When the specimen thereafter is quickly transferred to a transparent container filled with alcohol, any nitrogen trapped inside the can expands and produces a stream of bubbles thus showing location and size of a defect. In a few

cases helium leak testing has been tried to check the reliability of the nitrogen method. Both results are in good agreement.

Metrology. Profile readings are made with the help of an automatic profilometer. From each specimen three circumferential profiles and four profiles along the axis are recorded. The same readings are made before the irradiation. To avoid the difficulties of contamination of the unirradiated specimens, duplicate equipment has been installed for getting the preirradiation data.

γ-Scanning. γ-scans of the test specimens are taken with a collimator slit having the dimensions 0.5 mm x 20 mm x 700 mm. The specimen is rotating during the scan. Two energy regions are investigated, one being characteristic for Zr 95 - Nb 95 (700-800 keV), the other comprising Ce 144 and Xe 133 (100-150 keV). For short term irradiations the low-energy scan gives a good indication of the release of fission gas into the gas plenum, Fig.3.

X-Raying the specimens is considered to be one of the most important points in the post irradiation examination. The X-ray source is a betatron, emitting X-rays with a maximum energy of 18 MeV and having a focus of less than 1 mm^2. Details like dishing dimensions, cracks and central holes can easily be seen on the pictures. A quantitative evaluation of the X-ray photos concerning dimensional measurements shall be developed.

Fission Gas Release. After puncturing the can, the released gas is collected and analysed for krypton and xenon by means of gas chromatography. Gas volumes down to 0.5 mm^3 can be measured in this way. For double control, the activity of each fission gas fraction is determined in an ionisation chamber and the specific activity thus obtained is checked against the calculated value.

Sectioning of the Specimen. After the evaluation of γ-scans and X-ray-photos an individual sectioning plan is set up for each test specimen. Irregular zones of the fuel are picked out for ceramographic preparation, while samples for fission gas analysis and burnup analysis are taken from unaffected normal zones.

Burnup Analysis. The radiochemical method using Cs 137, Sr 90 and Ce 144 as burnup indicator has been applied in the past, when U 235 was the only fissionable isotope in the fuel. It is intended to turn to other methods like determination of fuel isotopic composition or of stable fission products.

Ceramographic Preparation. Cut sections of fuel pins are impregnated with epoxy resin under vacuum for ceramographic preparation. The solidified samples are mounted into bakelite holders and polished in the conventional manner. Grain structure is made visible by etching with HNO_3 and H_2O_2.

Autoradiography. Contact α- and β-γ-autoradiographs will be taken from samples which have been mounted for ceramographic examination. This point of examination is still in development.

Fission Gas Trapped in Pores. For determining fission gas, that is trapped in larger pores, fuel samples are ground to small size grains and the released fission gases are quantitatively determined by gas chromatography. The grinding process is controlled by intermittent determinations of the free surface of the powder using the BET-method.

Fission Gas Trapped in Fuel and Small Pores. Ultimatively the ground fuel samples are dissolved in nitric acid and the released fission gas is determined in the manner mentioned before. With all the fission gas data a balance is made, which shows how much of the gas is released at each step.

SHORT TERM IRRADIATION PERFORMANCE

Short term irradiations are being performed with the movable insert in the helium loop of the FR2. Different fuel variants are tested in order to study the parameter dependent behaviour of the fuel on operation after short duration, that is shortly after a reactor startup or on short power excursions. The parameters which are varied are the linear rod power, the density, the fuel form and the gap width at pellet type fuel. Furthermore central tempe-

rature measurements are carried out. Vibrocompacted fuel is irradiated as sintered and molten oxide, including pressure-buildup measurements in both cases. The Table II gives a survey of a detailed program.

Meanwhile most of the specimens were irradiated with the given parameter constellation. After some cooling time an examination routine comprising X-raying by a betatron device, γ-canning and ceramography was performed. The details of the results are presently compiled.

LONG TERM IRRADIATION PERFORMANCE

The long term irradiation program was started with the aim of making selection of the fuel type and form. It is being continued in order to study the long time behaviour of fuel pins, which are produced with respect to some hypothetical deliberations. Thus in the first group, which is described in Table III, UO_2 and UO_2-Mo were irradiated for fuel selection. As to the pellet form a choice of ground and unground condition and also dished and undished shape was investigated. The vibrational compacted powder was included in a sintered and a molten form. Furthermore, by application of advanced capsule types the fuel diameter could be reduced to 6.4 mm. Though very high burnups were the target, the start of the program was accompanied by pin failures mainly due to capsule failures. Thus the maximum burnup, which was achieved in the first run of experiments did not exceed 20 000 MWd/t. Meanwhile the irradiation performance has been improved very much by improvements in the capsule design. The introduction of sodium as a heat transfer medium restricted the corrosion and void formation at reactor shut down.

Up to now all capsules of the first part (up to KVE 7) are post irradiation examined. As all the different specimens of this part were operated with 360 W/cm, the results can reasonably be compared. The specimens had a fuel diameter of 10 mm, an active fuel length between 225 and 228 mm and were canned in stainless steel. In the following some remarkable features are illustrated

and discussed for vibrated powder as well as for pellet fuel.

Vibratory Compacted Powder Fuel. All specimens with vibro-fuel has a uniform fuel density of 85 % of theoretical density. They were irradiated up to burnups of about 8 000 MWd/t. Although some of the pins achieved a considerably lower burnup, their behaviour concerning gas release and their appearance were very similar to the higher burnup specimens.

Fig.4 and Fig.5 show two length sections of the same pin (KVE 3 / Specimen 3). The central hole is formed as expected with enlarged ends on both sides. The lower end of the central hole is filled with fuel debris up to a height of 1 cm. As close-up views show, the debris is mainly originating from the fuel zone adjoining the can with only a few particles coming from the surface of the central hole. Radial cracks might be the pathway by which the transport is made. The hole was filled at a rather early date after the end of irradiation as X-ray photos already show the filling. The cross sectional area of the central void covers 2.5 % of the internal pin cross section or 5 % of the densified central zone.

The fission gas data for the vibrated specimens are as follows:

52 - 56 % released on puncturing the can
14 - 20 % released on grinding fuel sample
30 - 34 % released on dissolving fuel sample

The combined fission gas data were precise enough as to be the basis for burnup calculations, which within $\pm$ 20 % agreed well with the radiochemical values.

Fig.6 (KVE 7/Specimen 8, UO_2-Powder Fuel) shows a cross sectional view with a central hole apparently located excentric. Higher porosity on one side (smaller powder particles) causing lower thermal conductivity of the fuel in this region has drawn the central void closer to this direction.

Powder fuel with molybdenum as additive, a fuel type considered promising at an earlier date, was tested in a few specimens in KVE 7. The idea that the addition of 10 % molybdenum will increase the thermal conductivity considerably, thus lowering the central temperature, is visibly confirmed by Fig.7 (KVE 7/Speci-

men M7). The compacted particles have been molybdenum coated. No grain growth is observed throughout the whole fuel area which means that central temperatures remained below 1500°C in the center. Fission gas release was reduced to less than 5 % with only an additional 10 % release found on grinding the fuel. More than 80 % of the gas was strongly trapped in the fuel itself.

Pellet Fuel. The fuel specimens referred to in this chapter contained pellets with fuel density of 91 % th.d. They were dished on one side with a dishing volume of 2.7 - 2.8 %. Cladding and fuel were separated by a gap of 60 µm diametral. With this type of fuel specimen burnups up to 8 000 MWd/t were achieved.

The influence of the gap width on the formation of the central hole can be discerned in a X-ray photo, although the gap width was not included as parameter in the study. A stepwise dislocation of the central hole in two neighbouring pellets, see Fig.8 (KVE 4/Specimen B) can be reasonably interpreted only as a result of different gap widths which are caused by filling the pellets in such a way into the canning that one side touches the canning while the other leaves a gap of the whole 60 µm.

A dislocated central hole and an additional cavity having the shape of a half moon is shown in Fig.9 (KVE 4/Specimen B). The cavity lies on the borderline between uneffected UO_2 and visible grain growth. From the shape of the cavity it can be concluded that this is what has been left over from a pellet dishing.

To study the function of the dishing in pellets was one of the objects of the irradiation test. Fig.10 (KVE 4/Specimen D) shows how the dishing was consumed by the expanding fuel. (The oblong inclusion in the middle pellet is an impurity introduced in the manufacturing process.) The pellets have protruded, apparently from both sides, and nothing is left of the dishing volume in the central zone. Only a doughnutlike ring in the cooler fuel zone marks the position of the former cavity. The consumption seems to be irreversible but there is not indication where the void volume has vanished.

In a stage where the central hole has been formed, the dishing volume becomes visible again, Fig.11 (KVE 4/Specimen D).

At pellet interfaces the central channel is widening to form the diamond shaped void. This form of the central void may be smoothed to constant diameter at longer irradiation times but with the burn-ups we achieved so far it was always discernable.

What can happen to a dishing void at low rod power is demonstrated in Fig.12 (KVE 4/Specimen A). Due to thermal cracking in a test specimen with a linear rod power of only 250 W/cm (marginal position in KVE 4) the dishing void is filled with a large piece of the upper pellet, thus transferring the void to another position. This phenomenon has only been observed once. So it will not be the cause for smearing the dishing void volume uniformly along the whole fuel length. That this is only a single case can be documented by Fig.13. The shifted void can be easily detected in the X-ray photo.

IRRADIATION PROGRAM CONTINUED

The irradiation performance tests are being continued in thermal and fast reactor environments. Presently a new series of UO_2-PuO_2-specimens for irradiation both in capsules and in the helium loop of the FR2 is prepared. Fast neutron irradiations are carried out in the Dounreay Fast Reactor for our program, where at present a trefoil rig with UO_2-PuO_2 pins is in operation and a bundle irradiation is being prepared. Also irradiation space in the hard spectrum of the BR2 reactor in Belgium is now available and will shortly be used for testing fast reactor fuel pins.

CONCLUSIONS

In view of the still rather limited experimental experience concerning the irradiation performance of fast reactor fuels the following conclusions are established:

1. With mixed oxide fuel a linear rod power of up to 500 W/cm maximum - as required for present fast reactor prototype

designs - can be achieved without central melting. Also the burnup range of up to 100 000 MWd/t is a reasonable target; it needs, however, a careful adjustment of the internal fuel geometry.

2. An additive to the fuel like molybdenum improves the thermal conductivity to some extent. This improvement - being lower than originally expected - is according to the recently established hypothesis not only not necessary, but also harmful to the radial swelling problem, as a reduced average fuel temperature lowers the availability of porous volume to fission products.

3. According to the present state of fabrication technology the pellet type oxide fuel has some preference compared to vibrocompacted powder fuel. But the irradiation results do not demonstrate any advantageous features for one of both "competitors".

4. The short term behaviour of the fuel (at startup or at power level changes of the reactor) seems to be quite smooth and normal, without any indication of undesired internal fuel movement.

5. The applied internal geometry parameters (low smeared density, gap width and dishing or non-dishing at pellets) are experimentally confirmed within the now evaluated limited burnup range.

ACKNOWLEDGEMENT

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Table I TYPICAL REQUIREMENTS FOR IRRADIATION PERFORMANCE TESTS

	Sodium Cooled Version	Steam Cooled Version
Max. Linear Rod Power, nominal (W/cm)	400-500	300-400
Max. Can Midwall Temperature, hot spot (°C)	650-700	700-750
Max. Burnup (MWd/t of U + Pu)	~90 000	~50 000
Fuel Diameter (mm)	5.0 - 5.5	6.0 - 6.5
Fuel Length (mm)	~1 000	~1 500

Table II SHORT TERM IRRADIATIONS

Specimen Identification for the Different Parameter Constellations

Linear Rod Power (W/cm)				500			750			1000			400
Irradiation Time				10 min	2 h	24 h	10 min	2 h	24 h	10 min	2 h	24 h	24 h
Fuel Form	Diam. Gap (μm)	Fuel Density (%th.d.)	Smear Density (%th.d.)										
Pellet	100	93	90	L1	L2Z		L1X	L2		L1Z	L2X		L3**
		88	85	L4	L5	L6	L7	L8	L9	L10	L11	L12	L13**
		93 (5% dish)	85	-	L14	L17	-	L15	-	-	L16	-	-
	250	90	85	-	-	-	L18	L19	-	-	L21	L20	L22**
		90 (5% dish)	80	-	-	L23	L24	L25	L26	-	-	-	L27**
Vibro Powder	Sintered Particles		85	-	L28	L29*	L30	L31	L32*	L33	L34	-	
	Molten Particles		85	-	L35	-	L36	L37	L38*	-	L39	-	

* Specimens with inside pressure measurement

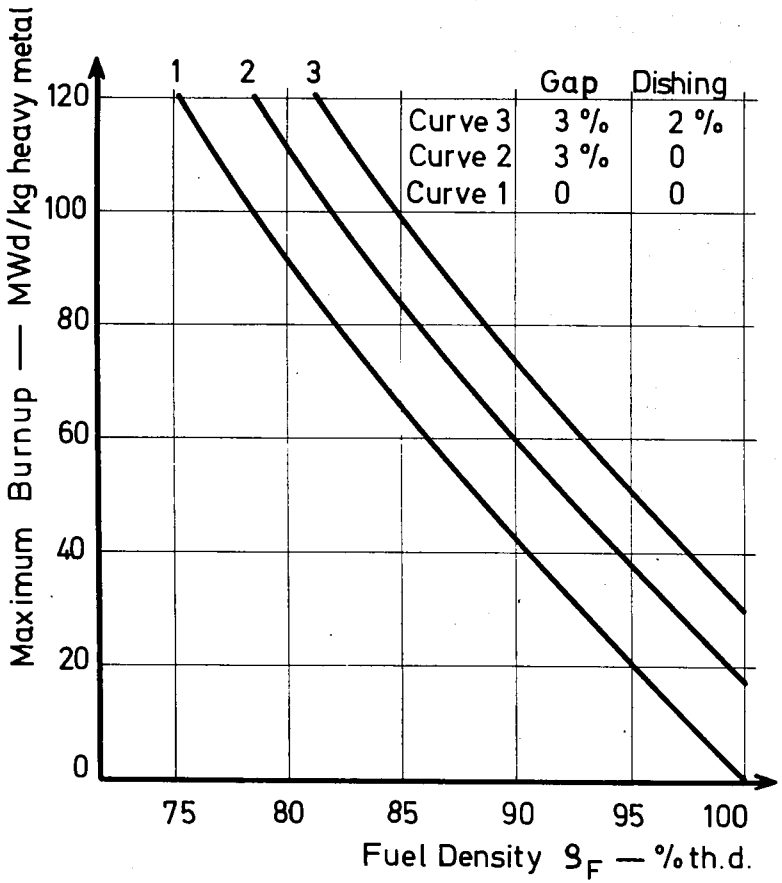
** Specimens with central temperature measurement

Table III LONG TERM IRRADIATIONS

Capsule No.	Number of Specimens	Fuel	Fuel Form	Fuel Density (%th.d.)	Fuel Diameter (mm)	Irradiat. Time (days)	Linear Rod Power (W/cm)	Max. Burnup (MWd/t)
KVE 3	4	UO_2	Sintered Particles, Vibrated	85	10	13	360	690
KVE 4	4	UO_2	Ground Pellets with Dishing	91	10	22	360	730
KVE 5	2 2	UO_2	Ground Pellets with Dishing Sintered Particles, Vibrated	91 85	10	147	360	7770
KVE 7	1 1 1 1	UO_2 UO_2-Mo UO_2-Mo UO_2	Sintered Particles, Vibrated Sintered Particles, Vibrated Ground Pellets with Dishing Ground Pellets with Dishing	85 85 90 91	10	86	360	4545
KVE 10	4	UO_2	Molten Particles, Vibrated	86	10	36	450	1300
KVE 11	4	UO_2	Unground Pellets without Dishing	90	10	58	450	2600
KVE 12	4	UO_2	Sintered Particles, Vibrated	86	8.6	+	500	16000
KVE 13	3	UO_2	Ground Pellets with Dishing	90	10	145	450	5000
KVE 17	1 1	UO_2	Sintered Particles, Vibrated Molten Particles, Vibrated	86	8.6	+	500	>8000
KVE 20	1 2	UO_2	Ground Pellets without Dishing Unground Pellets without Dishing	93	6.4	30	650	5000
KVE 21	2	UO_2	Unground Pellets without Dishing	93	6.4	+	650	

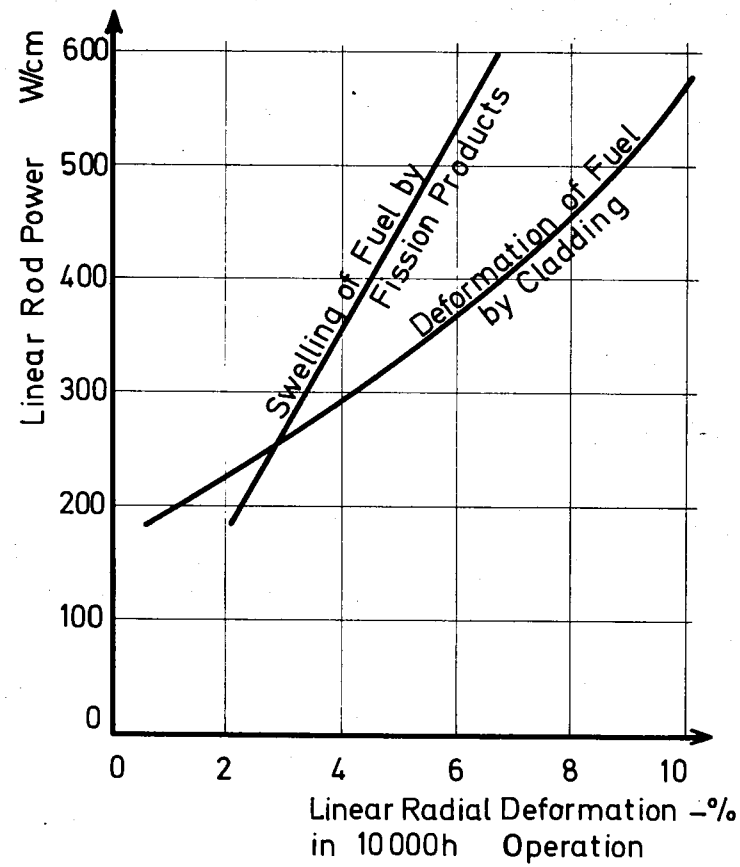
+ in operation

LIST OF FIGURE CAPTIONS



Linear Rod Power 500 W/cm
Fuel Surface Temperature 900 °C
Average Thermal Conductivity 0,022 W/cm °C

Fig.1 Maximum Burnup Evaluation



Cladding Temperature 550-650 °C
Temperature in the Swelling Fuel Ring 950-1000 °C

Fig.2 Fuel-Clad Interaction

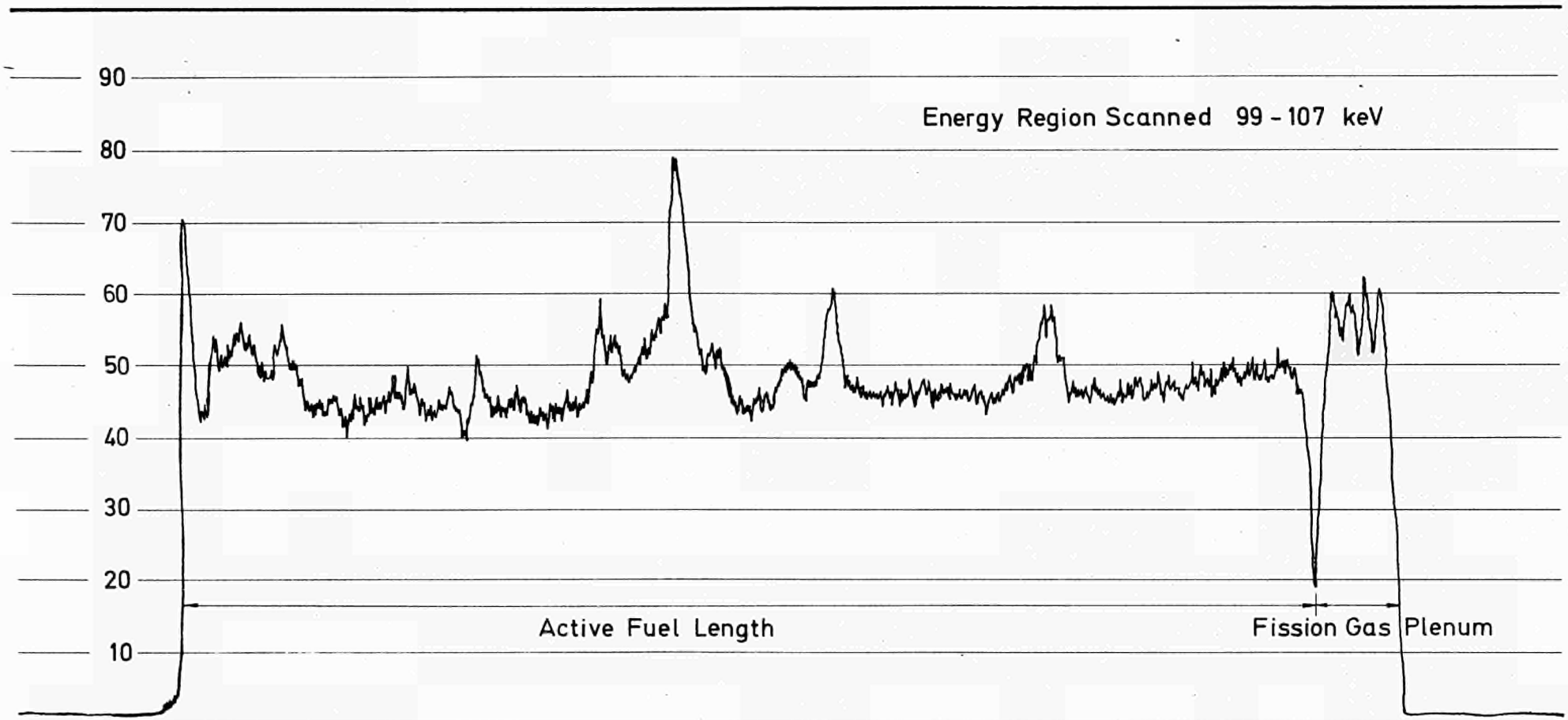


Fig. 3 γ - Scan of Specimen L 9

Fig.4 Bottom Section of a Powder Fuel Specimen (KVE 3/3)

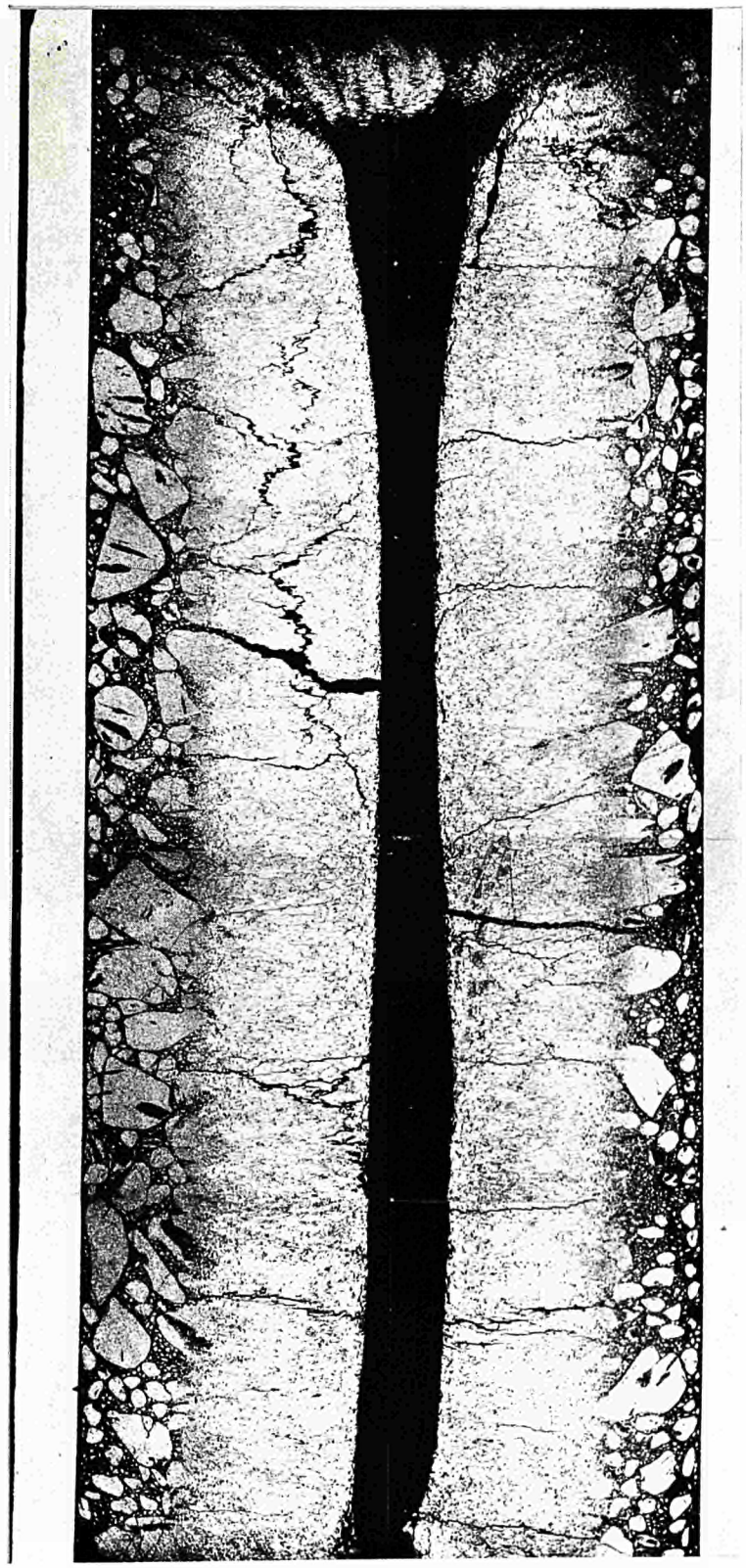


Fig.5 Top Section of a Powder Fuel Specimen (KVE 3/3)

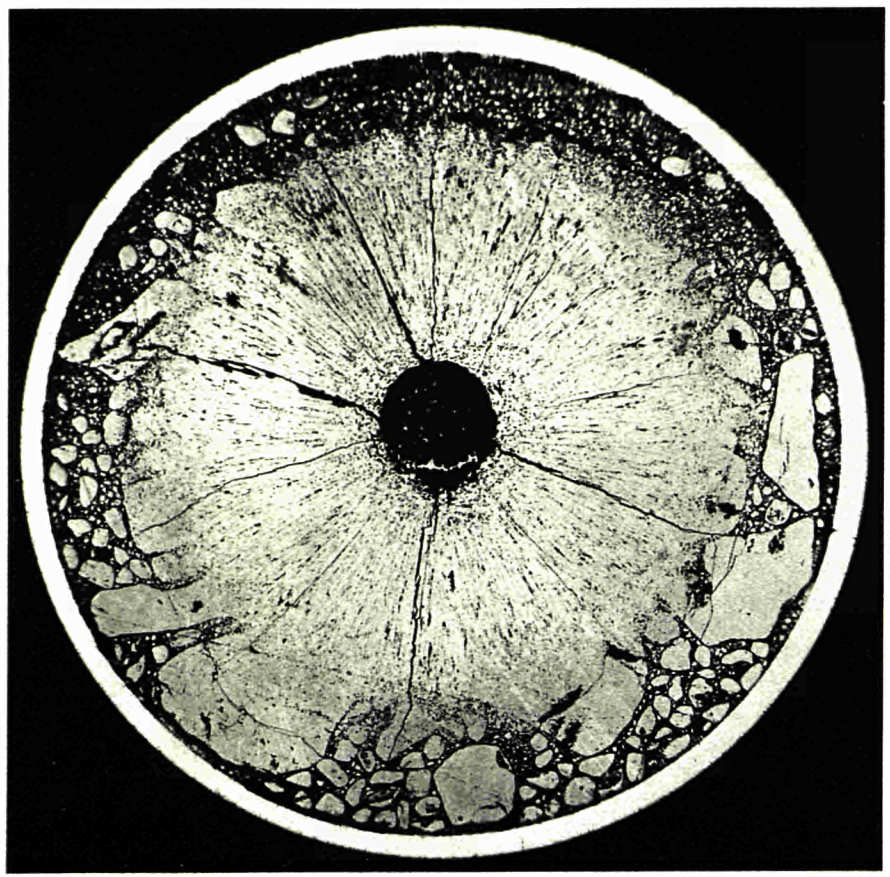


Fig. 6 Cross Section of a Powder Fuel Specimen with Excentric Central Hole (KVE 7/8)

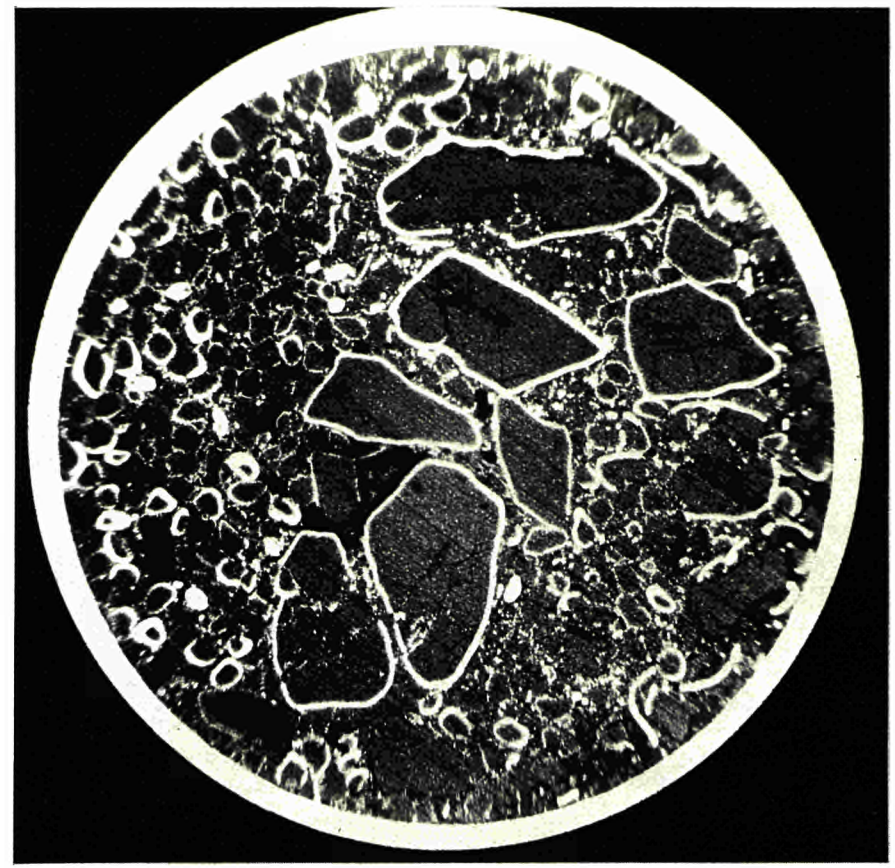


Fig.7 Cross Section of Mo-coated Powder Fuel Specimen (KVE 7/M7)

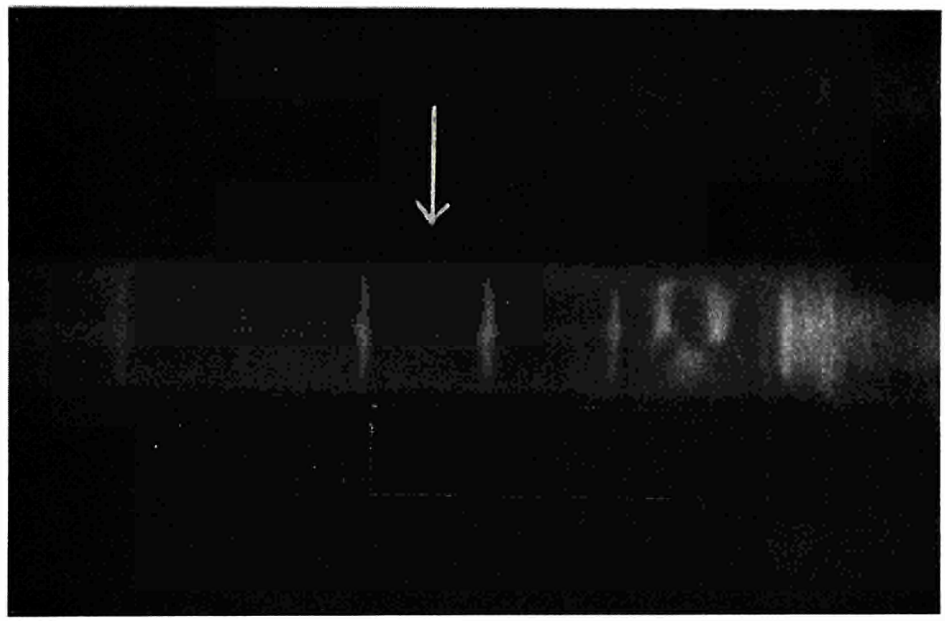


Fig.8 Stepwise Dislocation of the Central Hole in a Pellet Fuel Specimen (KVE 4/B)

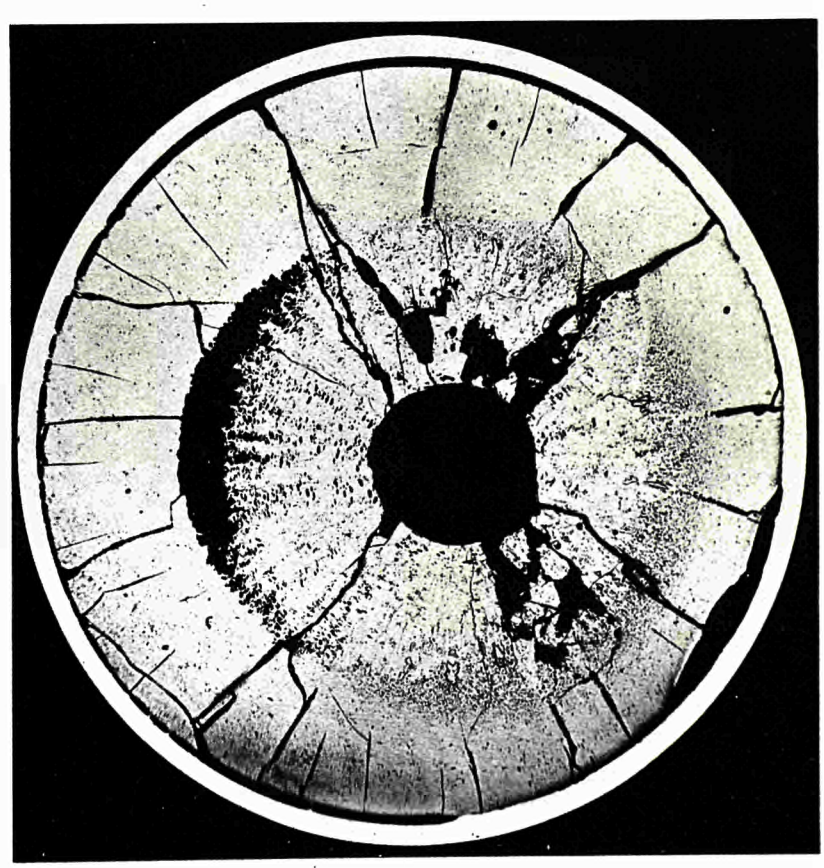


Fig.9 Cross Section of a Pellet Fuel Specimen with Dislocated Central Hole (KVE 4/B)

Fig.10 Disappearing Dishing
in Pellet Fuel (KVE 4/D)

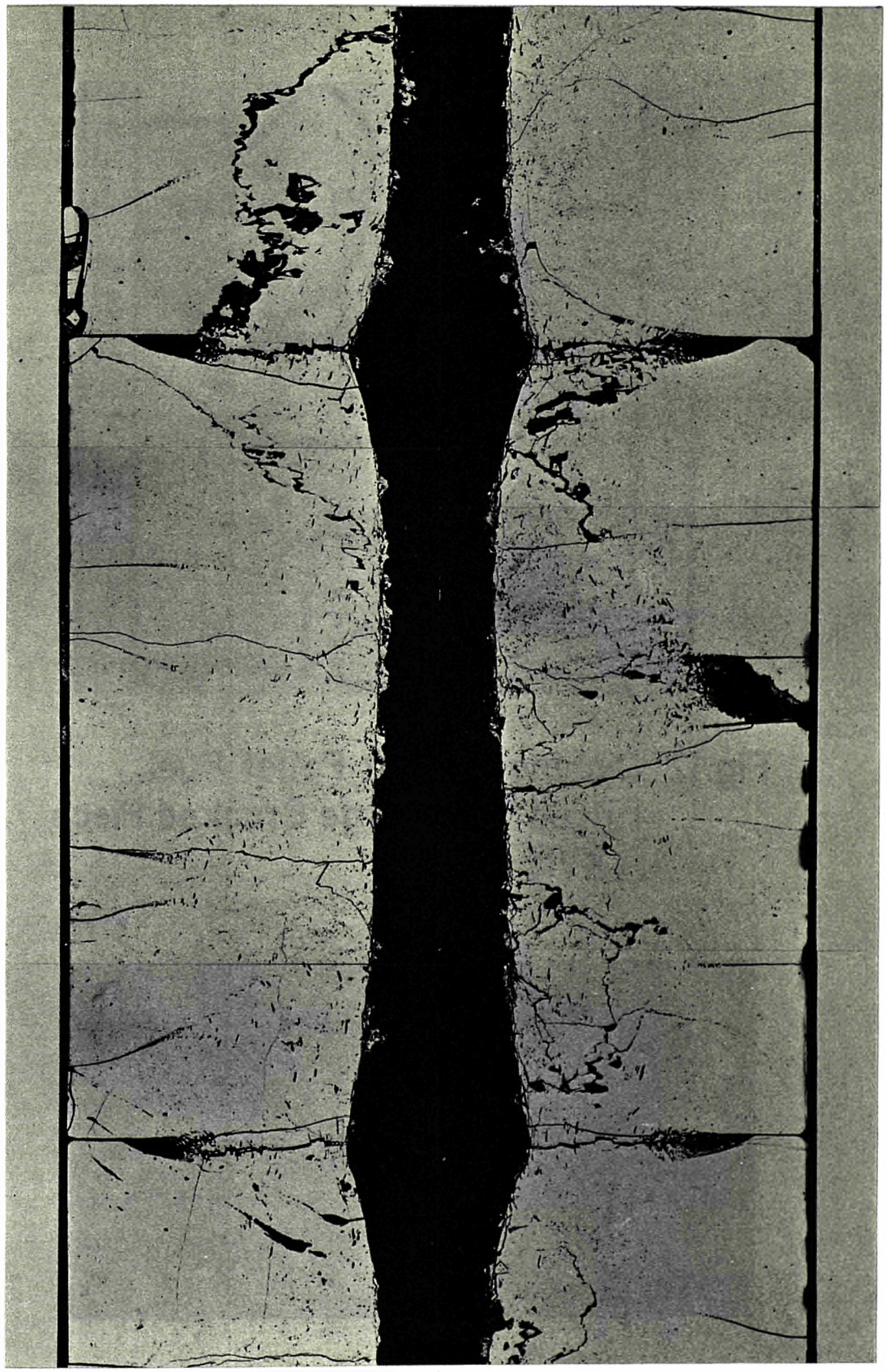


Fig.11 Diamond Shaped Void Formation in Pellet Fuel (KVE 4/D)

Fig.12 Dishing Void at Pellet Fuel Filled by a Large Cracked Piece (KVE 4/A)

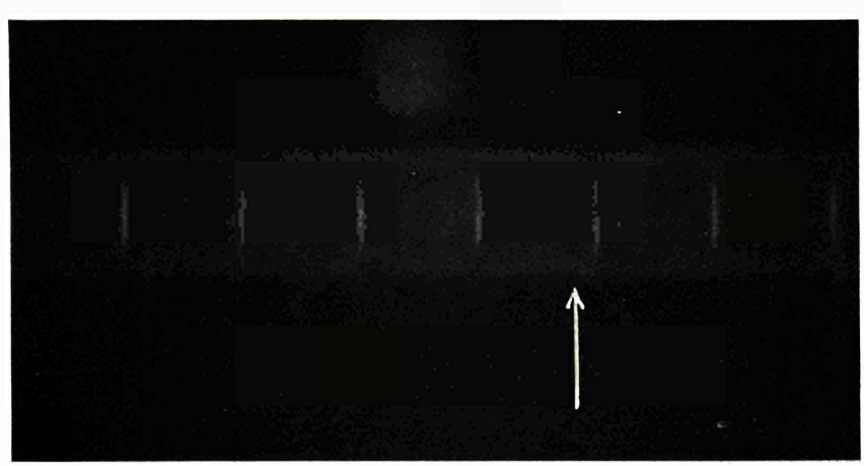


Fig.13 X-ray Fotografy of the Dished Pellet Fuel of Fig. 12