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An FRG-Proposal for an International Intercomparison of Codes for Radiation Protection Assessment of Transportation Packages

Introduction

In June 1979 CSNI convened a meeting to investigate the possibility of setting up international projects on fuel cycle safety. There most NEA members expressed their interest in exchanging information and experience on various aspects of spent reactor fuel transportation. Special points of interest were criticality, heat transfer and shielding code intercomparison.

Up to now the first part of the work, the criticality code intercomparison, has been conducted by a CSNI working group. A report containing problem description and a summary of results obtained by the participants, together with the conclusions reached was published in May 1982.

For the next topic, heat transfer assessment, the United States proposed a first set of model problems to be benchmarked.

The following proposal suggests a procedure for shielding code comparison and verification and includes a couple of example problems. It takes into account the comments on the first proposal made by the participants.

Shielding analysis for radioactive material shipping casks requires the solution of some problems by more or less specialized calculational techniques developed for this purpose. The computation of the neutron and gamma ray source term (absolute value and local distribution, respectively), the complex geometry e.g. of a fuel assembly consisting of a bundle of distinct rods which allow for streaming effects, presence of neutron absorbing materials in the cask wall and different types of cooling fins at the cask surface are some examples for the problems mentioned above.

As outlined in previous papers, "packages approved by national authorities for transport of fissile materials, including spent fuel, may not be accepted for international shipment unless the certification process in the originating country is examined and approved by the other countries in which the transport will occur". So it would be desirable to include the shielding calculational techniques for the approvement of spent fuel shipping casks into the international intercomparison. Special care should be taken to the point that for the intercomparison such methods should be used preferably which are standard practice in the participating countries.

This proposal suggests the procedure for accomplishing such an intercomparison exercise under the aegis of the NEA-CRP.

The concept on which the proposal is based is that it is necessary to validate calculational techniques against actual experimental data as well as comparing results among several codes applied to the same basic problem. For that purpose it would be very useful if the participants would contribute experimental data with respect to their special knowledge. In this manner, measures of both consistency and accuracy should be obtained when the assessment procedures are completed.

This comparison and verification would be accomplished by the calculation by each participating country of a series of problems important to package certification procedure. Code results would be made available to all the participants and report all the results issued.

Procedure

To carry out this exercise it is proposed that a working group be established, comprised of specialists from the interested participating countries. This group would be concerned with:

(1) Selection of standard problems, including:
- Use category (spent fuel storage, spent fuel transport, fresh fuel, plutonium, high active waste)
- Parameters of transport package to be benchmarked (general contructional design, cask inventory, materials, construction details)
- Basic quantities (fission spectrum, flux-to-dose-conversion-factors to be used)

(2) Compilation of available experimental data for related configurations due to special national contacts of the participants.

(3) Analysis of calculations and preparation of a report on the intercomparison results.

Specific Proposal for an International Standard Problem Exercise in

Shielding Code Benchmarking

The most stringent regulatory requirement governing the shield design of a shipping cask are the dose rate limits for normal operation conditions of the IAEA transport regulations. In the case of a spent fuel storage facility an additional restriction can arise when the national authorities set a limit for the environmental radiation exposure. So the most important quantities to be checked for are the mean value of the dose rate at the cask body surface and in some given distances. A point of some additional interest, especially for cask handling and maintaining, is the dose rate profile along the cask body. For a given transportation package to be licensed some points have to be considered which can affect considerably the calculational result:

- Arrangement of radiation sources in the basket (e.g. streaming effects in rod bundles, inhomogenities in source density)

- Basic parameters of the radioactive material (e.g. for spent fuel: composition of fresh fuel, irradiation history, reactor type)

- Constructional details (wet or dry cask, material composition, basket and cask body construction, location of neutron absorbing materials, cooling fins, instrumentation holes)

In principal most of these effects can be treated by exactly three dimensional modelling of the cask as a whole. Since this is a very time and money consuming way often the problem is simplified so that it is treatable by less sophisticated methods. But these simplifications have to be performed very carefully in order to avoid misleading results.

Having these margins in mind we suggest to divide the standard problem exercise into two main parts:

- A first one dealing with the influence of different geometrical models on the calculated dose rate

- A second one in order to check the different methods for determining the neutron and gamma source strength as an essential input data quantity for the following shielding calculation. This part is understood to be only a starting point which can be followed by some other problems of interest.

The main ideas in setting up the proposal were

- to start with a set of three rather simple problems in order to get a basis of consensus due to different calculational models.

- to add just one complexity in a subsequent problem. So we leave dimensions and materials unchanged in all model problems.

- to choose package conditions which require at least a minimum of neutron shielding. So we preferably choose a dry cask containing spent LWR-fuel.

- to use input data which are as realistic as possible. So we do not give a detailed description of the fuel assembly top and bottom fittings because these data are usually unknown to the shield designer.

On the next page a general view is given showing the main features of the different model problems. Each of them is discussed in detail in the following sections.

The nomenclature and basic quantities to be used can be found in table A1 and A2, respectively.

General view of Model Problems

PART I: Geometric Modelling

Attribute	Model Problem				
	I	II	III	IV	V-VI
predefined source	yes	yes	yes	yes	no
neutron shield	no/yes	yes	yes	yes	yes
cooling fins	no	no	yes	yes	no
detailed basket	no	no	no	yes	yes
wet/dry cask	dry/wet	dry	dry	dry/wet	dry

PART II: Source Term

Attribute	Model Problem	
	V	VI
type of reactor	PWR	PWR
type of fuel	UO_2	MOX
axially varying source	yes	yes

General Remarks on Output Needs

In general we would suggest to reduce the results to be reported by the participants to a minimum, that is

- dose rate at the lateral cask surface and in a distance of 1, 2 and 10 m, measured from the surface and fin-tips, respectively
- dose rate at the lid surface and in a distance of 1, 2 and 10 m
- dose rate at the bottom surface and in a distance of 1, 2 and 10 m.

The dose rates should be reported as surface averaged values as well as a function of position if these data are available from the calculation. Furthermore the dose rate contributions of

- source neutrons,
- source gammas,
- capture gammas.

should be reported separately. All values should be given in SI-units, preferably µSv/h.

In addition to this surface averaged scalar fluxes as a function of energy should be reported for the surface of the side wall.

On an additional sheet of paper a short but complete description should be given in order to make the computational method used understandable. It should show

- a short overall description of the method
- codes
- cross section libraries
- energy, angular and spatial mesh if appropriate
- any additional assumptions made by the user.

During the comparison of results it may be necessary the participants provide additional informations in order to allow detecting the causes of possible discrepancies.

Model Problem 1:
Cylindrical Iron Cask with Homogeneous Source

Model problem 1 is a set of three simple basic problems which is intended to ensure a basis of consensus is obtained. Essentially it consists of a cylindrical cask with a homogeneous source of given strength. It is splitted in the following three subcases in order to get a basis check for a range of applications as wide as possible:

- case a: pure cast iron shield,
  no neutron absorbing or moderating materials
  source region dry

- case b: double layer shield containing 6 cm of polyethylene on
  the outer surface
  source region dry

- case c: pure cast iron shield like case a,
  source region containing 30 wt % of water

A full description of the model characteristics is provided on the following page in tables I/1 to I/3.

Table I/4 should be used as a form for presenting the summary results for the subcases a, b, and c together. Plots of the dose rates at the surface of lid and side wall vs. position should be given in fig. I/1 if these data are available (z = 0 refers to the cask cavity midplane). Fluxes should be reported graphically using the prespecified plot scale provided in fig. I/2 to I/4 for ease of comparison. If the graph exceeds the plot frame please multiply by a proper power of 10. The description of the computational methods as specified in page 7 should be provided on an additional sheet of paper.

Table I/1:

Characteristics of Model Problem I, case a

Dimensions: See table A3

Material Assignment:

region	material (for explanation see table A7)
source region	homogeneous source (dry)
side wall	cast iron
cask bottom	cast iron
cask lid	1.4313
cooling fins	none
neutron shield	none

Source Description:

	neutrons	gammas
source region	identical with cask cavity	
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1} *)
energy distribution	see table A8	
spatial distribution	constant in the whole source region zero otherwise	

*) read as $5.0 \cdot 10^{16}$

Table I/2:

Characteristics of Model Problem I, case b

Dimensions: See table A3

Material Assignment:

region	material (for explanation see table A7)
source region	homogeneous source (dry)
side wall	cast iron + 6 cm of polyethylene
cask bottom	cast iron
cask lid	1.4313
cooling fins	none
neutron shield	polyethylene, set within the outer diameter of 156 cm

Source Description:

	neutrons	gammas
source region	identical with cask cavity	
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1} *)
energy distribution	see table A8	
spatial distribution	constant in the whole source region zero otherwise	

*) read as $5.0 \cdot 10^{16}$

Table I/3:

Characteristics of Model Problem I, case c

Dimensions: See table A3

Material Assignment:

region	material (for explanation see table A7)
source region	homogeneous source (wet)
side wall	cast iron
cask bottom	cast iron
cask lid	1.4313
cooling fins	none
neutron shield	none

Source Description:

	neutrons	gammas
source region	identical with cask cavity	
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1} *)
energy distribution	see table A8	
spatial distribution	constant in the whole source region zero otherwise	

*) read as $5.0 \cdot 10^{16}$

Table I/4:
Summary Results of Model Problem 1

	surface averaged[1] dose rates (µSv/h)								
	n			γ (FP)			γ (capt.)		
	a	b	c	a	b	c	a	b	c
surface									
side wall	–	–	–	–	–	–	–	–	–
lid	–	–	–	–	–	–	–	–	–
bottom	–	–	–	–	–	–	–	–	–
1 m distance									
side wall	–	–	–	–	–	–	–	–	–
lid	–	–	–	–	–	–	–	–	–
bottom	–	–	–	–	–	–	–	–	–
2 m distance									
side wall	–	–	–	–	–	–	–	–	–
lid	–	–	–	–	–	–	–	–	–
bottom	–	–	–	–	–	–	–	–	–
10 m distance									
side wall	–	–	–	–	–	–	–	–	–
lid	–	–	–	–	–	–	–	–	–
bottom	–	–	–	–	–	–	–	–	–

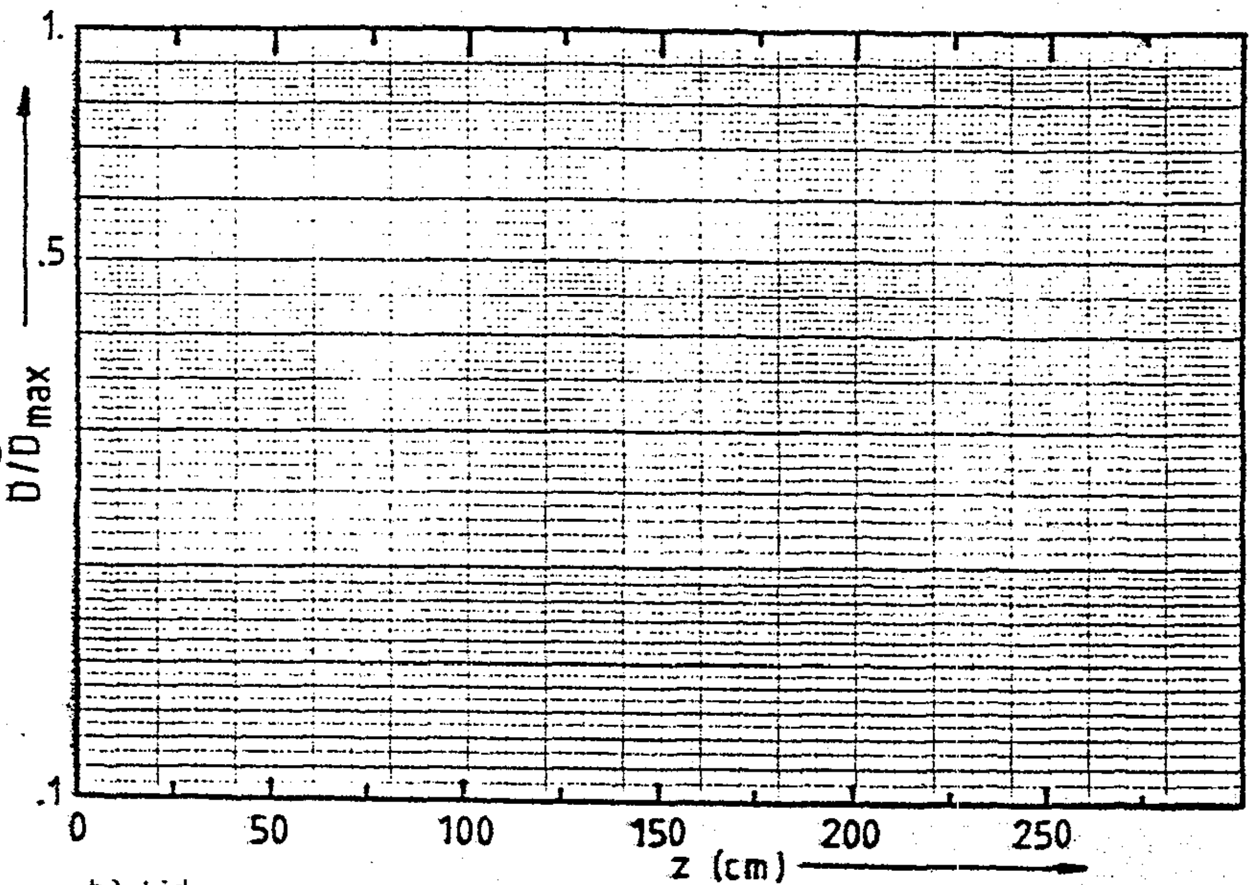
[1] averaging extends over
- the cavity height for the side wall dose rate
- the cavity cross sectional area for the top and bottom dose rate

Fig. I/1:

Model Problem 1: Plot of Dose Rate vs. Position

Full line: case a dashed line: case b dotted line: case c

a) Side Wall



b) Lid

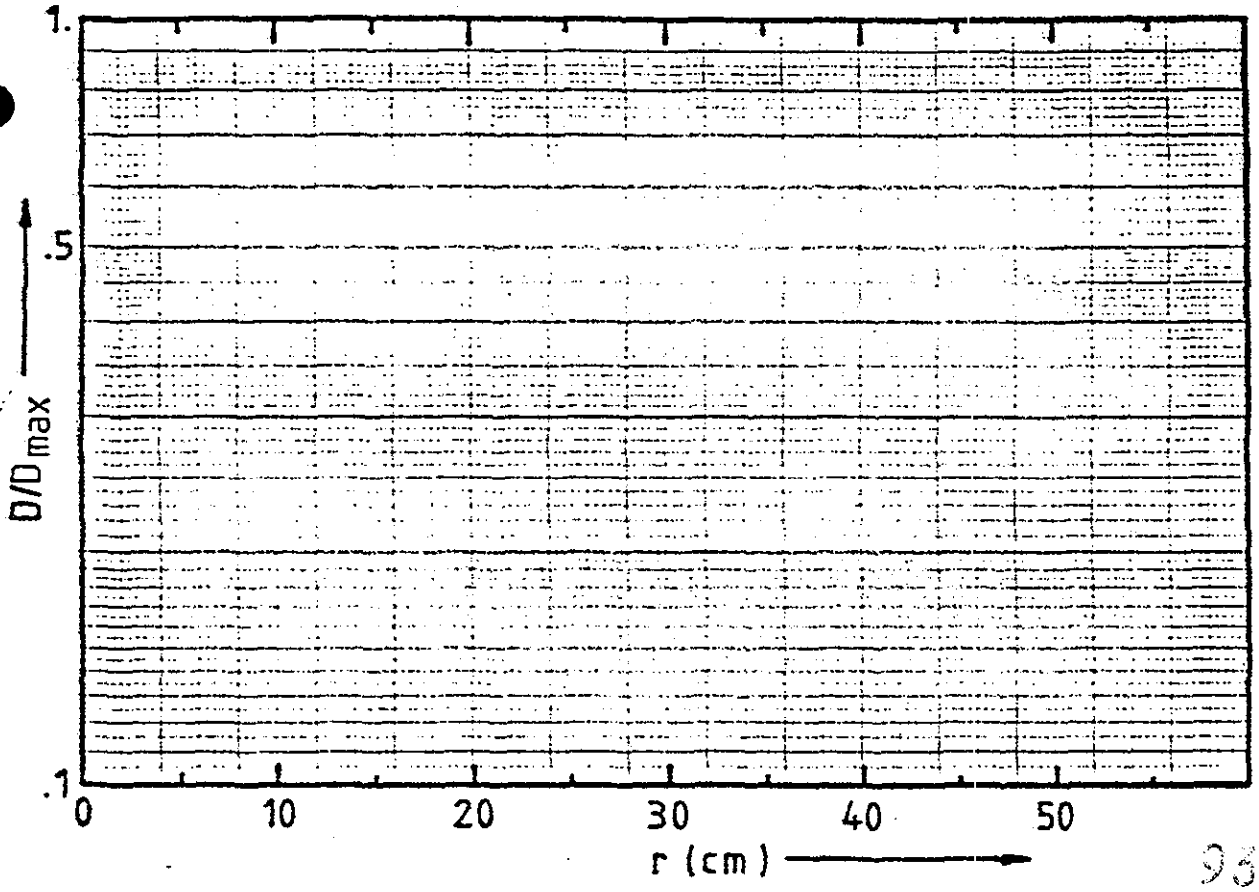


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Fig. 1/2:
Model problem 1, case a:
Plot of Surface Averaged Scalar Flux at Side Wall Surface vs. Energy
Averaging extends over the cask cavity height

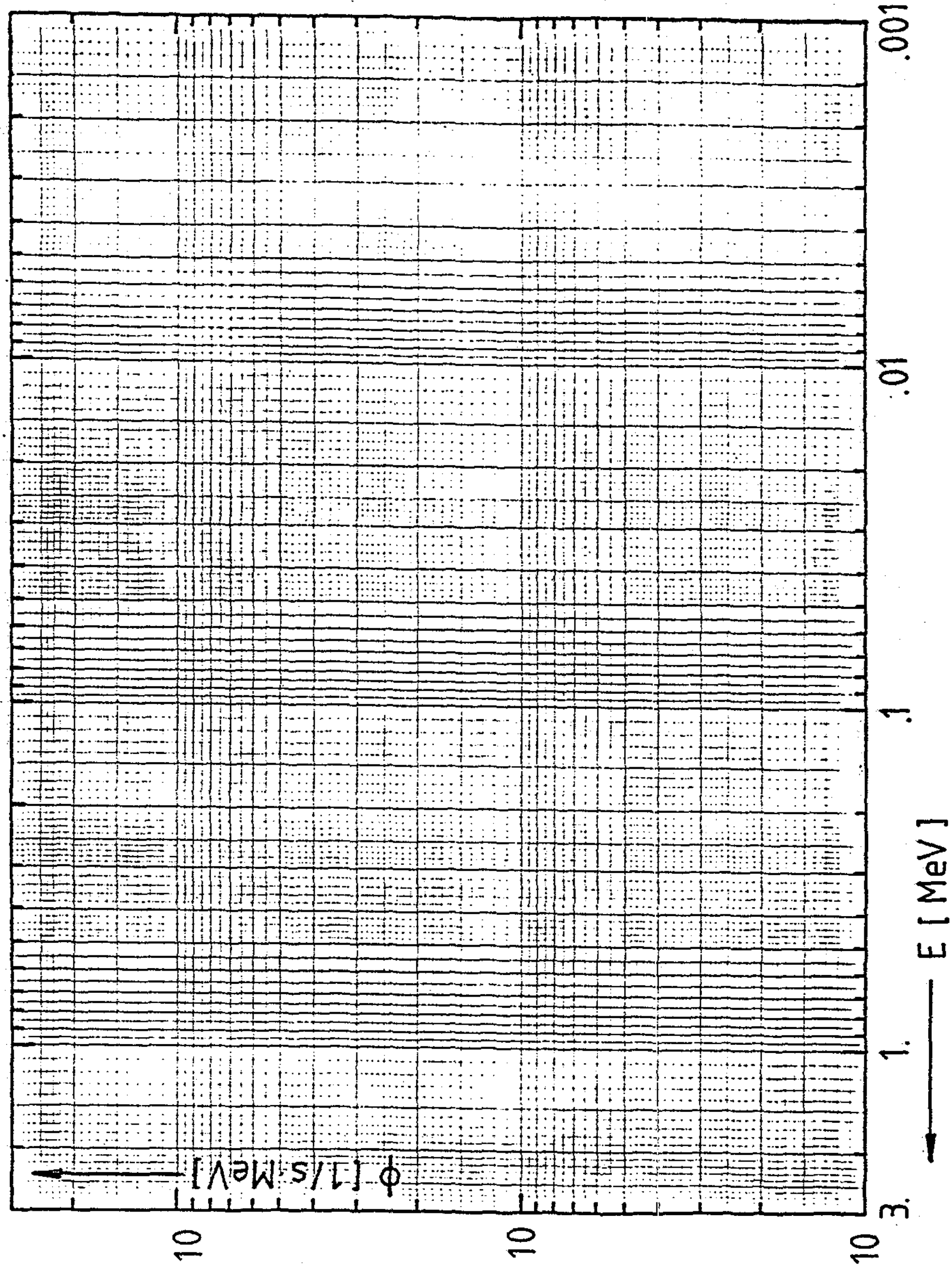

Fig. 1/3:

Model problem 1, case b:

Plot of Surface Averaged Scalar Flux at Side Wall Surface vs. Energy
Averaging extends over the cask cavity height

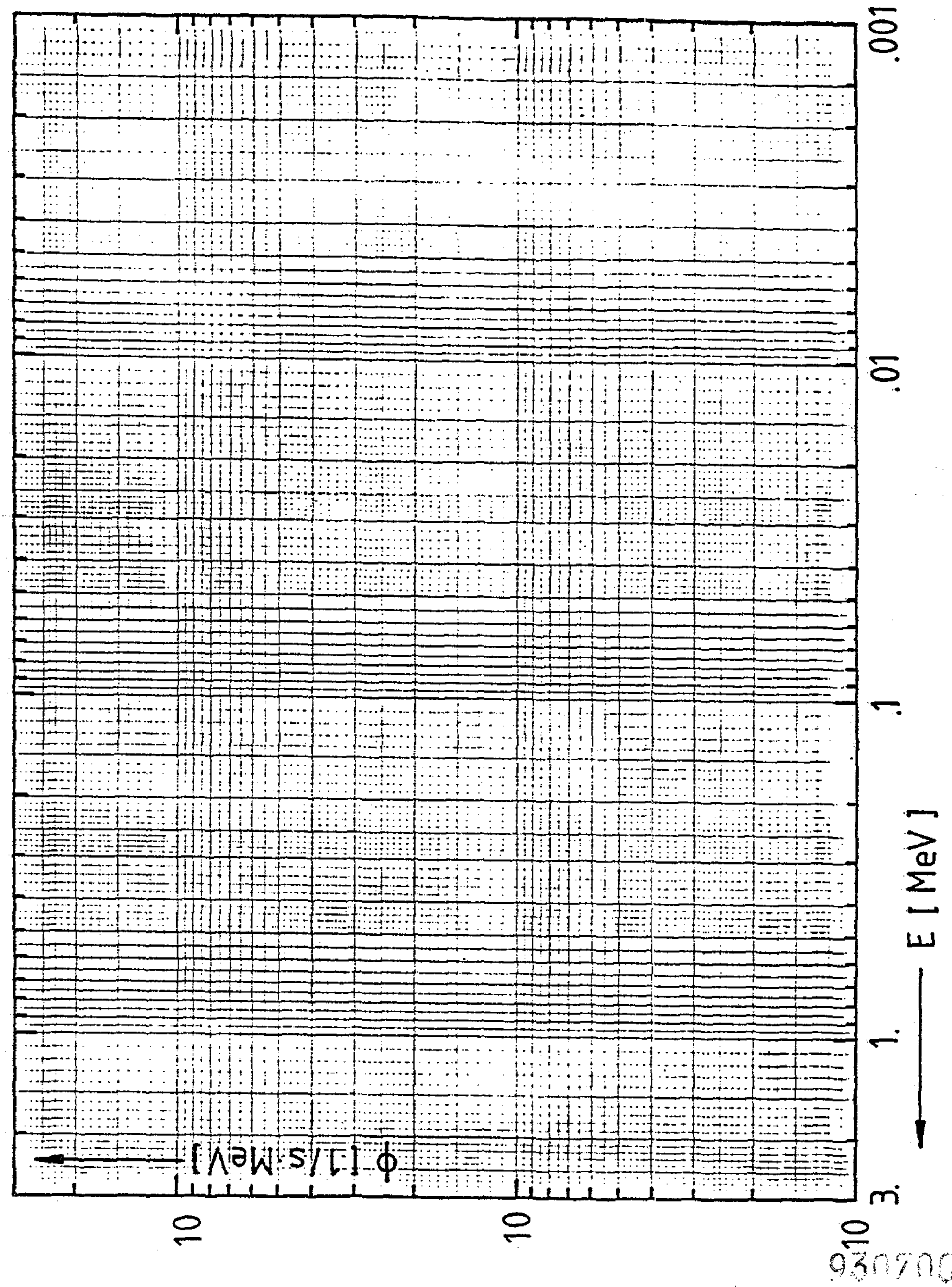

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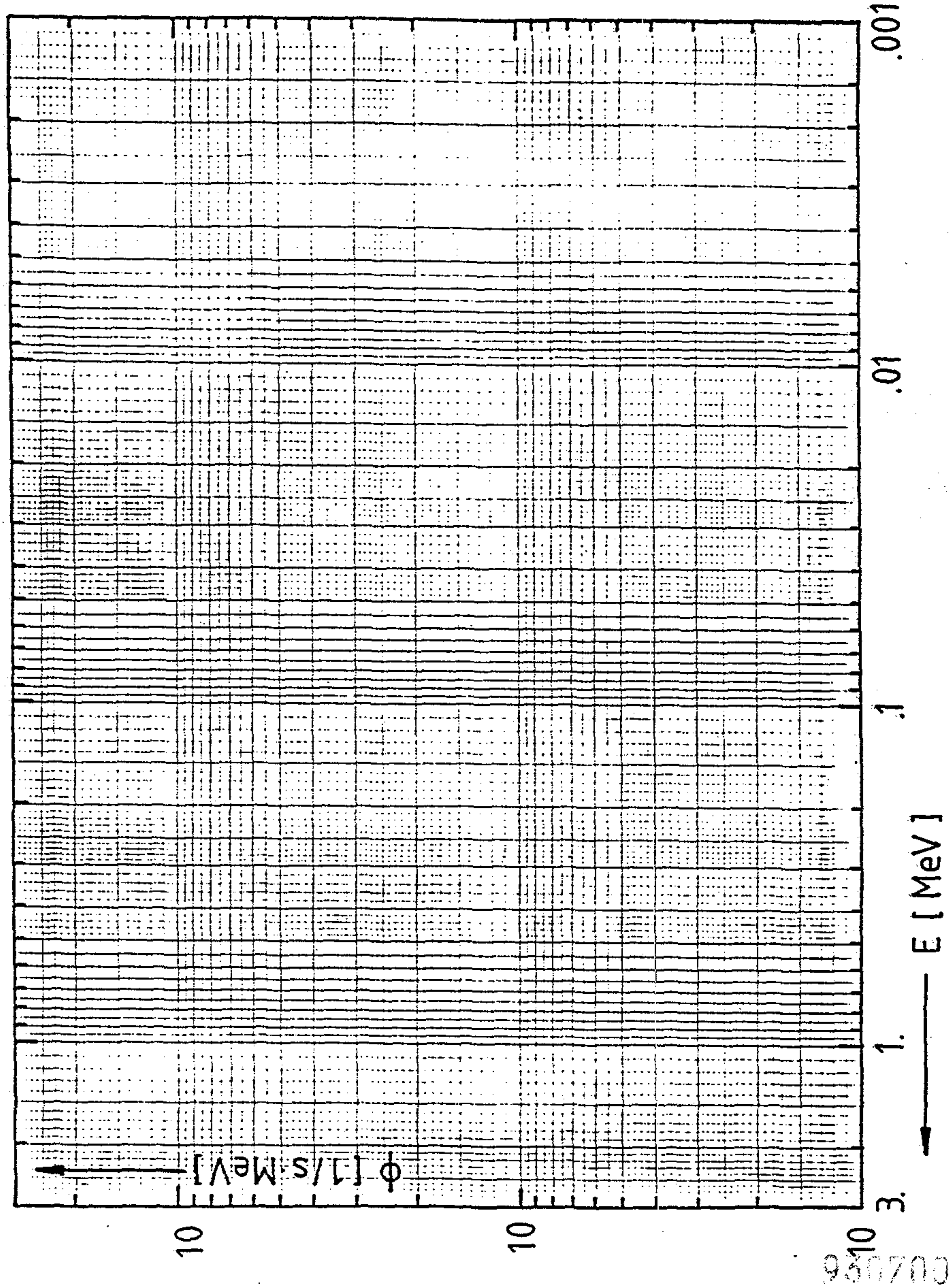
Fig. I/4:

Model problem 1, case c:

Plot of Surface Averaged Scalar Flux at Side Wall Surface vs. Energy
Averaging extends over the cask cavity height



Model Problem 2:

As model problem 1a with realistic neutron moderator designs added

Model problem 2 consists of just the same cask as used in model problem 1a with additional neutron moderator material added. Because there are at least two completely different philosophies on the best way to reduce the neutron dose rate we suggest to split model problem 2 as follows

- case a: Cylindrical polyethylene rods located in circular holes inside the cask wall serves as neutron moderator

- case b: Stripes of epoxi cast resin are fixed on the outer surface of the side wall, set within the outer diameter of 156 cm.

For the sake of simplicity we assume the moderating material extends over the full height of the cask. A description of the two subcases of modell problem 2 is provided on the following pages in tables II/1 and II/2, respectively. Since the addition of neutron moderating material should not affect the top and bottom dose rate to a great extent these results can be omitted.

Table II/3 should be used as a form for presenting the summary results for cases a and b together. A plot of the azimuthally averaged dose rate at the surface of the side wall vs. position should be given on fig. II/1 if these data are available (z = 0 refers to the cask cavity midplane). Fluxes should be reported graphically using the prespecified plot scale provided in fig. II/4 for ease of comparison. In the graphical presentation case a and b should be distinguished using a dashed and full line, respectively. The description of the computational methods used should report only the changes made relative to model problem 1 a.

Table II/1:

Characteristics of Model Problem II/case a

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
source region	homogeneous source (dry)
side wall	1.4541
cask bottom	1.4541
cask lid	1.4313
cooling fins	none
neutron shield	polyethylene location as shown in fig. II/1

Source Description:

	neutrons	gammas
source region	same as cask cavity	
total source strength	1.0 E9 s^{-1} (subcritical neutron multplication not to be taken into account)	5.0 E16 s^{-1}
energy distribution	see table A8	
spatial distribution	constant in the whole source region zero otherwise	

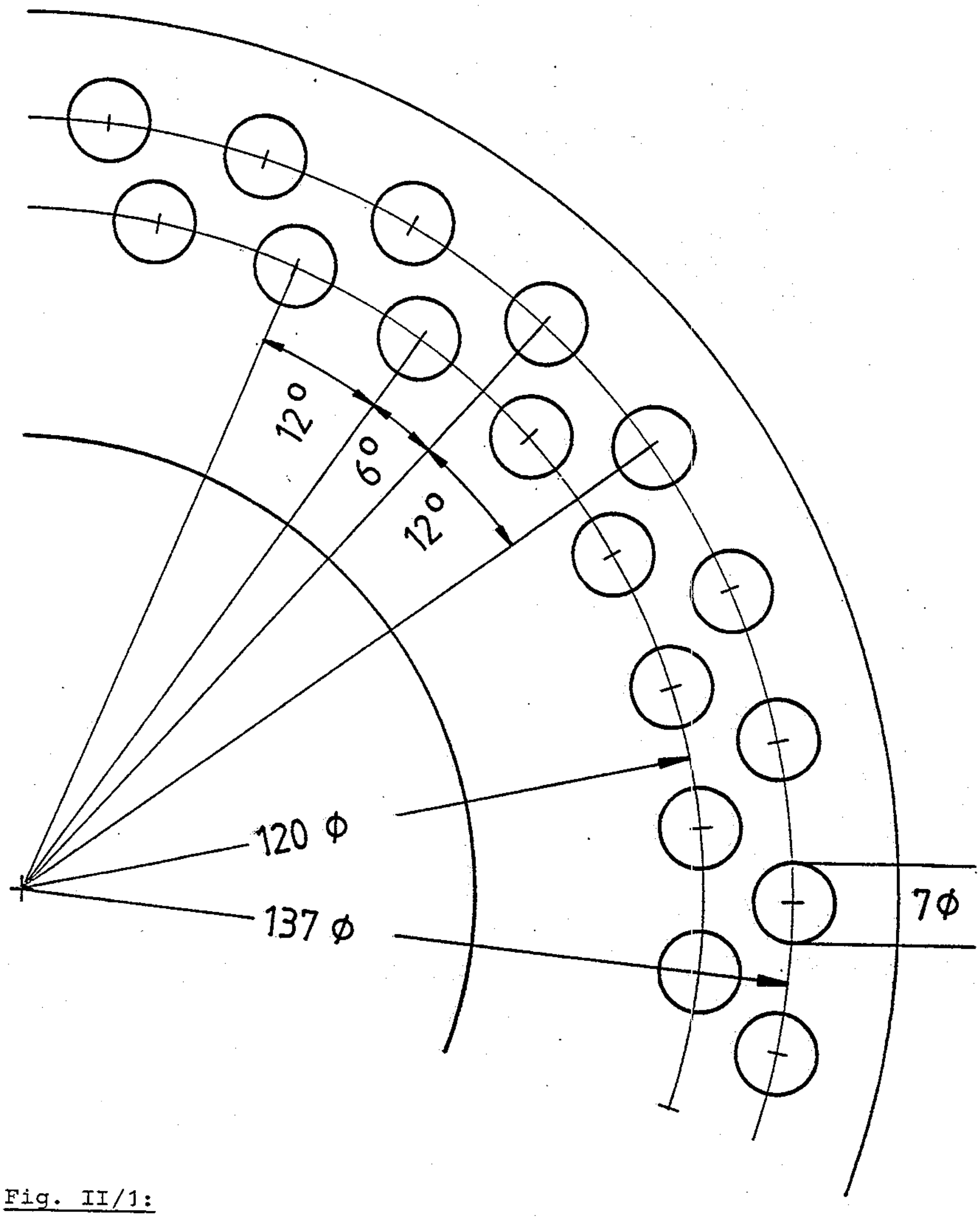


Fig. II/1:
Characteristics of Additional Neutron Shield
Case A: Polyethylene within the Cask Wall
(Dimensions given in cm)

Table II/2:

Characteristics of Model Problem II/case b

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
source region	homogeneous source (dry)
side wall	cast iron
cask bottom	cast iron
cask lid	1.4313
cooling fins	none
neutron shield	epoxi cast resin, set within the outer diamter of 156 cm location as shown in fig. II/2

Source Description:

	neutrons	gammas
source region	identical with cask cavity	
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1}
energy distribution	see table A8	
spatial distribution	constant in the whole source region zero otherwise	

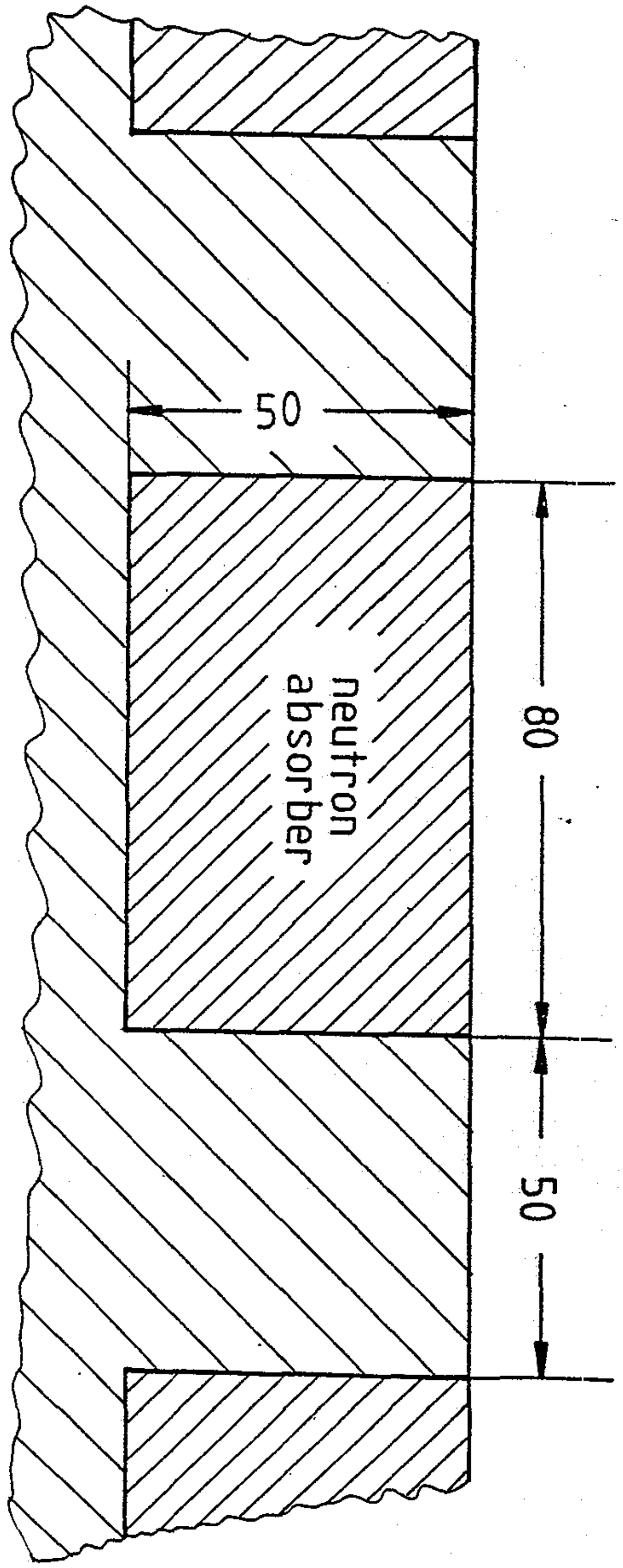


Fig. II/2:
Characteristics of Additional Neutron Shield
Case B: Epoxi Cast Resin at Outer Surface, Set within the
Outer Diameter of 156 cm
(Dimensions given in mm)

Table II/3:
Summary Results of Model Problem 2

	surface averaged[1] dose rates (μSv/h)					
	n		γ (FP)		γ (capt.)	
case	a	b	a	b	a	b
surface						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–
1 m distance						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–
2 m distance						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–
10 m distance						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–

[1] averaging extends over
- the cavity height for the side wall dose rate
- the cavity cross sectional area for the top and bottom dose rate

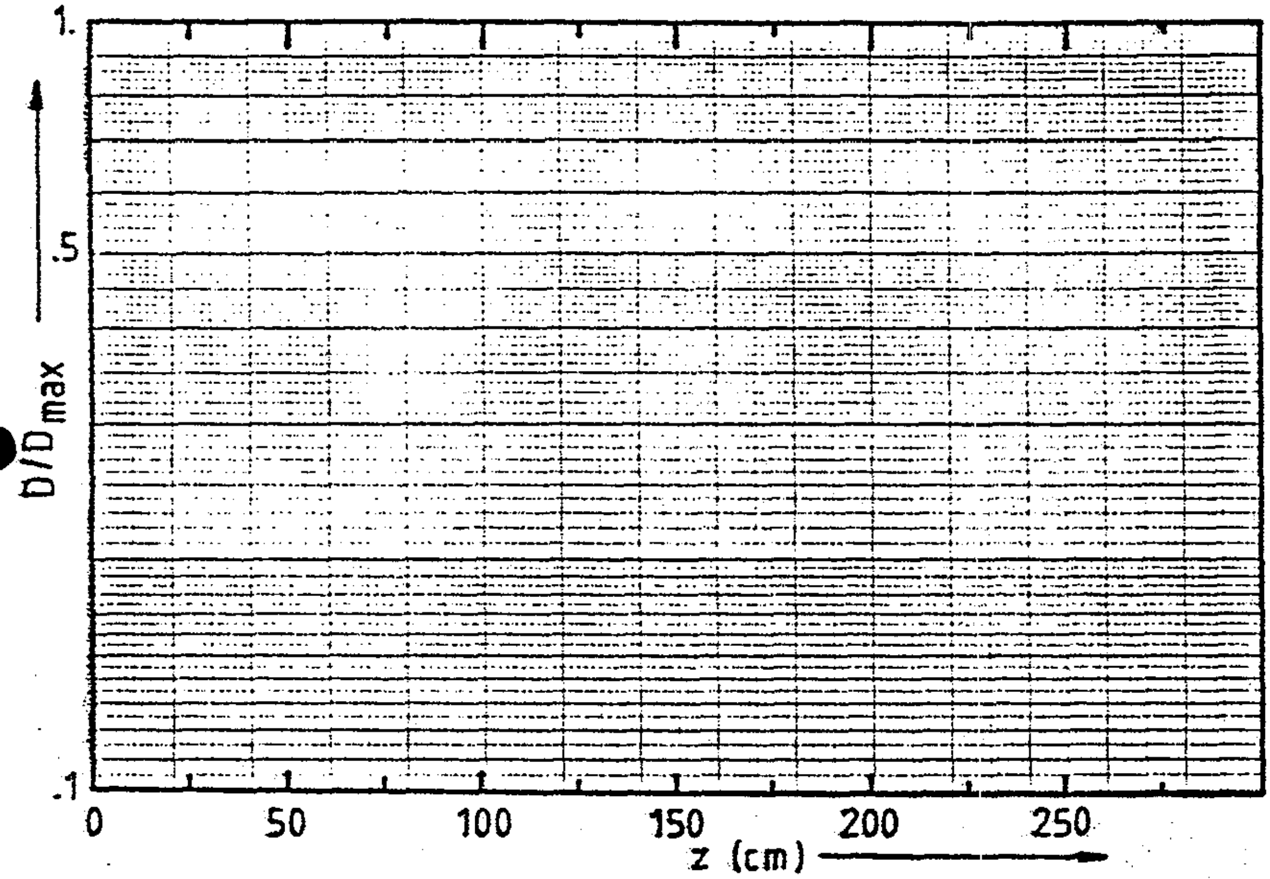
[2] results for top and bottom dose rates can be omitted

<u>Fig. II/3:</u>

<u>Model Problem 2:</u> Plot of Azimuthally Averaged Dose Rate vs. Position

full line: case a dashed line: case b

a) Side Wall



b) Lid (can be omitted)

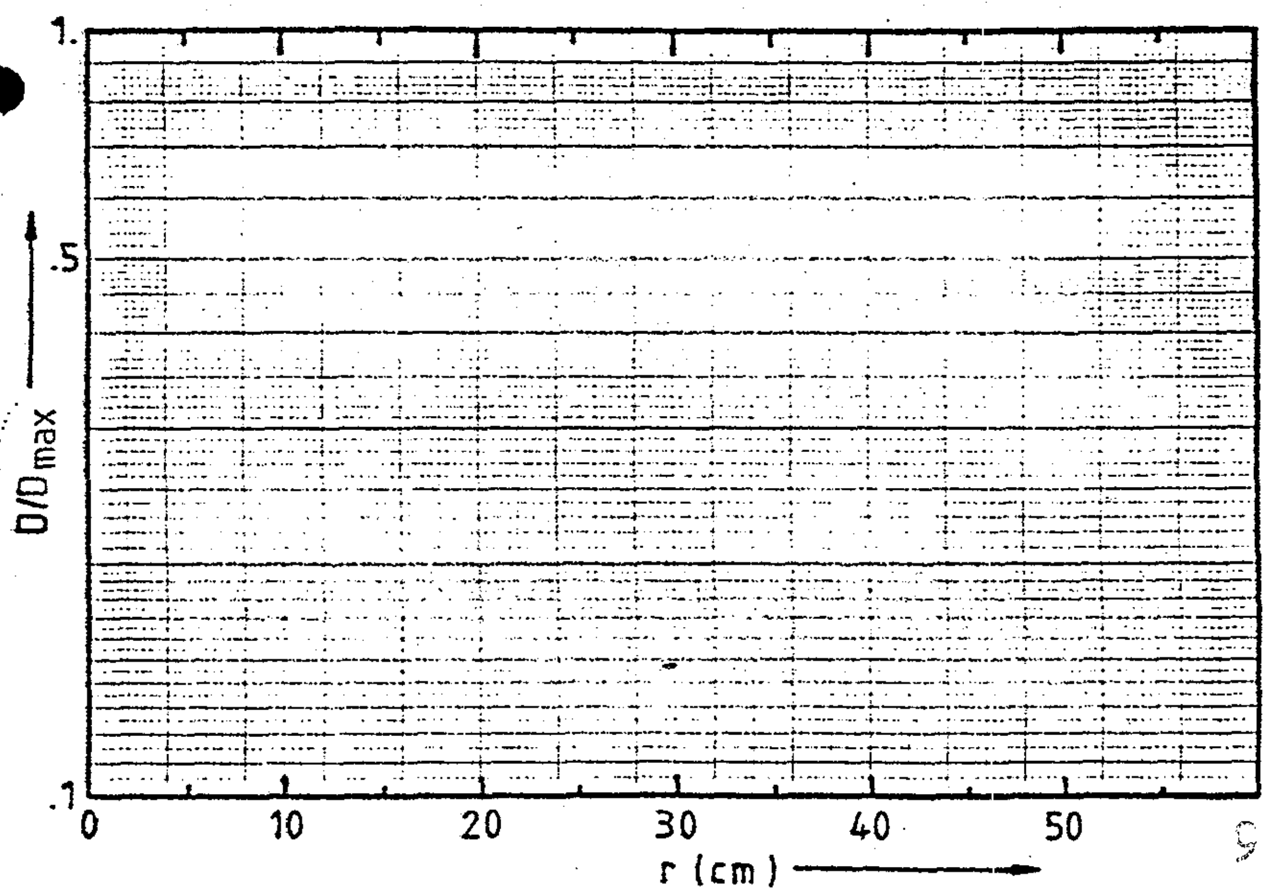
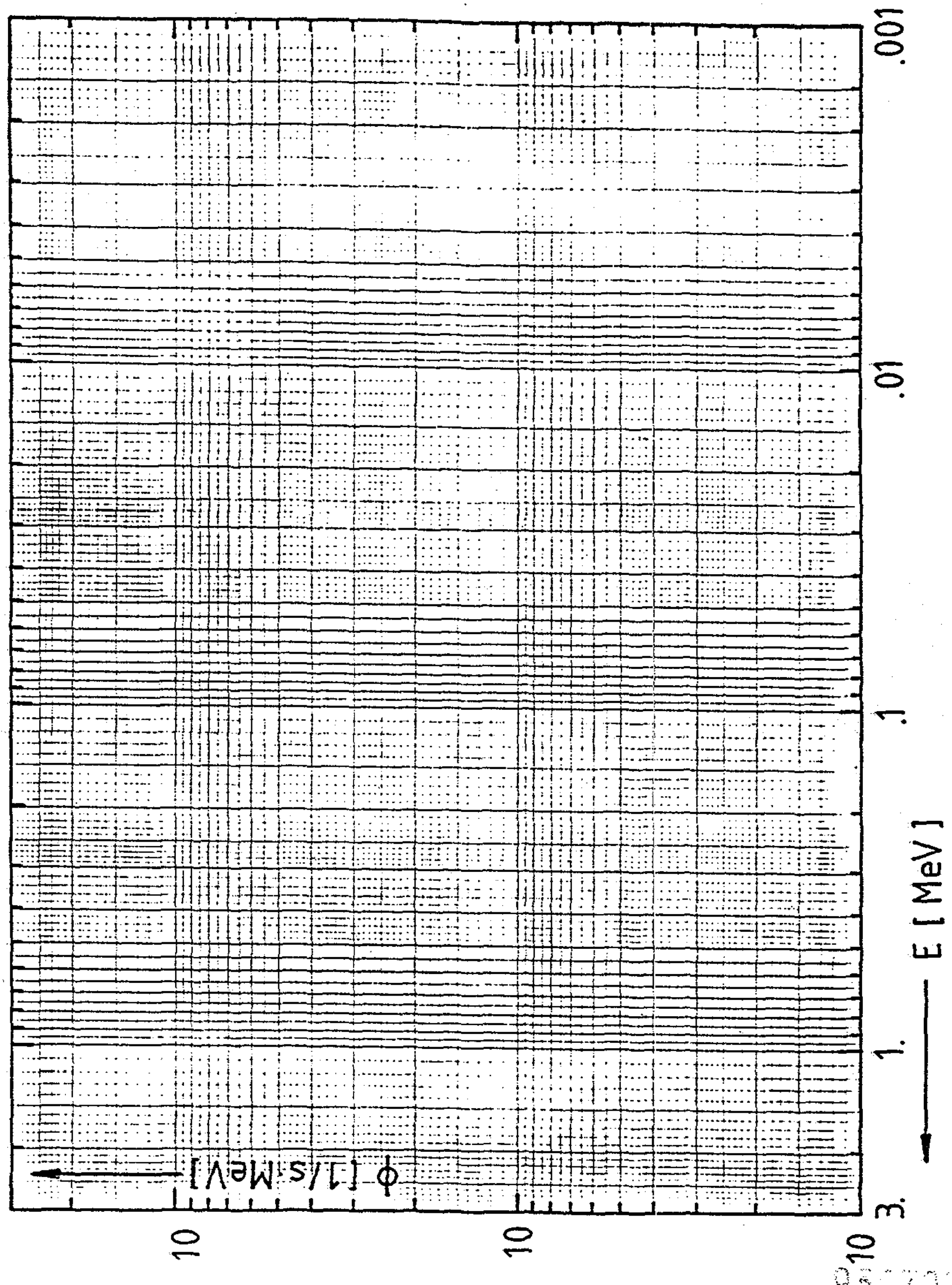


Fig. II/4:

Model problem 2:

Plot of Surface Averaged Scalar Flux at Side Wall Surface vs. Energy (averaging extends over the total cask cavity height)



Model Problem 3:

As model problem 2 with cooling fins added

Model problem 3 is identical with model problem 2 but cooling fins are added. Due to the different constructions of neutron moderator model problem 3 should also be splitted into two subcases:

- Case 3a: Neutron moderating material inside the cask wall.

- Case 3b: Epoxi cast resin serving as neutron absorber located in between the fins at the outer surface of the side wall.

A full description of the model characteristics is provided on the following pages in table III/1 and III/2, respectively.

Because the changes made relative to model problem 1 a should not affect the top and bottom results substantially these data can be omitted. Table III/3 should be used as a form for presenting the summary results for the cases a and b together. A plot of the azimuthally averaged dose rate at the side wall surface vs. axial position for a single fin interval in the cask mid plane should be given on fig. III/3 if these data are available (z = 0 refers to the cask cavity midplane). Fluxes should be reported graphically using the prespecified plot scale provided in fig. III/4 for ease of comparison. The description of the computational methods should be reduced to a listing of the changes relative to model problem 2.

Table III/1:

Characteristics of Model Problem III/case a

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
source region	homogeneous source (dry)
side wall	1.4541
cask bottom	1.4541
cask lid	1.4313
cooling fins	see fig. III/1 total number: 35 mid plane of the first fin on the top level of cask cavity
neutron shield	polyethylene same location as in problem 3 a

Source Description:

	neutrons	gammas
source region	identical with cask cavity	
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1}
energy distribution	see table A8	
spatial distribution	constant in the whole source region zero otherwise	

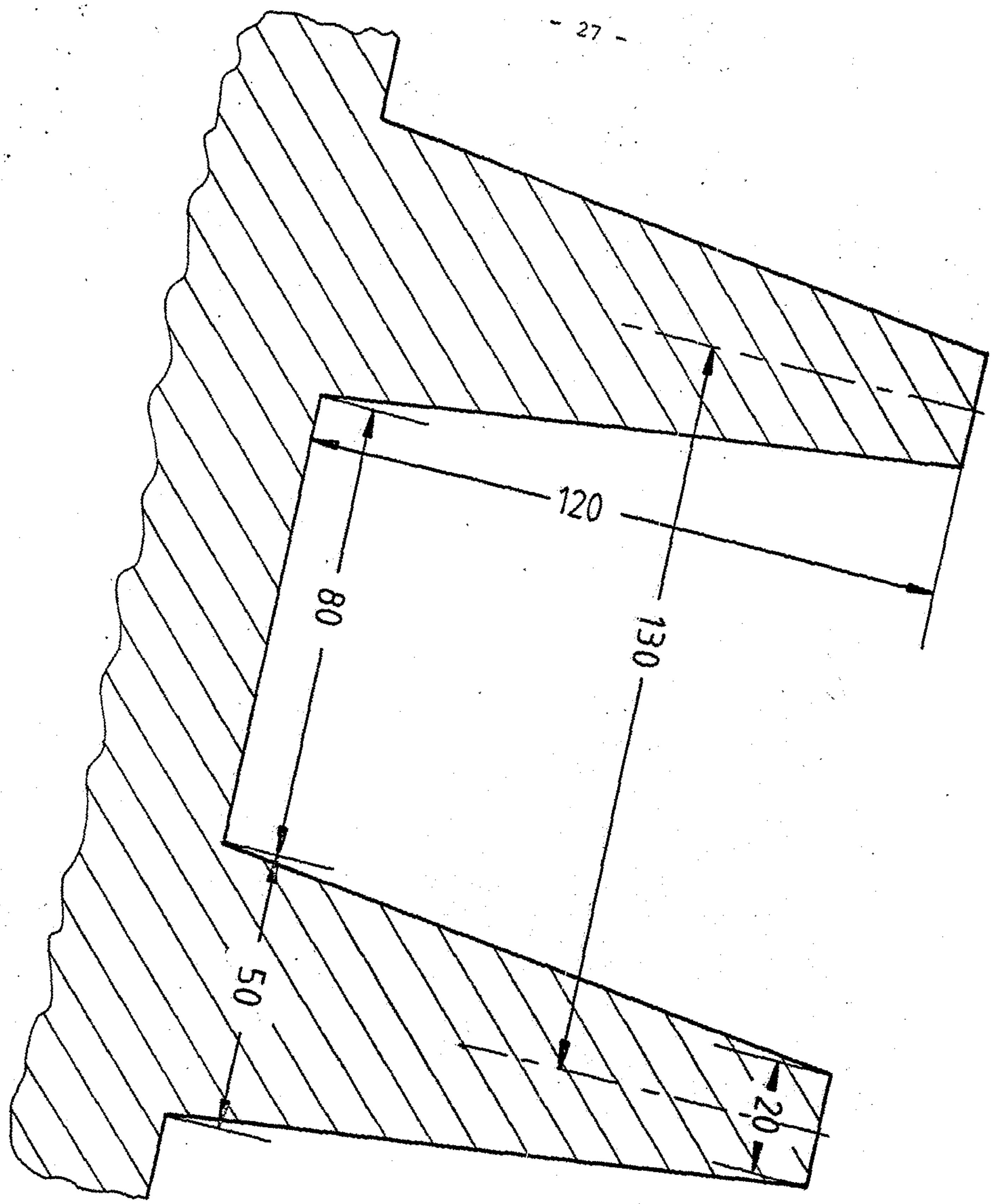


Fig. III/1:
Characteristics of Cooling Fins
Case A: Neutron Shield within the Cask Wall, the Midplane of the First Fin is Located on the Level of the Cask Cavity Top
(Dimensions given in mm)

Table III/2:

Characteristics of Model Problem III/case b

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
source region	homogeneous source (dry)
side wall	cast iron
cask bottom	cast iron
cask lid	1.4313
cooling fins	see fig. III/2 total number: 35 mid plane of the first fin on the top level of cask cavity
neutron shield	epoxi cast resin same location as in problem 2 b

Source Description:

	neutrons	gammas
source region	identical with cask cavity	
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1}
energy distribution	see table A8	
spatial distribution	constant in the whole source region zero otherwise	

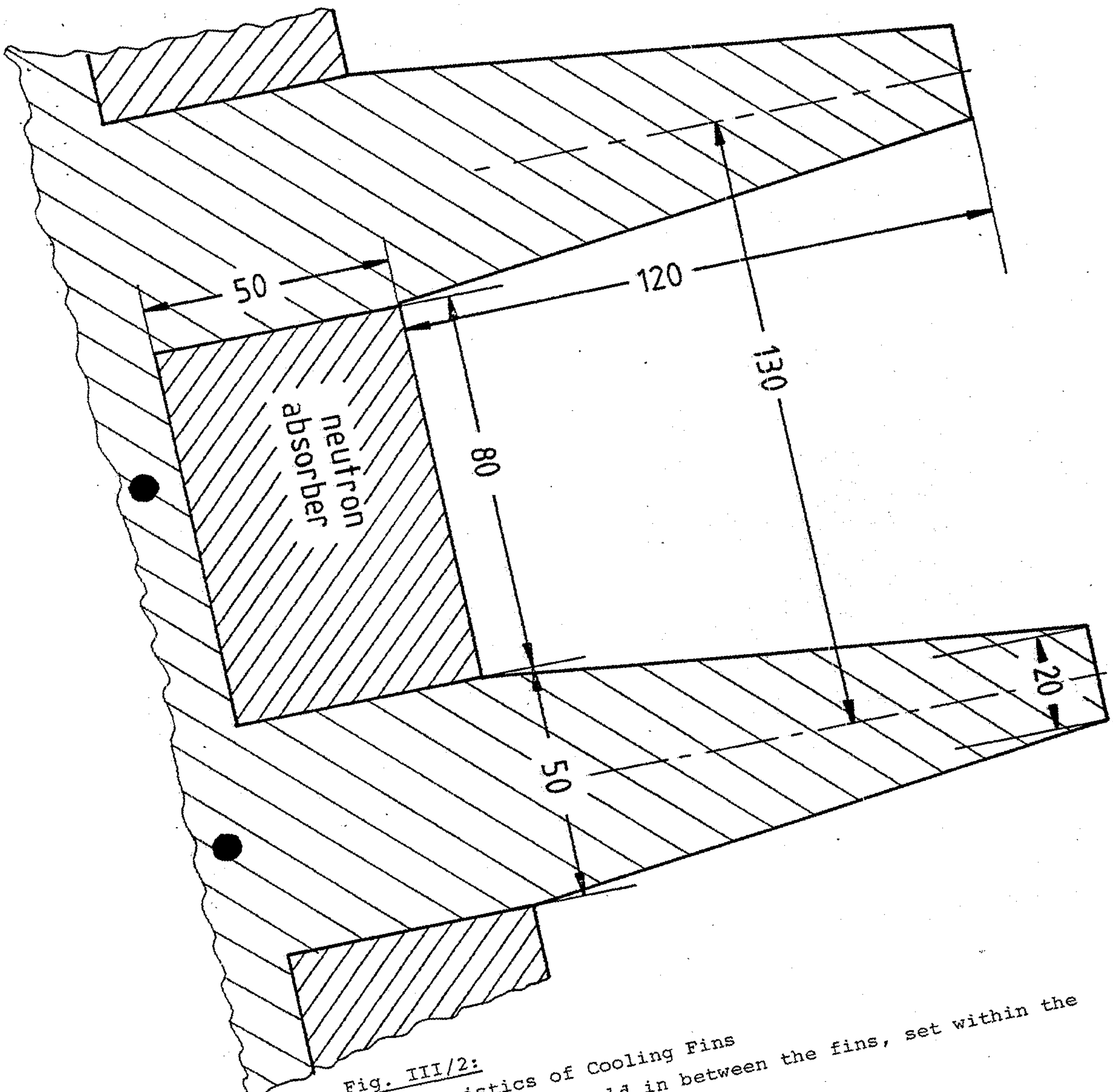


Fig. III/2:
Characteristics of Cooling Fins
Case B: Neutron shield in between the fins, set within the
outer diameter of 156 cm
(Dimensions given in mm)

Table III/3:

Summary Results of Model Problem 3

	surface averaged[1] dose rates (μSv/h)					
	n		γ (FP)		γ (capt.)	
case	a	b	a	b	a	b
surface						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–
1 m distance[3]						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–
2 m distance[3]						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–
10 m distance[3]						
side wall	–	–	–	–	–	–
(lid)[2]	–	–	–	–	–	–
(bottom)[2]	–	–	–	–	–	–

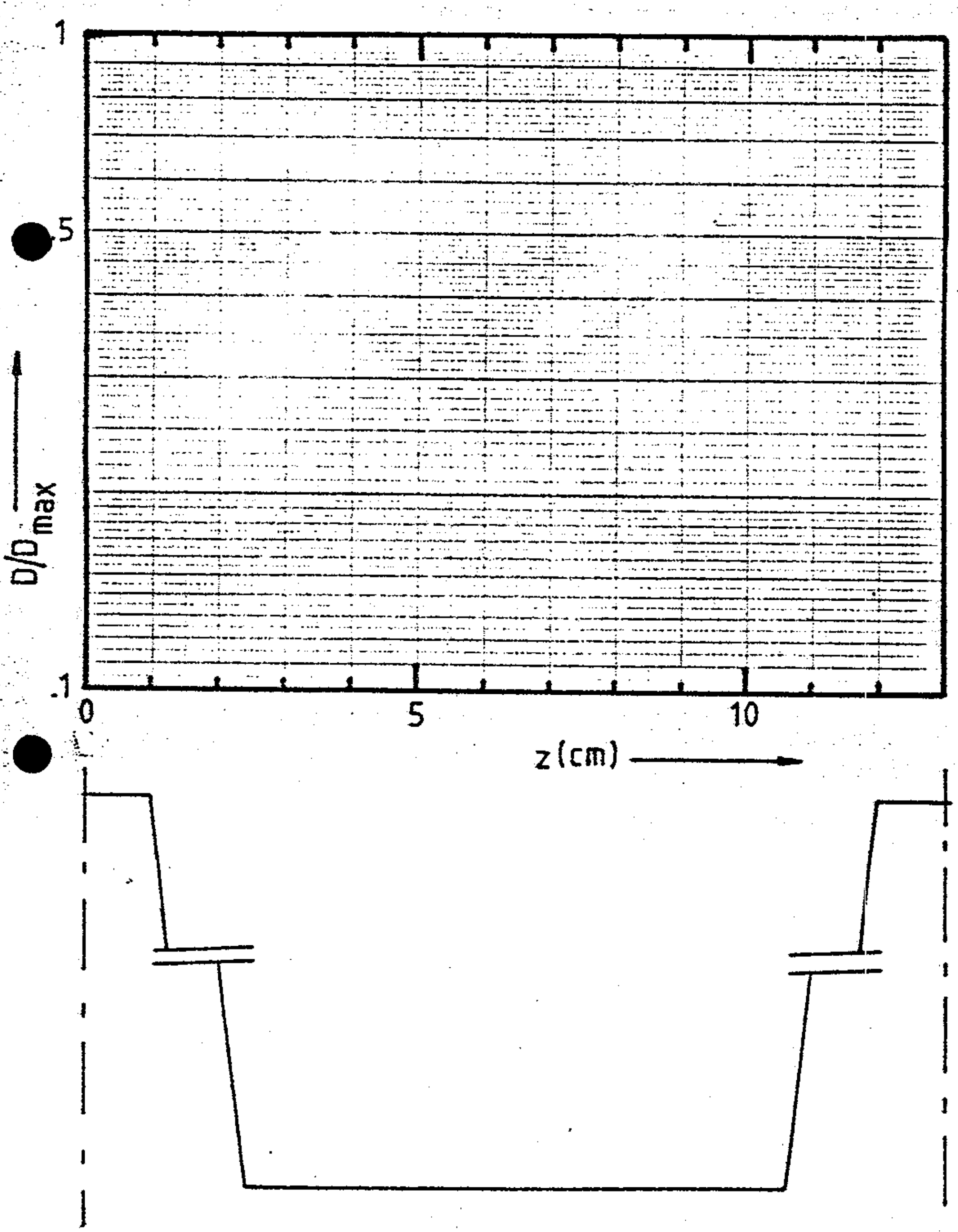
[1] averaging extends over
- the cavity height for the side wall dose rate
- the cavity cross sectional area for the top and bottom dose rate

[2] results for top and bottom dose rates can be omitted
[3] distances are measured from the fin-tips

Fig. III/3:

Model Problem 3: Plot of Azimuthally Averaged Dose Rate vs. Position for a single fin in the cask mid plane

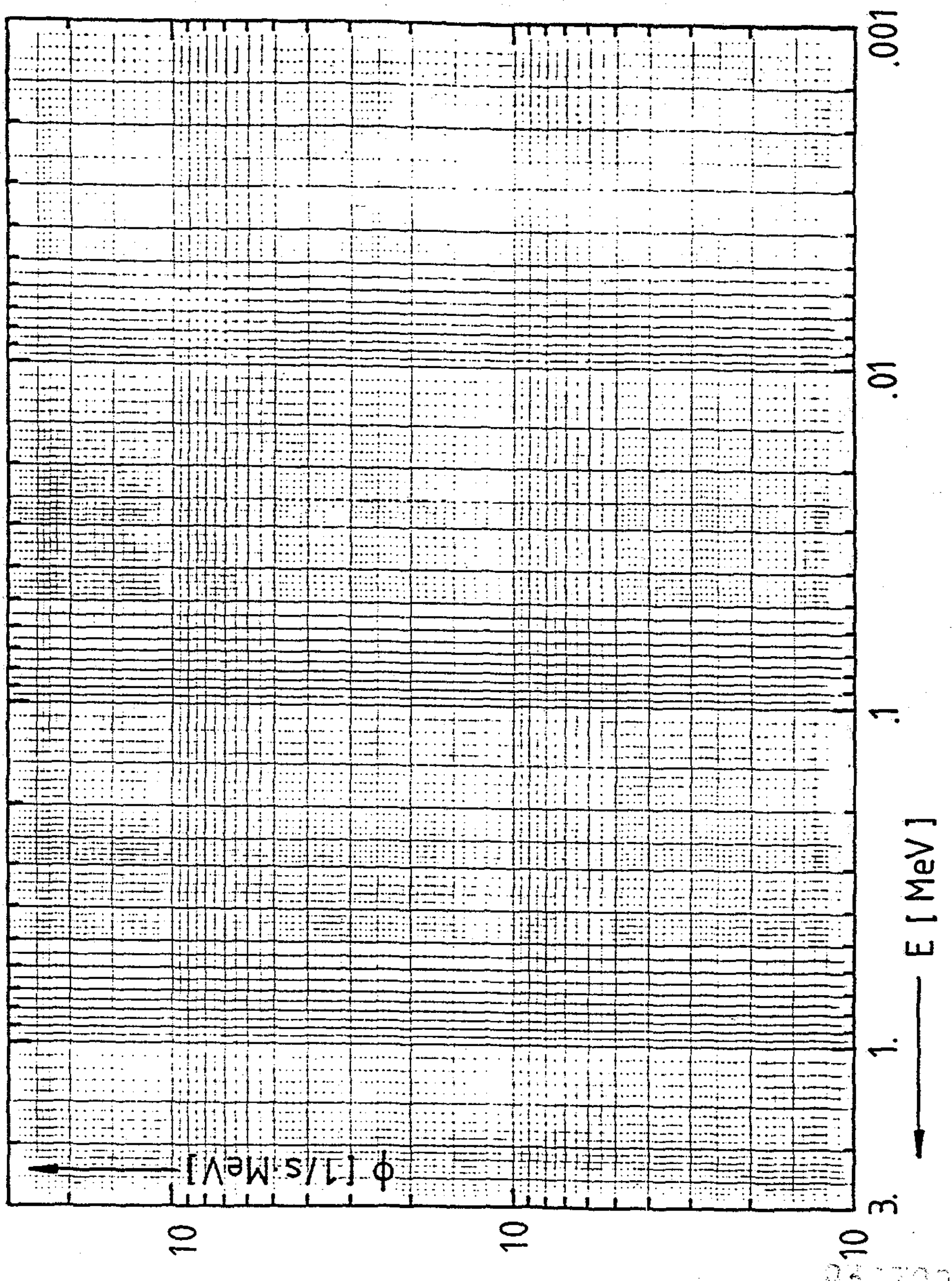
a) Side Wall



<u>Fig. III/4:</u>

<u>Model problem 3:</u>

Plot of Surface Averaged Scalar Flux at Side Wall Surface vs. Energy (averaging extends over the total cavity height on the level of fin-tips)



Model Problem 4:
Model problem 3 with detailed modelling of basket and fuel assemblies

Model problem 4 is identical with model problem 3 but the homogeneous volumetric source is replaced by a detailed stainless steel basket containing 5 fuel assemblies. A drawing of the basket and the model assembly can be seen in fig. IV/1 and IV/2, respectively. A list of the geometric data of basket and assembly is given in table A4 and A5, respectively. In order to reduce the number of calculations to be done we don't distinguish two different neutron moderation techniques. But due to the existence of dry and wet casks model problem 4 consists of two subcase again:

- case a: dry basket
- case b: wet basket

A full description of their characteristics is provided on the following page in tables IV/1 and IV/2, respectively. Due to the fact that exact descriptions of the top and bottom fittings for a particular fuel assembly usually are not available we only give their total length and masses in tables A4 and A5. In order to ensure the compatibility of the different results we suggest to model the fittings as being homogeneous over their length. This yields a steel density of 1.01 g/cm^3 for both of them. Nevertheless we suppose that for the assessment of the lid and bottom dose rate somewhat like a sensitivity analysis with respect to the modelling of fittings should be performed. So each participant is urged to estimate the influence of different modelling on the top and bottom results.

Table IV/3 should be used as a form for presenting the summary results for cases a and b together. A plot of the azimuthally averaged dose rate at the surface of lid and side wall vs. position should be given on fig. IV/3 and IV/4, respectively, if these data are available ($z = 0$ refers to the cask cavity midplane). If there are significant differences between upper and lower part of the cask both profiles should be reported using the same plot frame. Fluxes should be reported graphically using the prespecified plot scale provided in fig. IV/5 and IV/6 for ease of comparison. The description of the computational methods should contain the geometrical modelling of the cask cavity region.

Table IV/1:

Characteristics of Model Problem IV, case a

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
fuel region	detailed basket with 5 fuel assy (see fig. IV/1 and IV/2) dry
top fitting	1.4541, ρ = 1.01 g/cm^3
bottom fitting	1.4541, ρ = 1.01 g/cm^3
side wall	1.4541
cask bottom	1.4541
cask lid	1.4313
cooling fins	see fig. III/1
neutron shield	polyethylene location as shown in fig. II/1

Source Description:

	neutrons	gammas
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1}
energy distribution	see table A8	
spatial distribution	constant in the active zone of fuel rod zero otherwise	

Table IV/2:

Characteristics of Model Problem IV, case b

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
fuel region	detailed basket with 5 fuel assy (see fig. IV/1 and IV/2) voids totally filled with water (T = 20 °C, ρ = 1.0 g/cm³
top fitting	1.4541, ρ = 1.01 g/cm³
bottom fitting	1.4541, ρ = 1.01 g/cm³
side wall	1.4541
cask bottom	1.4541
cask lid	1.4313
cooling fins	see fig. III/1
neutron shield	polyethylene location as shown in fig. II/1

Source Description:

	neutrons	gammas
total source strength	1.0 E9 s^{-1} (subcritical neutron multiplication not to be taken into account)	5.0 E16 s^{-1}
energy distribution	see table A8	
spatial distribution	constant in the active zone of fuel rod zero otherwise	

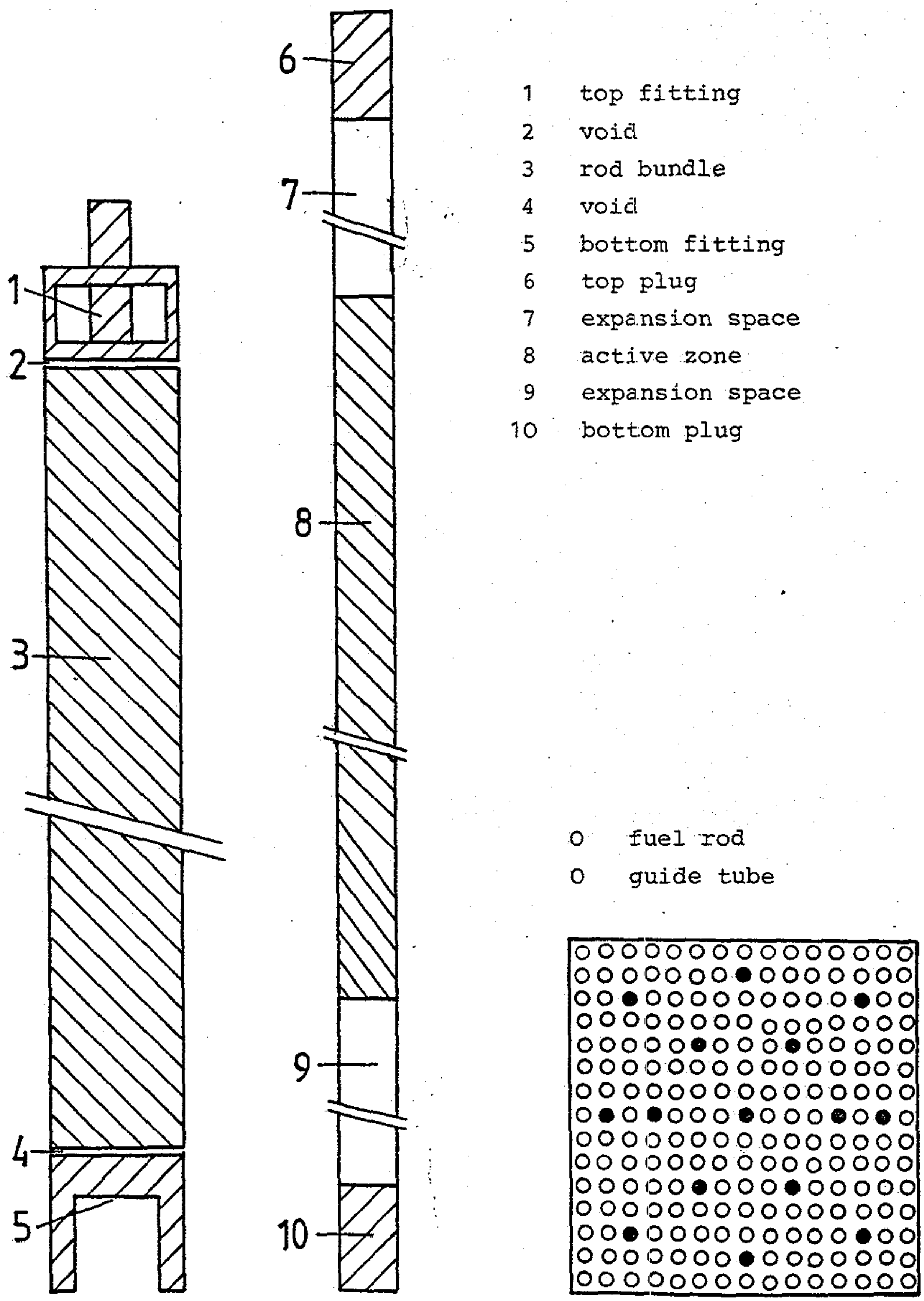


Fig. IV/1:
Cross Sectional View of Assembly and Single Rod
Arrangement of Fuel Rods and Guide Tubes

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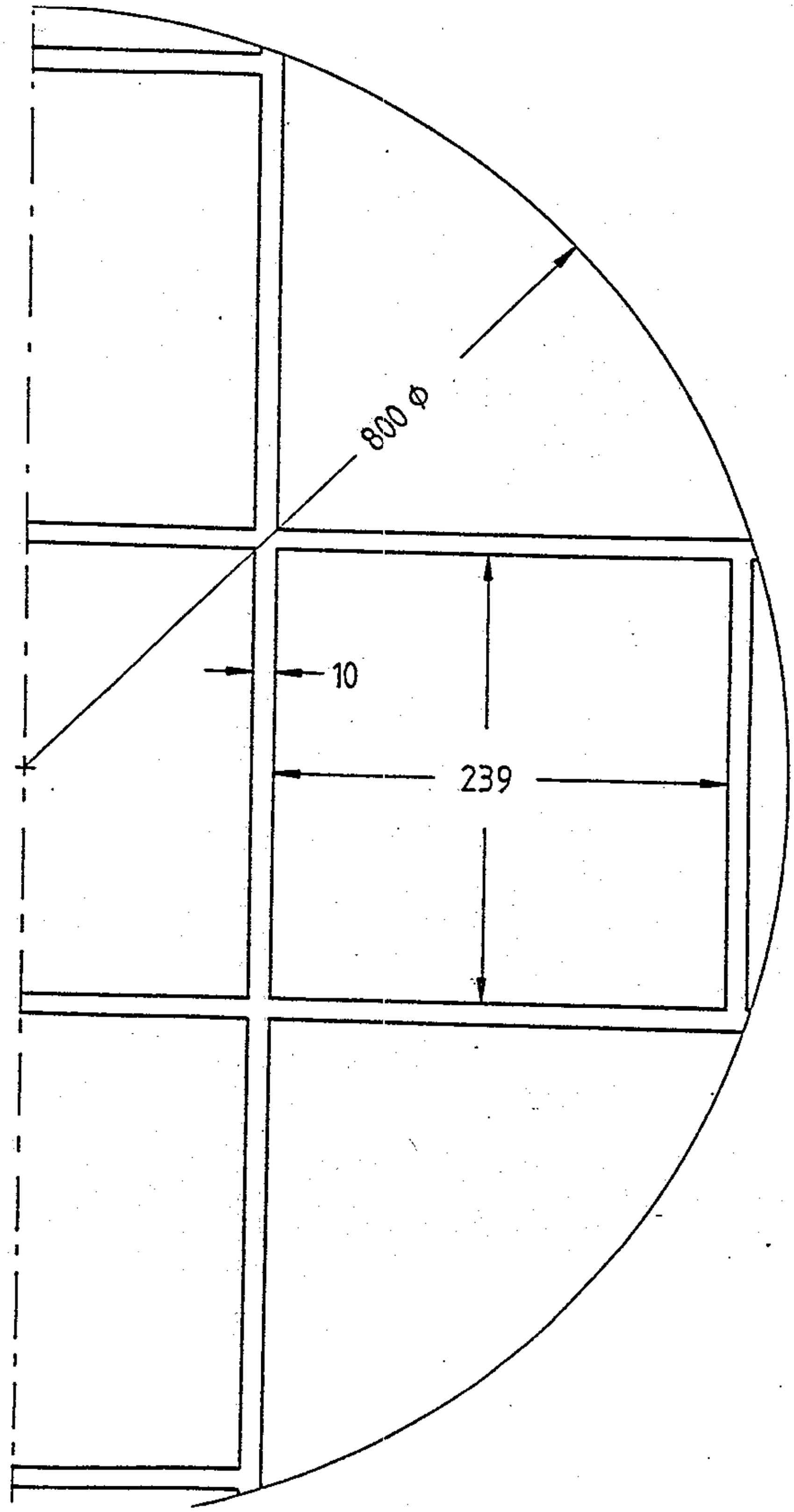


<u>Fig. IV/2:</u>
Construction of Basket
(Dimensions given in mm)

Table IV/3:

Summary Results of Model Problem 4, cases a and b

	surface averaged[1] dose rates (µSv/h)					
	n		γ (FP)		γ (capt.)	
case	a	b	a	b	a	b
surface[2]						
side wall	–	–	–	–	–	–
lid	–	–	–	–	–	–
bottom	–	–	–	–	–	–
1 m distance[3]						
side wall	–	–	–	–	–	–
lid	–	–	–	–	–	–
bottom	–	–	–	–	–	–
2 m distance[3]						
side wall	–	–	–	–	–	–
lid	–	–	–	–	–	–
bottom	–	–	–	–	–	–
10 m distance[3]						
side wall	–	–	–	–	–	–
lid	–	–	–	–	–	–
bottom	–	–	–	–	–	–

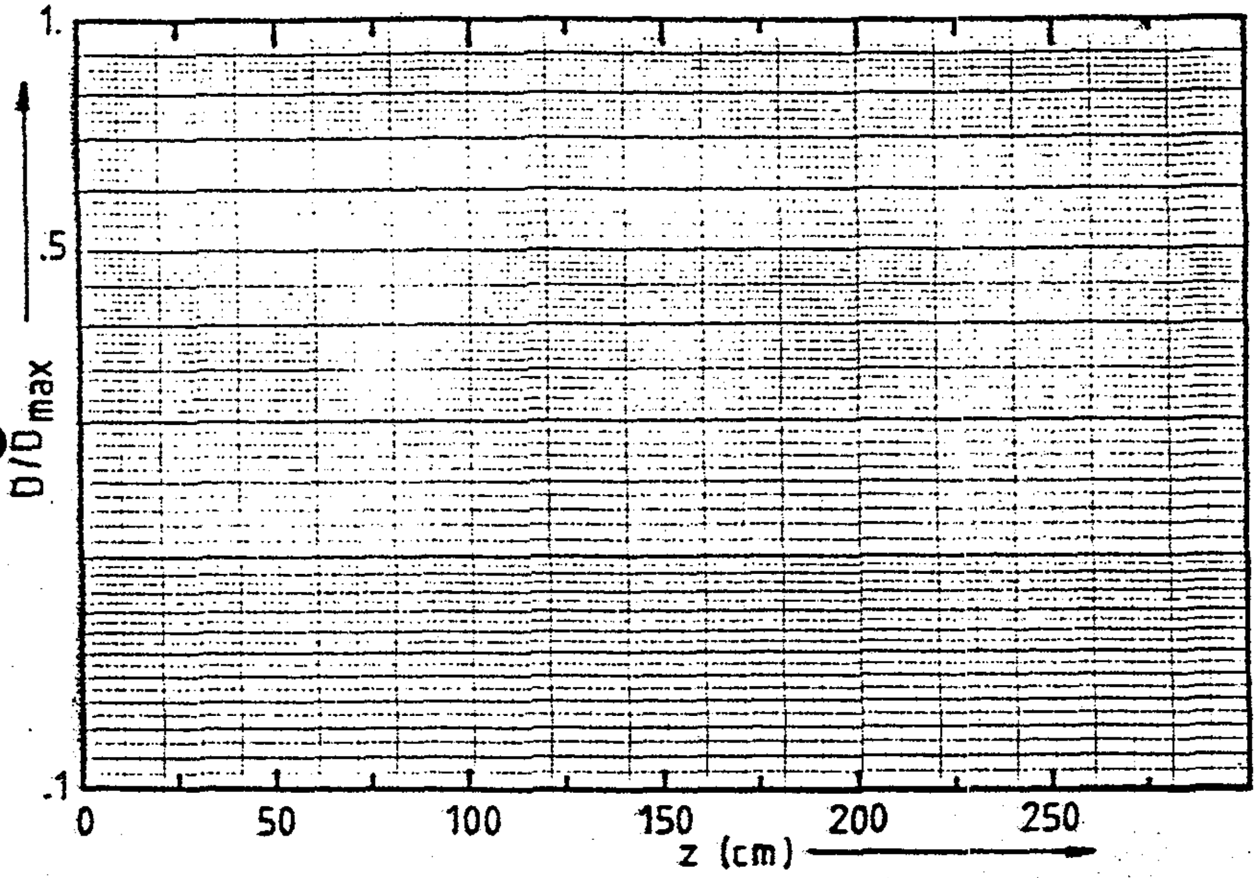
[1] averaging extends over
- the cavity height for the side wall dose rate
- the cavity cross sectional area for the top and bottom dose rate

[2] on the level of the fin-tips
[3] distances are measured from the fin-tips

Fig. IV/3:

Model Problem 4a: Plot of Azimuthally Averaged Dose Rate vs. Position dry cask

a) Side Wall



b) Lid

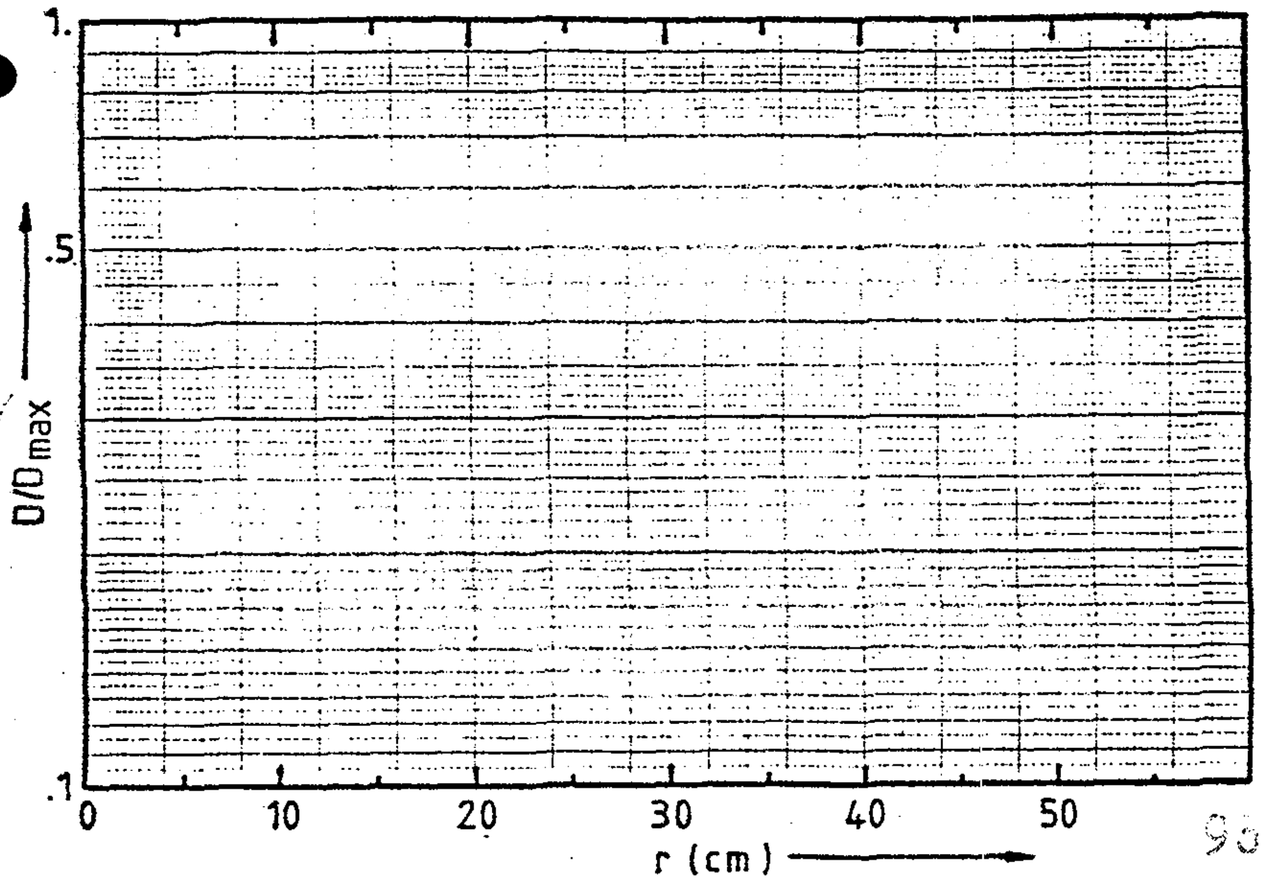
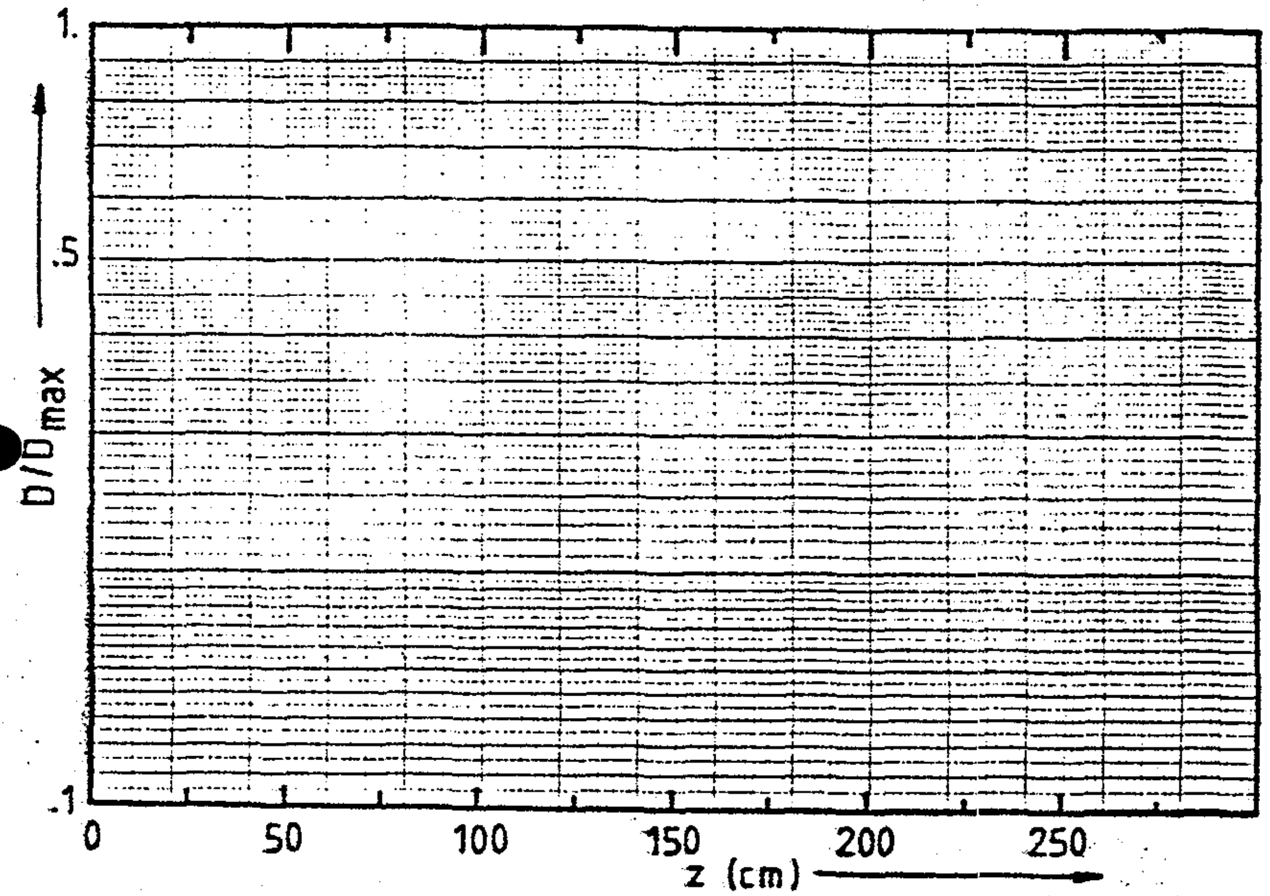


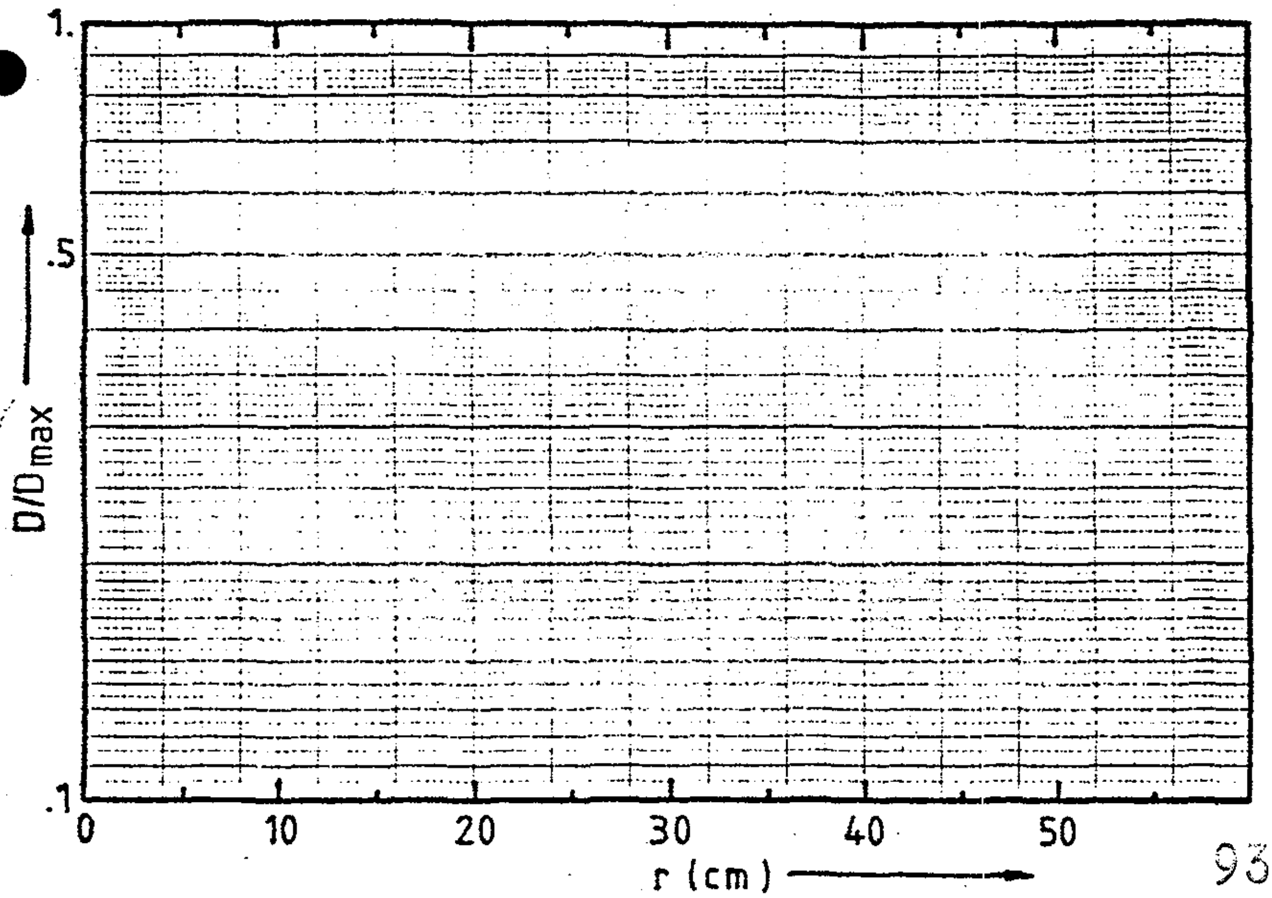
Fig. IV/4:

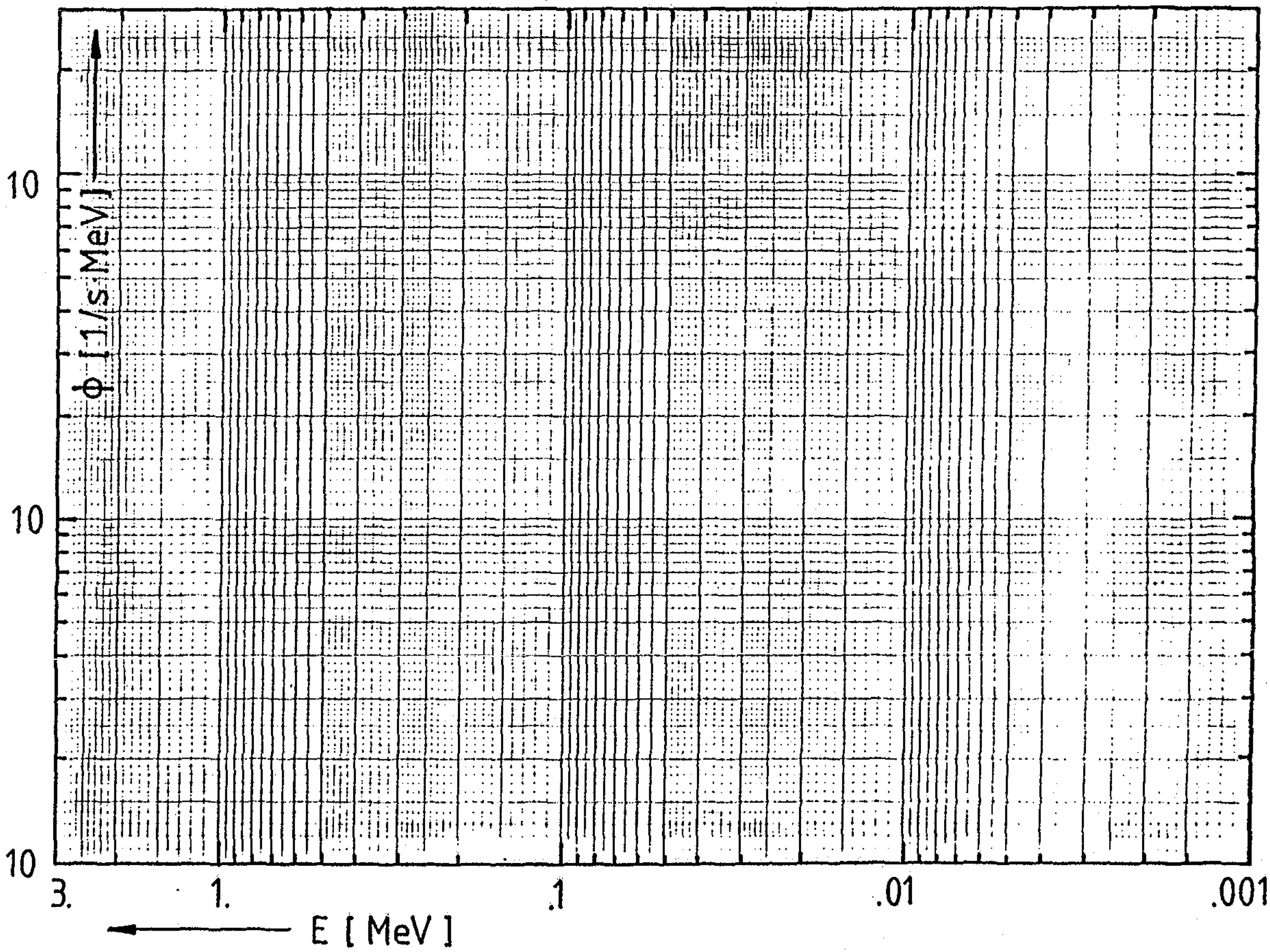
Model Problem 4b: Plot of Azimuthally Averaged Dose Rate vs. Position wet cask

a) Side Wall



b) Lid





Case a: full line

Case b: dashed line

Fig. IV/5: Model problem 4:
Plot of Surface Averaged Scalar Flux at Side Wall Surface (on the Level of Fin-Tips) vs. Energy
(averaging extends over the active length of fuel rods)

Model Problem V:

Cylindrical Iron Cask with Additional Neutron Shield

Source to be Calculated by the User

Preliminary Remark:

Model problem 5 is the starting point of a new phase of the shielding benchmark problem dealing with source term related problems. So it should be understood only as a first proposal in order to start the discussion. The experiences got from the first (geometrical) problems and the particular requests of the participants should lead to adequate modifications of the following model problems.

Model problem 5 consists of a cylindrical iron cask with detailed construction of a basket including 5 fuel assembly positions. Additional neutron shield is added as referred to for model problem 2, case a. For the sake of simplicity cooling fins are omitted. Instead of a predefined source the irradiation history of the fuel assemblies is given including all necessary operational reactor data. The burnup along the active zone of the fuel rods is assumed to be axially varying. Activation of construction steels in the neutron flux of the reactor is to be neglected.

A description of the model characteristics is given in table V/1. Basket and fuel assembly are described in tables A4 and A5, respectively. Tables V/2 and figures V/1 and 2 should be used to report output data as described in previous chapters. Additionally a short description of the burnup calculation performed should be reported including some representative results.

Table V/1:

Characteristics of Model Problem V

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
source region	detailed basket (see fig. IV/2) 5 fuel assemblies (see fig. IV/1)
side wall	cast iron
cask bottom	cast iron
cask lid	1.4313
cooling fins	none
neutron shield	polyethylene location as shown in fig. II/1

Source:

To be calculated by the participants using

- composition of fresh UO_2 fuel
- irradiation history
- reactor data
- axial burnup variation

as listed in table A6

Table V/2:
Summary Results of Model Problem 5

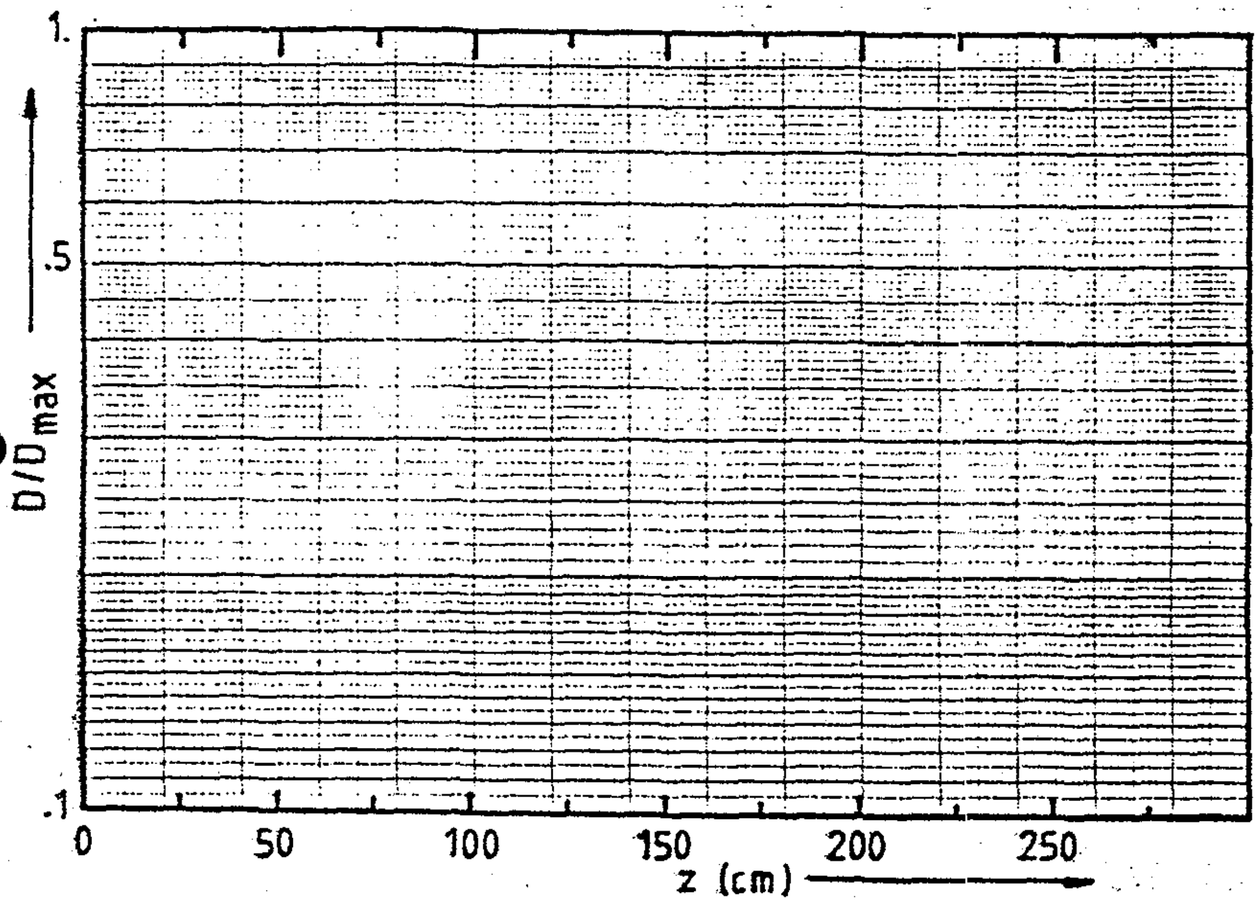
	surface averaged[1] dose rates (μSv/h)		
	n	γ (FP)	γ (capt.)
surface			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–
1 m distance			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–
2 m distance			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–
10 m distance			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–

[1] Averaging extends over
- the cavity height for the side wall dose rate
- the cavity cross sectional area for the top and bottom dose rate

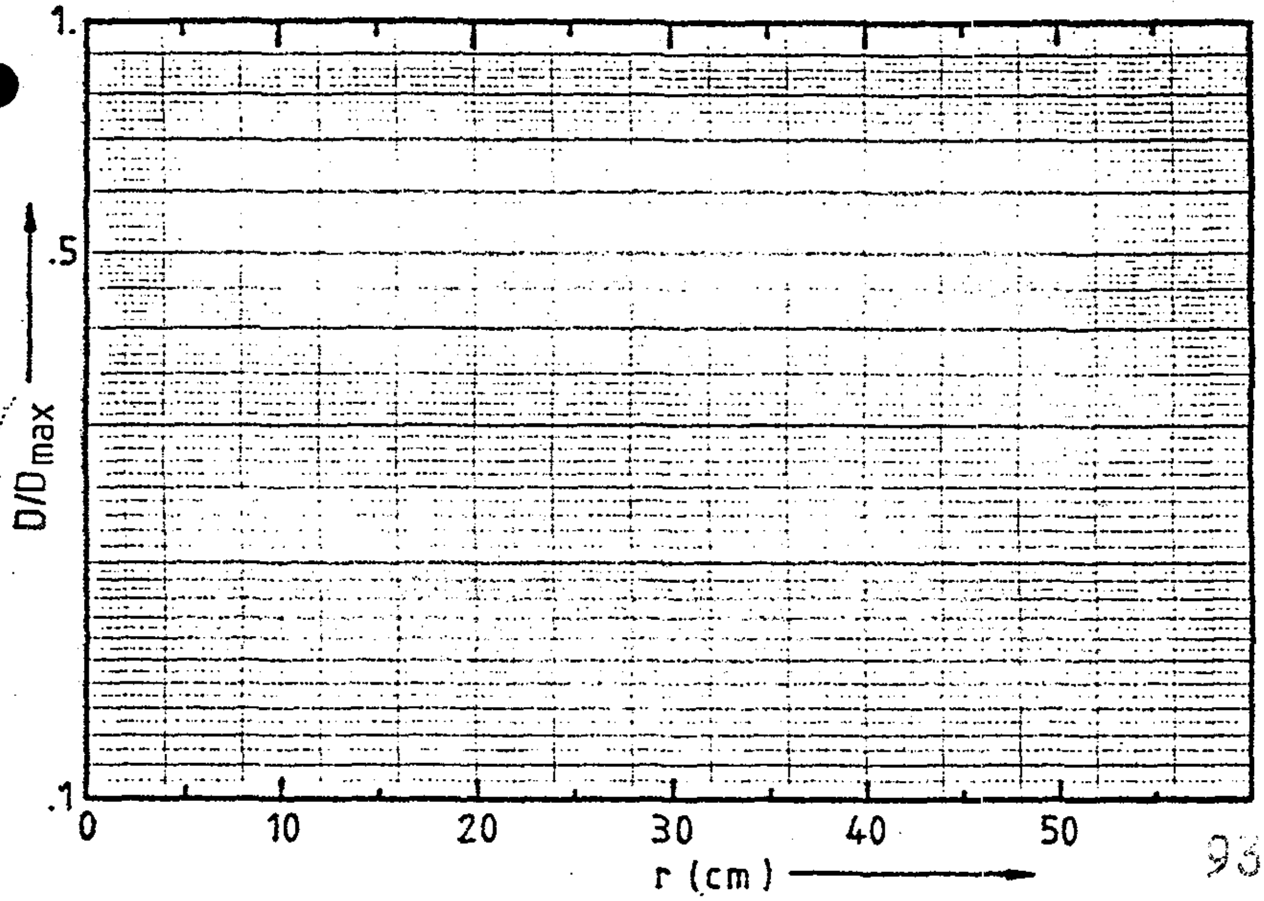
Fig. V/1:

Model Problem 5: Plot of Azimuthally Averaged Dose Rate vs. Position

a) Side Wall



b) Lid



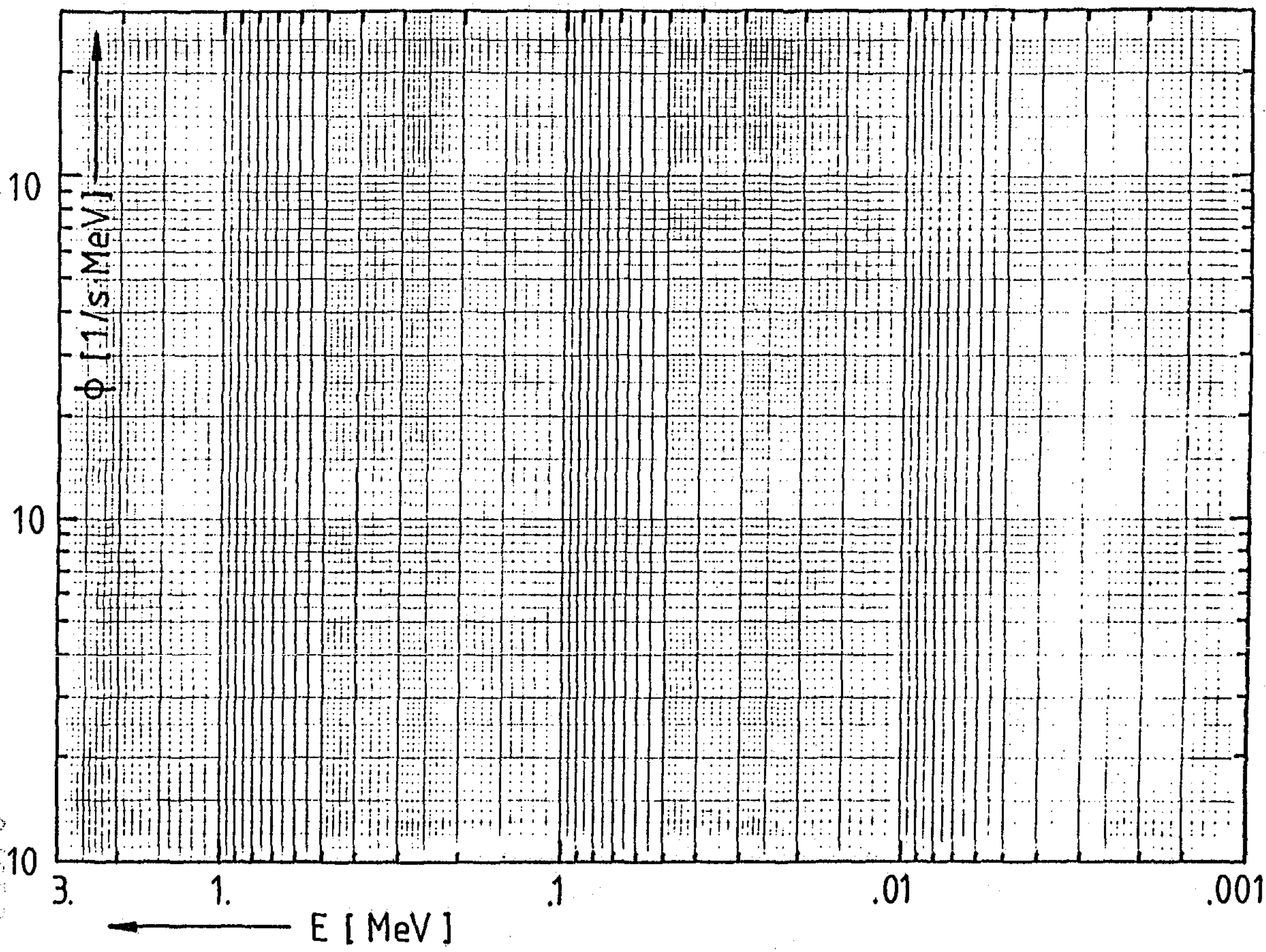


Fig. V/2:

Model problem 5:

Plot of Surface Averaged Scalar Flux at Side Wall Surface vs. Energy
(averaging extends over the active zone of fuel rods)

Model Problem 6:
As Model Problem 5 Using Mixed Oxide Fuel

Model problem 6 is identical with model problem 5 but the fuel to be irradiated is assumed to be mixed oxide fuel as referred to in table A4. A full description of the model characteristics is provided on the following page in table VI/1.

In table VI/2 and figures VI/1 and VI/2 results should be reported in the same manner as described for model problem 5.

Table VI/1:

Characteristics of Model Problem 6

Dimensions: See table A3

Material Assignment:

region	material for explanation see table A7
source region	detailed basket (see fig. IV/2) 5 fuel assemblies (see fig. IV/1)
side wall	cast iron
cask bottom	cast iron
cask lid	1.4313
cooling fins	none
neutron shield	polyethylene location as shown in fig. II/1

Source:

To be calculated by the participants using

- composition of fresh MOX-fuel
- irradiation history
- reactor data
- axial burnup bariation

as listed in table A6

Table VI/2:
Summary Results of Model Problem 6

	surface averaged[1] dose rates (μSv/h)		
	n	γ (FP)	γ (capt.)
surface			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–
1 m distance			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–
2 m distance			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–
10 m distance			
side wall	–	–	–
lid	–	–	–
bottom	–	–	–

[1] Averaging extends over
- the cavity height for the side wall dose rate
- the cavity cross sectional area for the top and bottom dose rate

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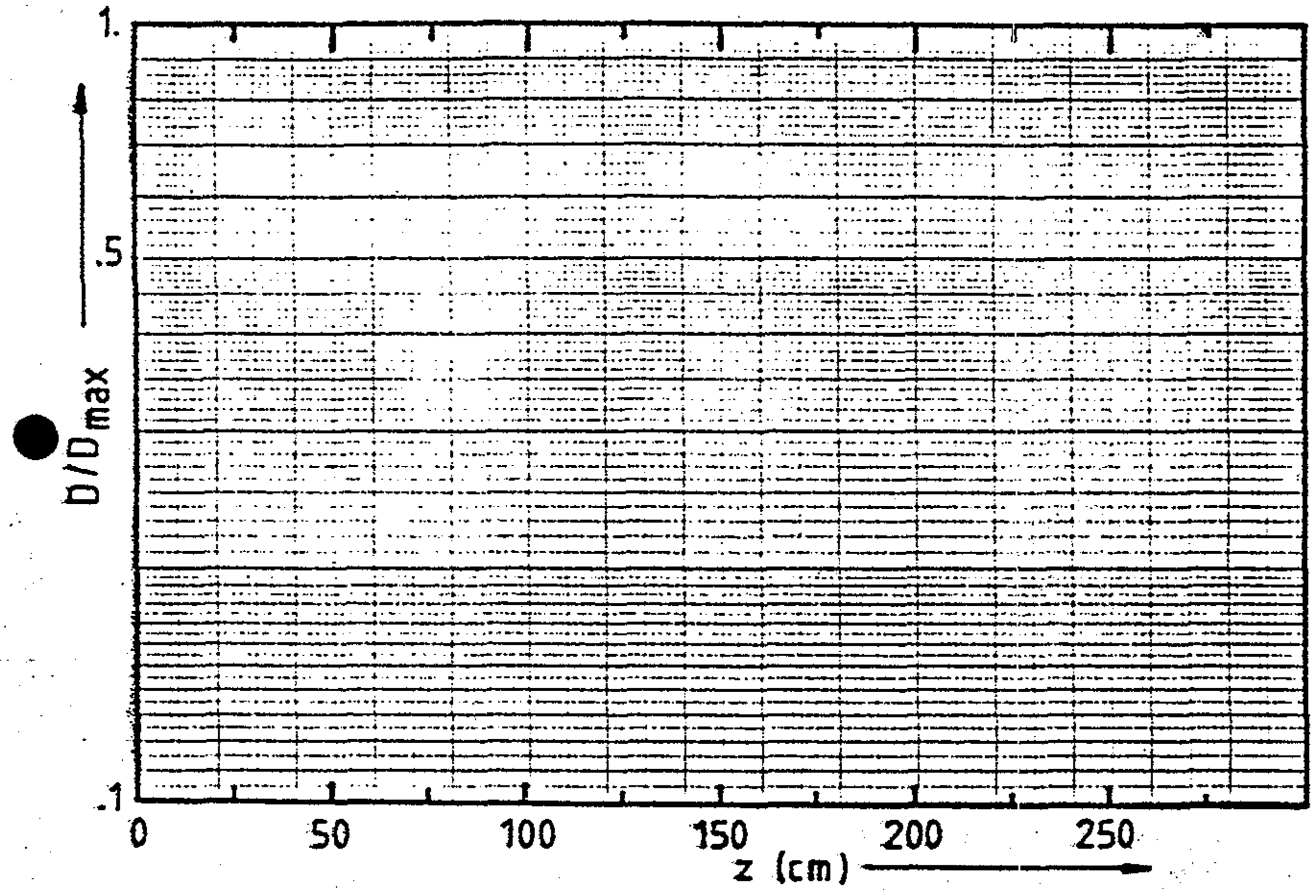
Fig. VI/1:

Model Problem 6: Plot of Azimuthally Averaged Dose Rate vs. Position

a) Side Wall



b) Lid

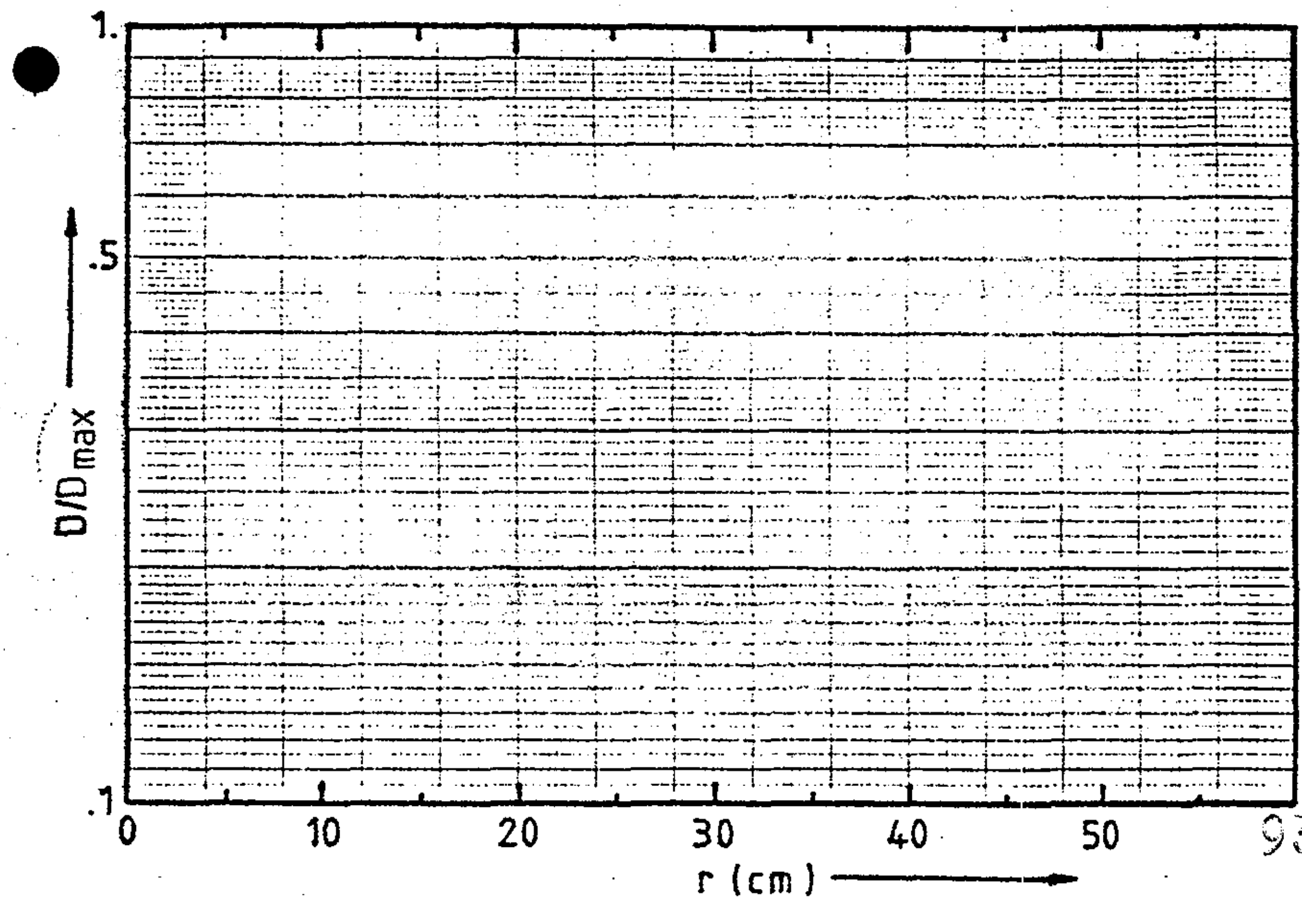
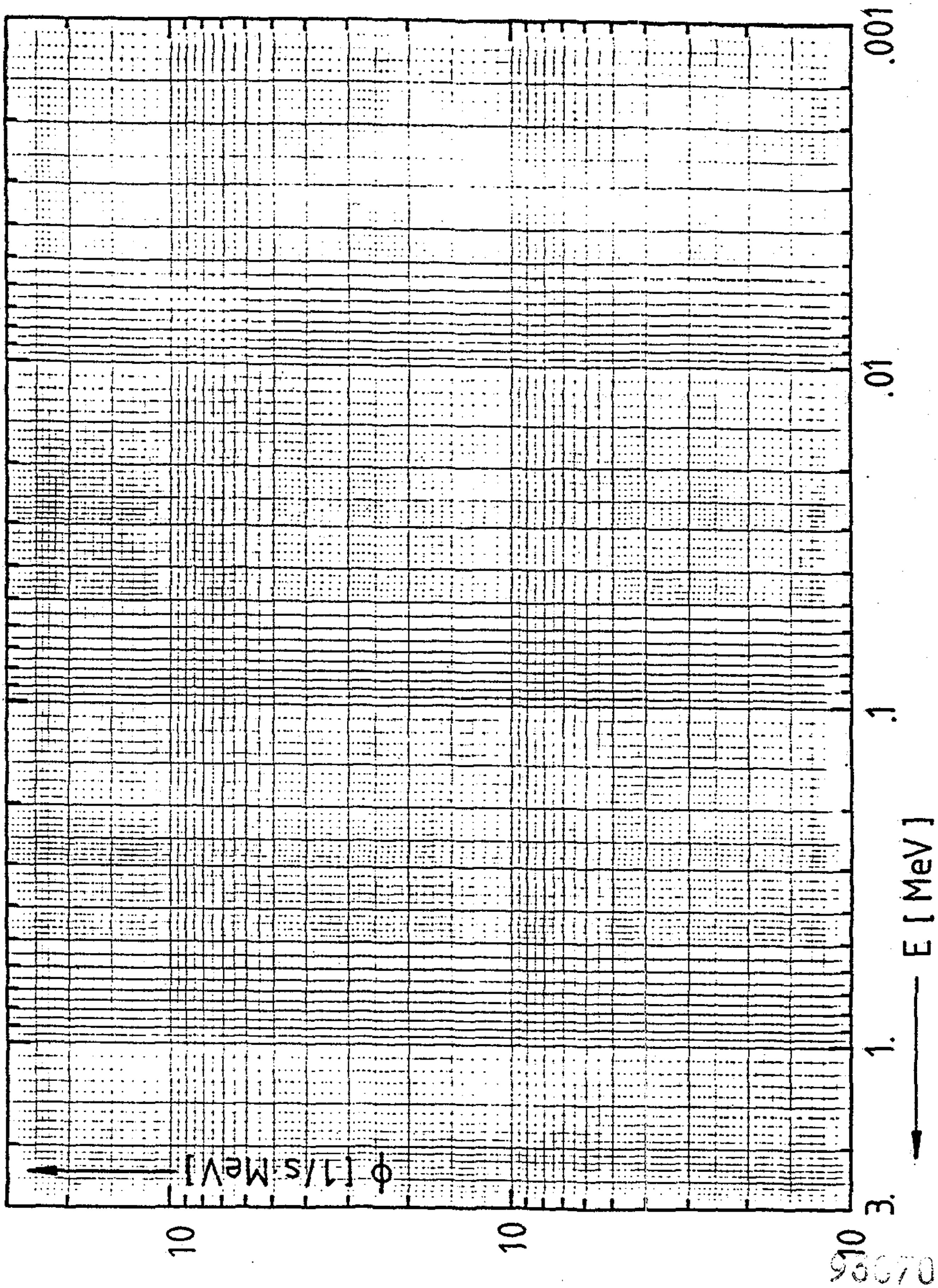


Fig. VI/2:

Model problem 6:

Plot of Surface Averaged Scalar Flux at Side Wall Surface vs. Energy (averaging extends over the active zone of fuel rods)



APPENDIX:

Tables and Figures of General Interest

Table A1:
Nomenclature

symbol	explanation
< >	average value
n	neutron
γ	gamma ray
E	particle energy
$\Phi(E)$	scalar flux density
DF(E)	flux-to-dose-conversion-factor
D	dose rate
$\Sigma(E)$	total macroscopic cross section
d	distance measured from the cask surface
z	axial coordinate
r	radial coordinate
ρ	density
T	temperature

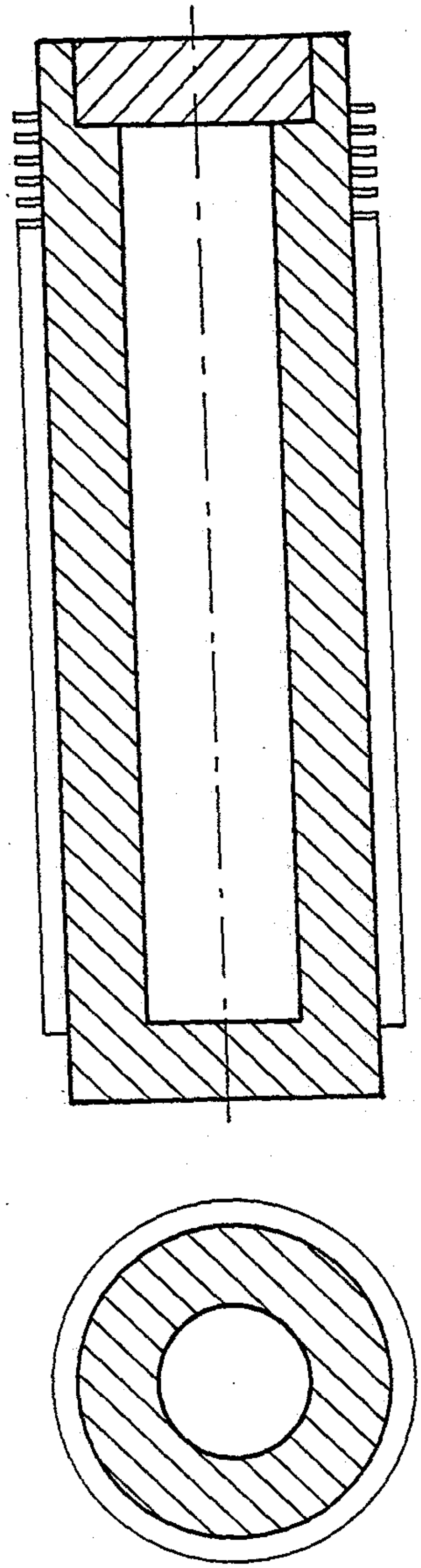


Fig. A1:
Cross Sectional View of Prototypic Cask

Table A2:

Basic Quantities to be Used for the Calculation

Avogadro's No.: $0.602252 \cdot 10^{24}$

Flux-to-dose-conversion-factors:

According to ANSI/ANS 6.1.1-1977 (see tables below)

Neutron Flux-To-Dose-Rate Conversion Factors. Polynomial Coefficients In Analytic Fit - - $\ln DF_n(E) = A + BX + CX^2 + DX^3$. $DF_n(E)$ = (rem/hr)/(n/cm²·s), E = neutron energy in MeV, and X = ln E

Neutron Energy (MeV)	A	B	C	D
2.5 −08 to 1.0−07	−1.2514 +01			
1.0 −07 to 1.0−02	−1.2210 +01	1.7165−01	2.6034−02	1.0273−03
0.01 to 0.1	−8.9302	7.8440−01		
0.1 to 0.5	−8.6632	9.0037−01		
0.5 to 1.0	−8.9359	5.0696−01		
1.0 to 2.5	−8.9359	−5.5979−02		
2.5 to 5.0	−9.2822	3.2193−01		
5.0 to 7.0	−8.4741	−1.8018−01		
7.0 to 10.0	−8.8247			
10.0 to 14.0	−1.1208 +01	1.0352		
14.0 to 20.0	−9.1202	2.4395−01		

Gamma-Ray-Flux-to-Dose-Rate Conversion Factors. Polynomial Coefficients in Analytic fit - - $\ln DF_g(E) = A + BX + CX^2 + DX^3$. $DF_g(E)$ = (rem/hr)/(photons/cm²-s), E = Photon energy in MeV, and X = ln E

Photon Energy (MeV)	A	B	C	D
0.01 to 0.03	−2.0477 +01	−1.7454		
0.03 to 0.5	−1.3626 +01	−5.7117−01	−1.0954	−2.4897−01
0.5 to 5.0	−1.3133 +01	7.2008−01	−3.3603−02	
5.0 to 15.0	−1.2791 +01	2.8309−01	1.0873−01	

Table A3:

Dimensions of Cask Body

(for further explanation see also fig. A1)

cavity (circular cross section)	
diameter	80 cm
height	450 cm
cask body (circular cross section)	
outer diameter	
without cooling fins	156 cm
including cooling fins	180 cm
total height	530 cm
thickness of side wall	38 cm
thickness of bottom	38 cm
thickness of lid	42 cm
azimuthal cooling fins	
number	35
height	12 cm
top width	2 cm
bottom width	5 cm
spacing	13 cm
location	mid plane of the first fin on the top level of cask cavity

Table A4:

Characteristics of Basket

(for further details see also fig. A3)

number of fuel assembly positions	5
material	1.4541
total weight	1043 kg
total height	440 cm
thickness of structure sheets	1.0 cm
basket phase	none
position of basket inside the cask	resting on the cask base
position of fuel assemblies inside the basket	resting on the cask base centrally positioned within the compartments

Table A5:

Characteristics of Model Fuel Assembly

As used in model problems IV - VI
(for further details see fig. IV/1)

Dimensions

total assembly length	445 cm
width over flats	23.0 cm
total rod length	393 cm
length of active zone	342 cm
- lower rod plug	1.8 cm
- bottom expansion space	31.4 cm
- top expansion space	16.0 cm
- upper rod plug	1.8 cm
height of bottom fitting	23.0 cm
- top fitting	27.0 cm
void between bottom fitting and rod	1.0 cm
void between top fitting and rod	1.0 cm
total number of rod positions	225
fuel rods	210
control rods	15
pitch (square lattice)	1.53 cm
pellet outer diameter	0.969 cm
cladding inner diameter	1.00 cm
- outer diameter	1.15 cm
guide tube inner diamter	1.25 cm
- outer diameter	1.40 cm

Table A5 (continued)

Materials and Weights per Assembly

	Material [1]	Weight/Assy. (kg)
fuel	UO_2	513.7
fuel cladding	Zr-4	135.9
guide tube	Zr-4	11.9
lower rod plug	1.4541	0.74
upper rod plug	1.4541	0.74
expansion spaces	void	---
bottom fitting [2]	1.4541	12.3
top fitting [2]	1.4541	14.4

[1] material composition listed in table A7

[2] for the first step the fitting mass should be homogenized over the fitting length (effective steel density ρ = 1.01 g/cm^3)

Table A6:

Composition of Fresh Fuel of Model Assembly

as used in model problems V and VI

Dimensions, materials and weights as specified in table A5

Isotope	UO_2-fuel wt %	MOX-fuel wt %
U -234	0.01	0.01
U -235	2.86	0.59
U -236	0.12	0.10
U -238	85.15	83.95
Pu-238	-	0.01
Pu-239	-	2.70
Pu-240	-	0.64
Pu-241	-	0.12
Pu-242	-	0.02
O	rest	rest

Table A6 (continued):

Irradiation History

time (days)	specific power (kW/kg hevimet)	burnup (MWd/kg hevimet)
300	32.0	9.6
40	0.0	9.6
300	26.0	17.4
40	0.0	17.4
300	38.0	28.8
40	0.0	28.8
300	37.3	40.0

Cooling time 2 y

Operational Data of Model Reactor:

Coolant pressure 160 bar
Average temperature 300 °C
Average boron content 450 ppm

Table A6 (continued):

Axial Burnup Distribution

rod position (cm) from	to	peaking[1]	rod positon (cm) from	to	peaking[1]
0	10	.42	170	180	1.23
10	20	.60	180	190	1.21
20	30	.78	190	200	1.19
30	40	.92	200	210	1.16
40	50	1.04	210	220	1.14
50	60	1.15	220	230	1.11
60	70	1.21	230	240	1.08
70	80	1.27	240	250	1.04
80	90	1.29	250	260	1.00
90	100	1.33	260	270	.94
100	110	1.33	270	280	.88
110	120	1.32	280	290	.78
120	130	1.31	290	300	.68
130	140	1.29	300	310	.56
140	150	1.28	310	320	.44
150	160	1.26	320	330	.31
160	170	1.25	330	342	.18

[1] defined as local burnup/average burnup

Table A7:

Composition of Materials Used

material	total density (g/cm^3)	element	content (wt %)
1.4541	7.8	Mn	2.0
		Cr	18.0
		Ni	10.0
		Fe	rest
1.4313	7.7	Mn	1.5
		Cr	12.5
		Ni	4.0
		Fe	rest
cast iron	7.0	C	3.25
		Si	1.4
		Ni	1.1
		Fe	rest
polyethylene	0.91	C	14.4
		H	85.6
epoxi cast resin	1.4	C	0.95
		H	0.09
		O	0.21
Zr-4	6.5	Zr	97.9
		Sn	1.6
		Fe	0.5
homogeneous source (dry)	1.97	U-235	1.6
		U-238	49.1
		O	6.1
		Zr-4	16.6
		1.4541	26.6
homogeneous source (wet)	2.81	U-235	1.1
		U-238	34.4
		O	4.3
		Zr-4	11.6
		1.4541	18.6
		water	30.0

Table A8:

Energy Spectrum of Neutron and Gamma Radiation Source

Neutrons:

continuous spectrum according to following formula:

$$s(E) = A \cdot \exp(-B \cdot E) \cdot \sinh(C \cdot E)^{1/2}$$

$A = 0.451$
$B = 1.035$
$C = 2.29$

Gammas:

discrete lines according to following table:

Energy (MeV)	Source strength (s^{-1})	
0.6	2.53 E16	1)
0.7	2.32 E16	
1.0	6.95 E14	
1.3	5.50 E14	
1.7	6.15 E12	
2.1	2.70 E14	
2.4	2.54 E12	
2.8	1.32 E12	

1) read as $2.53 \cdot 10^{16}$