SE7900239

Studsvik Arbetsrapport - Technical Report

Projektidentifikation – Project identification	Datum – Date	Org enh och nr – Report No.
	1979-03-23	K3-79/106

Titel och författare – Title and author

POWER MEASUREMENT IN BOILING CAPSULES IN R2 USING DELAYED NEUTRON DETECTOR

Gunnar Rönnberg

Distribution

Godkänd av – Approved by	Kontonr	Rapporten skall förhandsaviseras – Internal notes
		☐

ABSTRACT

LWR fuel testing is performed in the R2 reactor by irradiation in both loops and so-called boiling capsules.

The loops have forced cooling, and the power can be measured calorimetrically by conventional instrumentation.

The boiling capsules have convection cooling, and it has therefore been necessary to develop a special technique for power measurement, the delayed neutron detector (DND).

The DND is a pneumatic rabbit system, which activates small uranium samples in the boiling capsules and counts the delayed neutrons for determination of the fission rate.

This report describes the equipment used, the procedure of measurement, and the method of evaluation.

The 1 σ uncertainty in linear heat rate of the fuel rods is estimated to be

5 % at 0 burn-up
7 % at 20 MWd/kg U.

BL 4848 A

CONTENTS

1 Introduction

The boiling capsules are used for irradiation of fuel rods, normally in bundles of 4 rods per capsule. In order to obtain both a high fast neutron flux density and a rather low power density, the bundle is surrounded by a hafnium shield. In some cases the capsule is also surrounded by an outer hafnium shield. See figure 1.

The capsule is cooled by the R2 primary system. The rods are cooled by convection of the pressurized coolant in the capsu[illegible], and there is subcooled boiling at the surfaces of the [illegible].

Th[illegible] power is a very important irradiation parameter, wh[illegible]s needed for calculation of

- the temperature distribution in the fuel and cladding.
- the stress state of the fuel and cladding.
- the amount and distribution of fission products.
- the structure of the damage from fast neutron fluence.

2 Conventional methods of power measurement

The following methods have been considered for use in boiling capsules, but none of them seems to be satisfactory.

2.1 Calorimetric measurement by flow and temperature difference in riser

There are three major difficulties involved:

- the convection flow is slow and difficult to measure.
- the radial temperature gradient is very large (250°C difference from rod surface to reactor

primary coolant) and the corrections for radial heat flow should be large.

- the gamma heating correction should be large.

2.2 Calorimetric measurement using temperature difference between reactor primary coolant and BOCA coolant

The difficulties here are:

- uncertainties in heat transfer parameters.
- to define an effective average temperature inside the capsule.
- to correct for the large gamma heating in the whole capsule.

The method is used in the BOCA system as the standard method for relative power measurement after calibration with the DND system.

Both calorimetric methods could have been possible for absolute measurement, if some sort of calibration was performed. However, experiments using electrically heated rods for calibration have been unsuccessful.

2.3 Activation detectors

It should, in principle, be possible to use normal activation detectors for measurement of the neutron flux density and calculation of the fission rate in the rods. The problem is, that the very different energy dependances of the neutron cross sections give large corrections in the relation between fission rate and foil activity in the rather unusual neutron spectrum in the BOCA with its Hf shields.

Besides, foil handling, gamma counting, and evaluation should be expensive and cause a long delay between measurement and result.

2.4 Self-powered detectors

The neutron spectrum dependance and the gamma sensitivity of these make them useful not for absolute measurements but only for relative measurements after calibration.

They have been used in BOCA rigs after calibration with the DND system.

2.5 Fission chambers and fission foils

With fission chambers or fission foils it is possible to avoid the neutron spectrum corrections.

The neutron flux density in the BOCA rigs is at the upper limit for existing fission chambers, even in the (Campbell) variance mode of operation. In this flux range we do not consider them to be accurate enough for absolute measurements.

The evaluation of fission foil activity is much more difficult (and therefore also less reliable) than the delayed neutron counting.

3 Delayed neutron detector, DND

As mentioned above, a way to avoid the spectrum problems with the normal activation detectors is to use fissile material as detector. The dependance of the yields of fission products on neutron energy is so small that it can be neglected (except in fast reactors).

By measuring not the usual fission product activity as in fission foils but the delayed neutron activity, the detection is very much simplified, which also helps to obtain reliability. As the neutron activity decays rather fast, the time needed for a measurement also becomes small.

The delayed neutron detector (DND) method was therefore developed at R2 and has been used for power measurements in all boiling capsule experiments performed. An extension for use in water loops (in addition to the normal calorimetric system) is also under way.

A description of how the measurement of the (specific) fission rates

f_5, kg^{-1} s^{-1}, in U-235 and

f_8, kg^{-1} s^{-1}, in U-238

in the activation positions is performed, is given below. In chapter 4 and 5 the translation to fission rate in the fuel rods beside is described. The relation between fission rate and power is discussed in chapter 6 and an error estimate is given in chapter 7.

3.1 Mechanical equipment

A pneumatic rabbit loop system has been built with connections

- to a calibration position in the D_2O blanket of the R2 core. The neutron energy spectrum there is very well thermalized and well known.
- to two activation positions in every boiling capsule. The positions are at the axial maximum of neutron flux density (at average control rod level) and close to (3 mm away from) the fuel rods inside the hafnium shields. The two positions are symmetrically placed in the rig, one on the side towards the core centre, the other at the "back" side. See figure 1.
- and to a delayed neutron counting position.

Rabbits have been produced with a filling of uranium with different degrees of enrichment, from 0.2 % to 5 % U-235. The amount of uranium in each rabbit is very small and has not to be known.

3.2 Control and detector system

An automatic control system takes care of the blow-in, activation, blow-out, waiting, start of counting, and **stop of counting according to the following time schedule:**

Time seconds	Operation
0	Start of activation
258	End of activation
316	Start of counting
326	End of counting

This time schedule is very exact and reproducible.

The detector system for the delayed neutrons is using

- two Boron counter tubes symmetrically placed one on each side of the counting positions for the rabbits in holes of a piece of polyethylene.
- standard pulse counting system with pulse height discriminator. The gamma sensitivity of the system is negligible.

The dead-time has been measured by activating both a rabbit with uranium and rabbits with cobalt at different power levels of the reactor.

The dead-time corrections are normally about 10 %.

3.3 Procedure of DND-measurement

A DND-measurement is performed in three steps:

(1) activation of a cobalt wire in the calibration position and calculation of the fission rate, f_0 (kg^{-1} s^{-1}) for U-235 in this position (f_8 is negligible there). See 3.4.

(2) activation of a number of rabbits with different degrees of enrichment both in the calibration position and in the points of measurement in the boiling capsule.

(3) evaluation of the measurements giving
f_5 (kg^{-1} s^{-1}) = fission rate of U-235 and
f_8 (kg^{-1} s^{-1}) = "- U-238 in
the points of measurement in the boiling capsules. See 3.5.

Normally three cobalt wires (before, middle and after) are used to make sure that there is no drift in the neutron flux density in the calibration position during the measurement. Besides, every rabbit is run in the calibration position both before and after the other positions. In this way a good routine control of the system is obtained. Furthermore, the relation between fission rate and number of counts for each rabbit in the calibration position is compared with previous measurements (should be, and is, constant).

3.4 Measurement of fission rate in calibration position using Co-wire

A rabbit is loaded with a Co-wire of 0.5 mm diameter. It is activated using the normal rabbit system giving an activation time of 258 seconds.

The Co-60 activity is measured using NaI scintillation detector with multichannel energy analyser. A standard

Co foil is used for calibration. The foil has been delivered by IAEA and its activity has a nominal accuracy of 0.6 %.

The neutron spectrum is well known and deviates very little from the Maxwellian distribution with a neutron temperature T_n of 50°C.

The ratio between epithermal and thermal flux densities has been measured to be

$$\frac{\phi_{epi}}{\phi_o} = 0.015$$

And the relation between fast and thermal to be

$$\frac{\phi_{fast}}{\phi_o} = 0.01$$

The fission rate f_o is calculated from the cobalt activity using the ratio

$$\frac{579 \cdot 0.97 \cdot 1.016}{37.5 \cdot 0.90 \cdot 1.022}$$

between the effective cross sections of U-235 (fission) and Co-59.

The parameters are:

579 = 2200 m/s fission cross section for U-235 (barns)
37.5 = "- activation "- Co-59 "
0.97 = Westcott G-factor for U-235 with T_n = 50°C
0.90 = flux depression and self shielding factor for 0.5 mm Co wire
1.016 = correction for epithermal fission of U-235
1.022 = "- activation of Co-59

3.5 Evaluation of DND-measurement

From Co activation we know

f_o = fission rate (kg^{-1} s^{-1}) of U-235 in the calibration position.

A uranium rabbit gave the following number of counted delayed neutrons after dead-time correction:

n_o = after activation in calibration position

n = after activation in boiling capsule

The number of precursors of delayed neutrons per fission is rather well known:

$C_{51} = 0.00052 \pm 0.00005$	for U-235	in group	1
$C_{52} = 0.00346 \pm 0.00018$	"-		2
$C_{81} = 0.00054 \pm 0.00005$	" U-238	"-	1
$C_{82} = 0.00564 \pm 0.00025$	"-		2

group 1 precursors have a decay constant of

$\lambda_{51} = 0.0124\ s^{-1}$ for U-235

$\lambda_{81} = 0.0132\ s^{-1}$ for U-238

group 2 precursors have

$\lambda_{52} = 0.0305\ s^{-1}$ for U-235

$\lambda_{82} = 0.0321\ s^{-1}$ for U-238

(Other groups have already decayed after 58 seconds.)

Suppose now, that the neutron detectors have an efficiency of

ω_1 for group 1 delayed neutrons

ω_2 "- 2 "-

Suppose also that the (unknown) uranium amount in the rabbit is m and that the (known) degree of enrichment is a.

Then

$$n_o = f_o \cdot m \cdot a\,(\omega_1 \cdot C_{51} \cdot k_{51} + \omega_2 \cdot C_{52} \cdot k_{52})$$

$$n = f_5 \cdot m \cdot a\,(\omega_1 \cdot C_{51} \cdot k_{51} + \omega_2 \cdot C_{52} \cdot k_{52}) +$$
$$+ f_8 \cdot m \cdot (1-a)(\omega_1 \cdot C_{81} \cdot k_{81} + \omega_2 \cdot C_{82} \cdot k_{82})$$

Where

$$k_{ij} = \frac{1}{\lambda_{ij}} \cdot (1-e^{-\lambda_{ij} \cdot t_o}) \cdot (e^{-\lambda_{ij} \cdot t_1} - e^{-\lambda_{ij} \cdot t_2})$$

$t_o = 258$ s
$t_1 = 58$ s
$t_2 = 68$ s

Division of the equations gives

$$f_o \cdot n/n_o = f_5 + f_8 \cdot \frac{1-a}{a} \cdot C$$

where
$$C = \frac{k_{81} \cdot C_{81} \cdot \omega_1/\omega_2 + k_{82} \cdot C_{82}}{k_{51} \cdot C_{51} \cdot \omega_1/\omega_2 + k_{52} \cdot C_{52}}$$

ω_1/ω_2 has been found to be 1.16 by measuring the time-dependance of the count rate, fitting to two exponentials, and comparing with yield values. The reason why $\omega_1 \neq \omega_2$ is that the energy spectra of the neutrons differ.

With the parameters given above, $C = 1.3 \pm 0.1$.

Thus we finally have obtained

$$f_o \cdot (n/n_o) = f_5 + f_8 \cdot \left(\frac{1-a}{a}\right) \cdot 1.3$$

If two or more rabbits with different degrees of enrichment (a) are used, the unknown fission rates f_5 and f_8 can be found.

3.6 Example of DND-measurement

An example of a measurement is given below (measurement performed 77-01-20).

Activation position	Rabbit enrichment a	Number of pulses before dead-time corr	after	$\frac{1-a}{a}\cdot 1.3$	$f_o \cdot n/n_o$ $10^{16} kg^{-1} s^{-1}$
7A	Cobalt				
7A	0.0022	41693	42185		
4A	"	87048	89223	590	7.09
4B	"	114422	118209	590	9.39
7A	"	41673	42165		
7A	Cobalt				
7A	0.0163	224029	239022		
4A	"	304378	332735	78	4.66
4B	"	366091	407903	78	5.72
7A	"	223817	238781		
7A	0.025	608183	733007		
4A	"	768039	978457	51	4.52
4B	"	882432	1172014	51	5.41
7A	"	597808	717989		
7A	0.05	704928	878283		
4A	"	866972	1144898	24.7	4.40
4B	"	977042	1344994	24.7	5.17
7A	"	696765	865648		
7A	Cobalt				

7A is the calibration position.
4A and 4B are DND tubes in a boiling capsule behind and in front of the fuel rods.

The dead time is $2.8 \cdot 10^{-6}$ s.

The activity of the Co detectors was

	$8.72 \cdot 10^9$	$kg^{-1}\ s^{-1}$	for first detector
	$8.74 \cdot 10^9$	"-	" second "
	$8.63 \cdot 10^9$	"-	" third "
Mean value:	$8.70 \cdot 10^9$		

Then f_o can be calculated (see 3.4).

$$f_o = 3.35 \cdot 10^{16} \ \mathrm{kg}^{-1} \ \mathrm{s}^{-1}$$

In figure 2 $f_o \cdot (n/n_o)$ is plotted as a function of $\frac{1-a}{a} \cdot 1.3$.

By comparison with the formula

$$f_5 + f_8 \cdot \frac{1-a}{a} \cdot 1.3 = f_o \cdot (n/n_o) \qquad \text{we find}$$

$$f_5 = 4.28 \cdot 10^{16} \ \mathrm{kg}^{-1}\mathrm{s}^{-1} \text{ in 4A and } 5.06 \cdot 10^{16} \text{ in 4B}$$

$$f_8 = 4.8 \cdot 10^{13} \quad \text{"-} \qquad\qquad 7.3 \cdot 10^{13} \text{ "-}$$

The results obtained from the 0.0022 enriched rabbit is outside the figure, but has been included in the fitting of the straight lines.

The evaluation of the measurements is normally computerized, using the same method as in this example.

4 Power distribution in BOCA rigs

The relation between neutron flux density in the DND activation positions and in the fuel rods has been measured by activation on low power of Co wires in the DND tubes and in central holes of special fuel rods.

The measurements have been corrected for small differences between the activation rigs and the real rigs and for the differences in fuel rods.

The measurements have been compared with calculations performed with a multigroup transport code BUXY-CLUCOP. The calculations give the ratio between the mean fission rate in the two DND positions and the mean fission rate in the four rods at axial maximum. The results are found in table 5.

The measurements have been performed in different reactor positions. Both measurements and calculations have been performed with different Hf shields. The differences between measurements and calculations are about 2 %. (See table 5).

The results normally used are

- calculations for the ratio between mean of DND positions and mean of fuel rods.
- measurements for distribution of power between the four rods.

5 Burn-up corrections

The burn-up of the fuel rods introduces new problems:

- burn-up of U-235 must be calculated.
- build up of Pu isotopes must be calculated.
- ratio between fission rate of Pu isotopes and U isotopes must be estimated.
- changes in neutron flux depression in rods has to be calculated.

The corrections are calculated by the program BUXY-CLUCOP using 25 neutron energy groups and a geometry, which is the true one in the boiling capsule and uses a typical mean R2 core mix outside the capsule.

BUXY-CLUCOP is a multigroup two-dimensional transport S4 code, which has been developed in Studsvik, mainly to be used for cell calculations. It has been very much used for burn-up calculations for BWR (and PWR) fuel.

A control of the calculations is obtained by comparison of the calibration factors for calorimetric power measurement (see 2.2) at low burn-up and at higher burn-up.

No reason has been found so far to introduce further corrections.

The error in the burn-up correction is rather difficult to estimate. The estimate in table 1 (5 % at 20 MWd/kg) is obtained by comparison of different calculations and from discussions with people with extensive experience of the program.

6 Power density

When the fission rates ($kg^{-1}s^{-1}$) are known, the only parameters needed to calculate the linear power density are

- the energy absorbed in the rod per fission. The constants used are

$2.86 \cdot 10^{-11}$	J/fission	for	U-235
$3.14 \cdot 10^{-11}$	-"-		U-238 and
$2.95 \cdot 10^{-11}$	-"-		Pu isotopes

- the amount of fuel per unit length of the rod. No error is supposed in this figure, which is obtained from the fuel supplier.

- the gamma heating. This has been measured in many positions of the R2 core, and is routinely calculated for every experiment position in R2 cores. The calculated value is the heating in stainless steel. A 40 % higher value is used for UO_2 (estimated from calculations and from measurements with lead).

The axial neutron flux distribution is known from activations as a function of control rod level of R2, and thus the linear power density in every axial position is known.

Corrections for axial burn-up distribution are obtained from the BUXY-CLUCOP results.

7 Error analysis

Estimates of the errors in the parameters used are given in tables 1 to 4.

Table 2 shows, that the

error in f_5 is about 2 % (1 σ)
"- f_8 "- 10 % "

The error in f_8 can be decreased, but as it has no significant influence on the error of the rod power, no effort has been put on this so far.

Table 1 shows, that the error in linear power density is

about 5 % (1 σ) at 0 burn-up and
about 7 % (1 σ) at 20 MWd/kg

Table 1

Power measurement in boiling capsules. Error estimate.

Parameter	Typical value	Error, 1 σ
DND-measurement of fission rate in DND-tubes in BOCA rigs. See fig 1 and table 2.	$4 \cdot 10^{16}$ in U-235 $4 \cdot 10^{13}$ in U-238 (kg^{-1} s^{-1})	2 % 10 %
Relation between mean fission rate in fuel rods and in DND-tubes. Measured by activation. Calculated with BUXY-CLUCOP. Error = difference between meas and calc.	0.87	2 %
Power distribution factors for individual rods. Measured by activation.	≈ 1	2 %
Correction for burn-up and Pu build-up in fuel rods. Calculated with BUXY-CLUCOP.	1 - 0.8 (0 - 20 MWd/kg)	0 - 5 %
Relation between linear power density from fissions and fission rate. Energy absorbed/fission and amount of fuel per unit rod length.	$6 \cdot 10^{-13}$ $(W/m)/(kg^{-1}s^{-1})$	1 %
Addition of calculated gamma heating. See appendix 1.	3 kW/m	0.5 kW/m 2 %
Measurement of power changes by coolant temperature and selfpowered detectors.		2 %

Total error, 1 σ: $(2^2 + (0.05 \cdot 10)^2 + 2^2 + 2^2 + 1^2 + 2^2 + 2^2)^{1/2}\ \% \approx$

$\approx$ 5 % at 0 burnup

and $(21 + 5^2)^{1/2}\ \% \approx$ 7 % at 20 MWd/kg

Table 2

DND-measurement of fission rate. Error estimate.

Parameter	Typical value	Error, 1 σ
Measurement of fission rate f_o in U-235 in calibration position. See table 3.	$3 \cdot 10^{16}$ kg^{-1} s^{-1}	2 %
Measurement of relation between fission rate in rig and in calibration position. See table 4.	1	0.4 %
Evaluation of fission rates f_5 and f_8		$\approx$ 0 % for f_5 $\approx$ 10 % for f_8

Total error in f_5: $(2^2 + 0.4^2)^{1/2}$ % $\approx$ 2 %

"- f_8 $\approx$ 10 %

Table 3

Measurement of fission rate f_o in calibration position. Error estimate.

Parameter	Value	Error, 1 σ
Activation time for rabbit with 0.5 mm Co wire.	258 s	0.3 %
Half-life of Co-60	5.26 years	0.2 %
Measurement of Co-60 activity by NaI crystal and multichannel analyser. Error from standard foil activity.		1 %
Weight of Co wire		≈ 0
Correction for self-shielding and flux depression in Co wire.	0.90	1 %
Relation between 2200 m/s cross sections for U-235 and Co-59.	579/37.5	0.3 %
Correction for Westcott G-factor T_n = 50 C.	$\frac{0.97\ (U)}{1.00\ (Co)}$	0.5 %
Correction for epithermal neutrons. α = 0.015.	$\frac{1.016\ (U)}{1.022\ (Co)}$	0.2 %
Statistical error from gamma counting and from neutron flux changes in calibration position. Found from differences between consecutive measurements.		1 %

Total error in fission rate f_o:

$$(0.3^2 + 0.2^2 + 1^2 + 1^2 + 0.3^2 + 0.5^2 + 0.2^2 + 1^2)^{1/2}\ \% \approx 2\ \%$$

Table 4

Measurement of relation between fission rate in rig and in calibration position. Error estimate.

Parameter	Value	Error, 1 σ
Relation between number of counted delayed neutrons.	~ 1	
Statistical error (variation in neutron flux included in table 3).		≈ 0.4 %
Correction factor for dead time. About same correction in both positions.	~ 1	≈ 0
Correction factor for neutron energy dependance of delayed neutron yield.	1	≈ 0

Total error in relation: 0.4 %

Table 5

Measured and calculated distribution of power in BOCA rigs.

Core Pos	Hf shields Inner	Hf shields Outer	Co-wire activation measurement: Rel power density in rod pos 1	2	3	4	$\overline{\phi}_r/\overline{\phi}_{DND}$ 2.8 %	$\overline{\phi}_r/\overline{\phi}_{DND}$ 3.5 %	BUXY-CLUCOP calculations: $\overline{\phi}_r/\overline{\phi}_{DND}$ in enrichments 2.8 %	3.5 %	Difference calc - meas %
F1	1/4	1	1.093	.852	.925	1.130	.884				
F1	1	1	1.090	.860	.926	1.123	.911		.878		-3.6
I2	1	0	1.033	.951	.962	1.054	.890		.873		-1.9
I9	1	0	1.043	.933	.971	1.053		.848		.847	0
E10	1	1	1.080	.871	.942	1.107		.871		.852	-2.2
A7	1	0	1.084	.840	.902	1.174	.865			.847	-2.1
	0	0							.853		
	0	1							.863		

$\overline{\phi}_r$ = mean neutron flux density in the fuel rods (axial max).

$\overline{\phi}_{DND}$ = "- DND activation positions.

Table 5

Fig 1. Geometry of boiling capsule

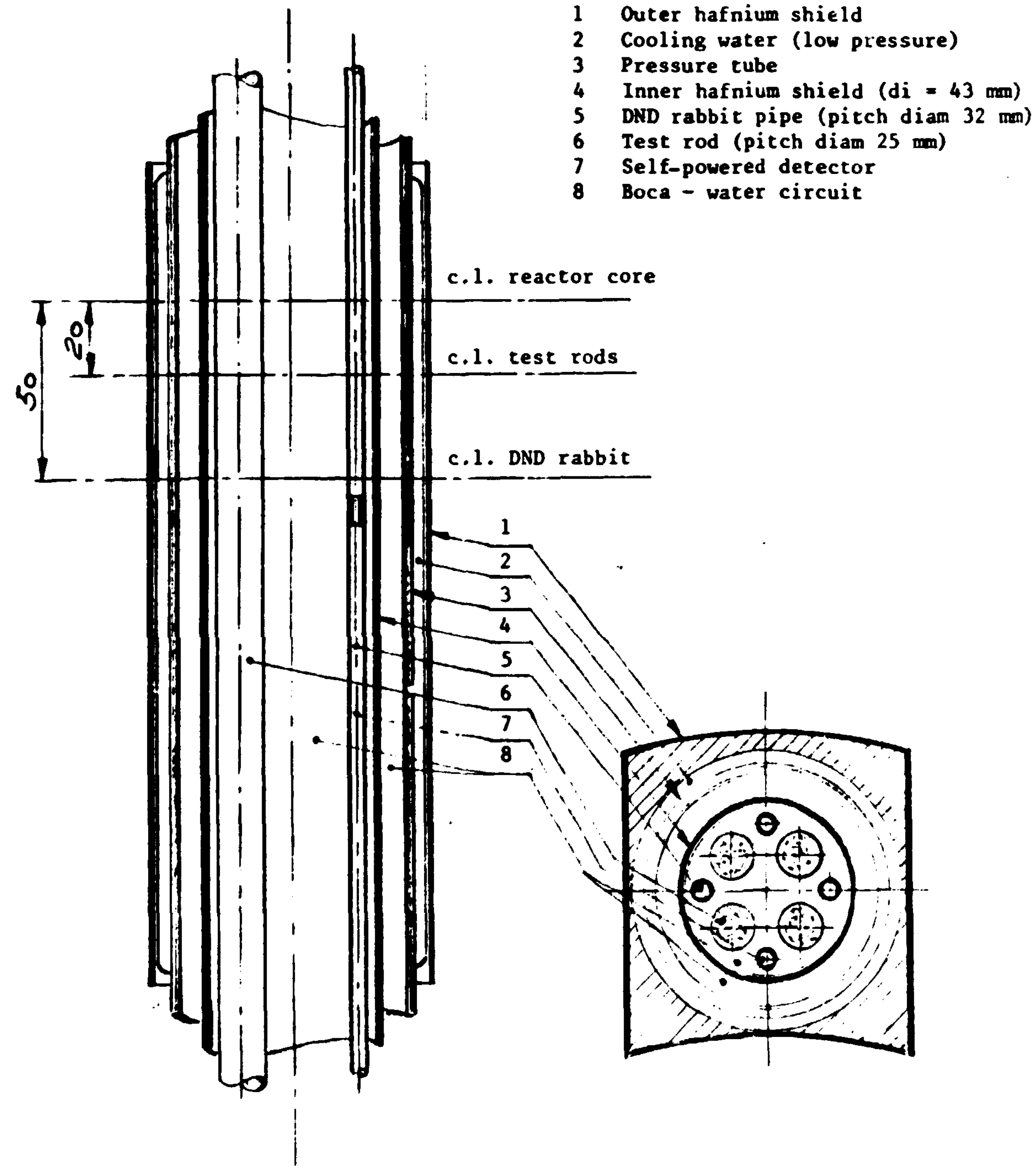
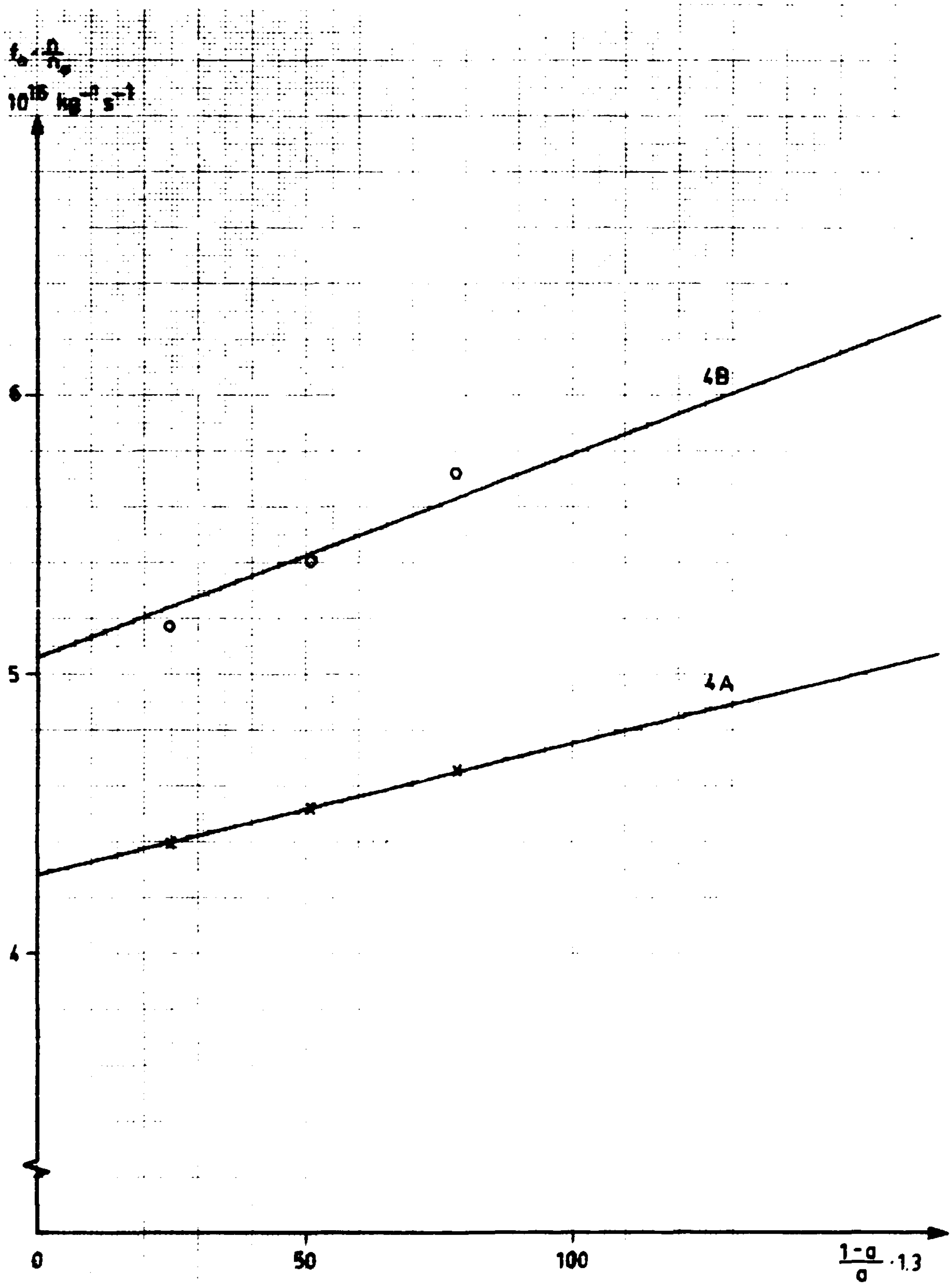


Fig 2. Example of DND measurement



Error estimate of gamma heating

The gamma heating has been measured in many positions of the R2 core using gamma thermometer, and a relation has been found between the power distribution around an experimental position and the gamma heating in stainless steel in that position. The measurements have been reported in ref 1.

This relation is routinely used for calculation of gamma heating distribution in every R2 core.

According to reference 1 the relation gives an error of 5.5 % (1 σ).

The influence of a 5 mm stainless steel shield was also measured. It was found to decrease the gamma heating by a factor of 0.78 ± 2 %.

The relation between gamma heating in lead and in stainless steel was also measured and found to be 1.37 ± 4 %.

In the boiling capsules the stainless steel tube walls are 3 mm thick. We have estimated, that this gives a shielding factor of 0.9 ± 5 % compared to the gamma thermometer rig.

For Zr cladding we estimate that the gamma heating is the same as in stainless steel within 1 %.

For UO_2 we use a factor 1.4 for gamma heating compared to stainless steel. We estimate the error in this factor to be 10 %.

This would give a total error in the gamma heating of $(5.5^2 + 5^2 + 10^2)^{1/2} \approx 12$ %.

Reference 1: R Carlsson and L G Larsson
Measurement and analysis of gamma heating in the R2 core.
AE-464 (1972).