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Publication date:
1980

Document Version
Publisher's PDF, also known as Version of record

RISØ-M-2185

CALCULATION OF HEAT RATING AND BURN-UP FOR TEST FUEL PINS IRRADIATED IN DR3

C. Bagger, H. Carlsen and K. Hansen

Abstract. A summary of the DR3 reactor and HP1 rig design is given followed by a detailed description of the calculation procedure for obtaining linear heat rating and burn-up values of fuel pins irradiated in HP1 rigs. The calculations are carried out rather detailed, especially regarding features like end pellet contribution to power as a function of burn-up, gamma heat contributions, and evaluation of local values of heat rating and burn-up. Included in the report is also a description of the fast flux- and cladding temperature calculation techniques currently used.

A good agreement between measured and calculated local burn-up values is found. This gives confidence to the detailed treatment of the data.

INIS Descriptors. BURN-UP, CALORIMETRY, COMPUTER CALCULATIONS, DR-3, FISSION, FUEL ASSEMBLIES, FUEL PELLETS, FUEL PINS, MATHEMATICAL MODELS, POWER DISTRIBUTION, RADIATION HEATING, URANIUM DIOXIDE, ZIRCALOY.

UDC 621.039.548 : 621.039.516.22

January 1980
Risø National Laboratory, DK 4000 Roskilde, Denmark.

ISBN 87-550-0655-8
ISSN 0418-6435

Risø Repro 1980

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1. INTRODUCTION

The research reactor DR3 at Risø has now been used during more than ten years for irradiation of test fuel pins manufactured at Risø. As it seems still more important to obtain accurate power histories for the irradiated assemblies, e.g. in case of high burn-up rods and power ramps, the procedure for power evaluation has been revised. This report presents the currently used procedure.

A summary of the reactor and rig design is given. Then follows a detailed description of the calculation procedure for obtaining the linear heat rating and burn-up of the fuel pins. These calculations involve some more details than previously, especially regarding features like end pellet contribution to power as a function of burn-up, gamma heat contributions, and evaluation of local values of heat rating and burn-up. Included is also a description of the existing fast flux- and cladding temperature calculations.

2. EXPERIMENTAL CONDITIONS

2.1. Reactor design and operation

The DR3 reactor, shown in Fig. 1, is a 10MW(th) research reactor moderated and cooled by heavy water. The core consists of 26 vertical fuel elements, each composed of concentric fuel tubes. Each fuel element initially contains highly enriched U-235 (120 or 150 g) alloyed with and clad in aluminium. The fueled length (i.e., the reactor core height) is 610 mm. A 50 mm diameter hole in the center of the fuel element may accomodate an irradiation rig.

The reactor core and heavy water are contained in an aluminium tank of 2 m diameter. Vertical and horizontal test holes of 100 and 175 mm diameter are located in the radial D_2O reflector. Outside the aluminium tank is a 300 mm graphite reflector with 100 mm vertical test holes.

The reactor is controlled by means of cadmium absorbers. There are seven coarse control arms and one fine control rod. The coarse control arms move like signal arms between the rows of fuel elements. The fine control rod and two safety rods are situated in corners of the core.

The normal reactor cycle comprises 2 Ms (23½ days) at full reactor power and 0.4 Ms (4½ days) shut-down for exchange of experiments and maintenance.

2.2. Irradiation rigs

Two rig types named HP1 and HP2 have been developed.

Cooling of HP1 rigs is performed by a cooling jacket connected to a special light water experiment cooling system, while cooling of HP2 rigs takes place through the reactor primary heavy water coolant circuit. This report only deals with irradiations in HP1 rigs.

The HP1 type rig is designed for operation in the hollow fuel elements. The HP1-MK II rig - Fig. 2 - has an aluminium alloy pressure vessel permitting operating pressures up to 8.2 MPa. The fuel pin is surrounded by a special Zr disc arrangement, acting as a flow guide for the primary cooling water. The heat developed in the fuel pin is transferred by local, natural convection to the pressure vessel wall. The pressure vessel is cooled by a secondary cooling jacket; the secondary cooling water flow rate and temperature rise are measured continuously, permitting an accurate determination of the total rig thermal output.

The HP1-MK III rig - Fig. 2 - has a Zr pressure vessel permitting operating pressures up to 15.3 MPa. The fuel pin is surrounded by a concentric riser tube. The primary cooling water circulates upwards along the fuel pin by natural convection and reverts outside the riser tube, where the heat is transferred through the pressure vessel wall to the secondary, metered coolant flow.

2.3. Outpile systems

The irradiation rigs are serviced by outpile systems as illustrated in Figure 3. A small flow, about 10^3 mm^3/s is continuously taken from the primary coolant for cleaning in an ion exchanger and then returned to the rig. Downstream from the rig, this flow passes a radiation monitor, which also serves as indicator for fuel pin cladding failure; the calculated flowtime from fuel pin to monitor is ~ 360 s. One outpile system can service up to four rigs in series. The primary coolant system is pressurized by means of He. In PWR-type irradiations, the pH is increased by addition of a predetermined $N_2 + H_2$ mixture to the He cover gas, thereby obtaining an $N_2 + 3H_2 \rightleftarrows 2NH_3$ equilibrium. Typical levels of conductivity and pH of the primary coolant are as follows:

Condition	BWR	PWR
Conductivity (μS/m)	50	1200
pH	5.8	10

As a result, only very thin surface deposits have been observed on fuel pins even for irradiations lasting several years. These deposits are easily removable in the hot cells. No firmly adhering crud layers have been observed.

2.4. Fuel pin assembly

An irradiation test usually consists of one, two or three fuel pins inserted as a whole assembly into the irradiation rig, (see Figure 4). A full-length pin ("maxi") normally has a fuel stack length of 500 mm; half-length ("midi", 210 mm) or third-length ("mini", 128 mm) pins are screwed together axially, using Zr dummy pins as required to make up the length corresponding to a maxi pin. The whole assembly is screwed onto a lifting rod (SS tube), suspended from a flange on the rig top. The bottom pin has a short nose piece fitting into the centering hole in the bottom of the rig pressure vessel. Pins longer than "maxi" can be accomodated geometrically by shortening the lifting rod; the upper part of the pin will then see lower neutron flux levels.

It is easy to remove a test from a rig during a shutdown and insert a new irradiation experiment. A movable lead box can be placed above a rig and a test pulled up for visual inspection. A test with intact cladding can be taken to the Risø Hot Cells in a shielded transport flask that will accomodate up to four tests at a time. After nondestructive examination, a test can be returned to the reactor for continued irradiation; prior to reassembling with a lifting rod the combination of individual pins may be changed as required. A failed fuel pin is unloaded from the reactor while remaining in the rig; the rig is then transferred to the hot cells and the fuel pin unloaded.

3. DETERMINATION OF RIG POWER

3.1. Calorimetry

The rig power is calculated from measured flow and temperature rise in the secondary cooling water:

$$Q[RIG] = \rho \cdot \dot{V} \cdot C_p \cdot \Delta T, \text{ Watt} \qquad (1)$$

where: ρ = density , kg/m^3
$\dot{V}$ = flow , m^3/s
C_p = specific heat , J/kg/K
ΔT = temperature rise , K

The flow measurement is made by means of the pressure drop over an orifice and the temperature rise is measured by differentially coupled Chromel-Alumel thermocouples in inlet and outlet.

3.2. Flux proportioning

For prediction purposes, estimates of rig power values can be obtained from neutron flux calculations (refs. 1,2) briefly to be described here.

For a given core position the mean thermal flux depends upon the content of U-235 in the surrounding driver element and the total content of U-235 in the core. At a reactor power of 10 MW and for a core content of 2485 g U-235, $\bar{\Phi}$[TH,CURVE,10 MW] is given in Figure 5, where the content of U-235 in the driver element at the beginning of the period and the position group are parameters. The thermal mean flux is then proportioned to the actual core content of U-235 (X g) using the expression

$$\bar{\Phi}[\mathrm{TH},\ 10\ \mathrm{MW}] = \bar{\Phi}[\mathrm{TH},\ \mathrm{CURVE},\ 10\ \mathrm{MW}] \cdot \frac{2485}{X} \qquad (2)$$

At reduced reactor power (Y MW) the mean thermal flux is calculated from the equation:

$$\bar{\Phi}[\mathrm{TH},\ Y\ \mathrm{MW}] = \bar{\Phi}[\mathrm{TH},\ 10\ \mathrm{MW}] \cdot \frac{Y\ \mathrm{MW}}{10} \qquad (3)$$

3.3. Reactor power

During start-up and operation measurement of the thermal flux by an ion chamber (MRLR-signal, Multiple-Range-Linear-Recorder) is used for reactor power control.

The total heat output of the core is measured calorimetrically with stationary temperature conditions in the reactor. After correction of the measurement for the power added by circulation pumps this calculated reactor power is used for calibration of the MRLR signal.

4. EVALUATION OF TEST PARAMETERS

4.1. Heat rating and burn-up

The heat balance of a rig with a test fuel assembly during irradiation is shown in Figure 6. The total amount of heat generated (Q[RIG]) is removed by cooling water and determined calorimetrically as described in section 3.1. Heat is generated in the test fuel from fissions (31.4 ± 0.3 pJ/U235 fission and 32.4 ± 0.3 pJ/Pu239 fission, see Appendix B). The gamma field from surrounding fuel elements induces heat in the various assembly- and rig components, the amount depending upon irradiation position in the core and core power.

Evaluation of burn-up distribution and heat rating is based upon Q[FISS] as shown in Figure 7, which gives a general view of the sequence of calculations leading to a desired resulting value. Q[FISS], which is the power due to fissions in the fuel, is found from

$$Q[FISS] = Q[RIG] - Q[\gamma] \ . \tag{7}$$

$Q[\gamma]$ is determined from

$$Q[\gamma] = (Q[\gamma,FUEL] + Q[\gamma,METAL] + Q[\gamma,LIFT.ROD] + Q[\gamma,RIG\ MAT'LS]) \cdot CPF \qquad (8)$$

$Q[\gamma,FUEL]$ is the product of fuel weight W[FUEL] in the test fuel assembly and the specific gamma heat $G[\gamma,FUEL]$ relevant for the actual irradiation position at a core power of 10 MW (Table 1). $Q[\gamma,METAL]$ is the product of the weight of metal parts, W[METAL], constituting the assembly, i.e. tubing, plugs, coupling pieces, nose cone and in some cases dummy pins,and the specific gamma heat $G[\gamma,METAL]$ relevant for the actual irradiation position at a core power of 10 MW (Table 1).

$Q[\gamma,LIFT.ROD]$ is estimated to 50 W at a core power of 10 MW.

$Q[\gamma,RIG\ MAT'LS]$ is the experimentally measured heat output of the rig without the test fuel assembly minus the power generated by gamma absorption in the water column replacing the assembly

$$Q[\gamma,RIG\ MAT'LS] = Q[\gamma,RIG] - Q[\gamma,H_2O] \qquad (9)$$

$Q[\gamma,RIG]$ and $Q[\gamma,H_2O]$ are found for the relevant irradiation positions from Table 1 and apply to 10 MW core power.

CPF is a correction factor to be applied for core power $\neq$ 10 MW, as the gamma field in the core is assumed to vary linearly with core power.

Q[FISS] may thus be calculated from

$$Q[FISS] = Q[RIG] - (Q[\gamma,RIG] - Q[\gamma,H_2O] + Q[\gamma,LIFT.ROD] + Q[\gamma,METAL] + Q[\gamma,FUEL]) \cdot CPF \qquad (10)$$

Q[FISS] is a measure of the fission rate and may therefore be used directly in the calculation of the burn-up distribution.

As for the heat rating calculations, the total amount of heat transported in the fuel material, Q[FISS] from fission plus Q[γ,FUEL] from absorption of gamma radiation from the external field, has to be considered.

The heat rating P[FUEL] may therefore be written in the general form

$$P[FUEL] = P[FISS] + P[\gamma,FUEL] \cdot CPF \qquad (11)$$

In cases, where the heat rating values are to be used in hydraulic calculations, the heat transport to the cooling water also involves the heat due to absorption of gamma radiation from the external field in the cladding tube material.

4.1.1. End pellet effect

The ratio RP[E/C] between fission densities of end pellets with initial enrichment E% and central pellets with initial enrichment C% varies with burn-up because of different rates of Pu build-up.

Reactor physics calculations of RP[E/C] as a function of burn-up of the enriched pellet, BU[C], has been carried out for two enrichment combinations E/C = 0.71%/1.5% and E/C = 0.71%/2.28% (Appendix 1).

Because other combinations of enrichments are encountered there is a need to extend the calculations to an arbitrary combination of low enrichments.

The following calculation technique is based upon the assumption that the fraction of the total number of fissions in the driver-element plus test fuel, which occurs in the test fuel pellets at an arbitrary infinitesimal time interval, is a linear function of the initial enrichment of the pellets.

From Table 1, Appendix A, an approximated linear function is derived for each of the irradiation times 0, 30, 60, 120, 180, 240, 300, 360 and 420 days:

$$\% \text{ fission} = F(\text{enrichment}) \qquad (12)$$

The functions are plotted in Figure 8.

Similarly for each irradiation time the approximated linear function

$$\text{Burn-up} = F(\text{enrichment}) \qquad (13)$$

is derived. The functions are plotted in Figure 9.

From these figures, nine values of

$$RP[E/C] = \frac{\%\ Fiss[E\%]}{\%\ Fiss[C\%]} \qquad (14)$$

as a function of BU[C] may be derived for the arbitrary enrichment values E = Ø and C = Z in the following way:

From Figure 8 (% fission [Ø]) and (% fission[Z]) are found for each value of irradiation time and

$$RP[Ø/Z] = \frac{\%\ Fiss[Ø\%]}{\%\ Fiss[Z\%]} \qquad (15)$$

is calculated.

From Figure 9 BU[Z] is found for each of the corresponding irradiation times.

In this way nine corresponding values of RP[Ø/Z] and BU[Z] are obtained.

To obtain a mathematical expression

$$RP[E/C] = F(BU[C]) \qquad (16)$$

the curve points are fitted with an expression of the general form

$$y = a \cdot x^b + c \text{ , } y \leq 1 \tag{17}$$

When y according to this equation is found > 1, it is set equal to 1, as RP[E/C] cannot exceed 1 when end peaking effects are neglected.

For BU[C] values obtained in irradiation times between 0 and 30 days a linear function is used.

Expressions have been formed for the two combinations of enrichment of Appendix A, 0.71%/1.5% and 0.71%/2.28%, and the calculation technique described above has been used to make expressions valid for the enrichment combination 2.28%/3.16%.

Values of a, b and c are given in Table 3 together with burn-up limits of validity.

4.1.2. Calculation of assembly values

The average assembly fuel heat rating is

$$P[FUEL,ASS'Y] = \frac{Q[FISS] + Q[\gamma,FUEL] \cdot CPF}{L[FUEL]} \tag{18}$$

The accumulated burn-up at any time is the sum of burn-up increments - BUI[ASS'Y] - from preceeding irradiation periods:

$$BU[ASS'Y] = \sum_i \frac{Q[FISS,i] \cdot IT[i]}{W[FUEL,ASS'Y]} \tag{19}$$

The heat rating due to fission in central pellets alone is important for later calculations leading to local values.

The heat load of enriched pellets is

$$P[FISS,C] = \frac{Q[FISS]}{L[ASS'Y,C] + L[ASS'Y,E] \cdot RP[E/C]} \text{ ,} \tag{20}$$

where RP[E/C] has to be calculated by iteration with

$$BUI[ASS'Y,C] = \frac{P[FISS,C] \cdot L[FUEL,C] \cdot IT}{W[FUEL,C]} \tag{21}$$

and

$$BU[ASS'Y,C] = \sum_i BUI[ASS'Y,C,i] \tag{22}$$

BU[ASS'Y,C,i-1] is used for a first calculation of RP[E/C] (section 4.1.1.).

With this preliminary RP[E/C] value in equation (20) a preliminary P[FISS ,C] is obtained which can be used for calculating

$$RP[E/C] = a\ (BU[ASS'Y,C,i] + \frac{BUI}{2}[ASS'Y,C,i])^b + c \tag{23}$$

according to eqs. 17,21 and 22 (a, b and c to be chosen according to BU[C], see Table 3).

A new RP value based upon the improved RP-value of equation 23 is used for determining an improved P[Fiss,C], etc.

The iteration is continued until subsequent values of RP[E/C] differ by less than 10^{-3}.

4.1.3. Calculation of local values

The distribution of heat rating and burn-up within an assembly is based on gamma scans. Because of the increase of RP[E/C] with burn-up scanning curve areas corresponding only to central pellets are used.

Cs-137 scans are normally used, but may in special cases be replaced by scans of other isotopes.

Because gamma scans represent an integrated number of fissions, axial depletion effects and flux shape changes contribute to the measurement with their time averaged effect. Thus, the heat rating and burn-up distributions will not be fully correct. A comparison of scans of long-lived isotopes with those of short lived isotopes indicate, however, that the error is less than 5%. In cases where drastic changes in flux shape are expected, e.g. where assemblies to be ramp tested are moved from their preconditioning irradiation position to the ramp position, intermediate gamma scanning is carried out. The scanning curves thus obtained are used for the distribution calculation corresponding to the pre-ramp irradiation. After ramp-testing scanning on short lived isotopes, e.g. Ba/La140, yields the scans valid for distribution calculation corresponding to the ramp situation.

Figure 10 shows a typical schematic gamma scan shape for three pins constituting an assembly. The sum of the three hatched curve parts is equivalent to the integrated number of fissions in the enriched pellets.

The average height H[C] of the three curve parts over the hatched area therefore represents the average linear heat load of enriched pellets P[FISS,C], as calculated in section 4.1.2.

The heat rating at an arbitrary axial position of enriched pellets may be calculated from the corresponding gamma scan height H[X], using

$$P[\mathrm{FUEL,POS\ X}] = \frac{H[X]}{H[C]} \cdot P[\mathrm{FISS},C] + P[\gamma,\mathrm{FUEL}] \cdot \mathrm{CPF} \qquad (24)$$

The burn-up in the position is found from BU[ASSY,C], (section 4.1.2.) as

$$BU[\mathrm{POS\ X}] = \frac{H[X]}{H[C]} \cdot BU[\mathrm{ASS'Y},C] \qquad (25)$$

The determination of heat rating and burn-up of an end pellet (pos.y) necessitates calculation of BU[POS X] for all preceding irradiation periods for the adjacent enriched pellet in pos.x.

With

$$BU[\mathrm{POS\ X}] = BU[\mathrm{POS\ X},i-1] + \frac{BUI[\mathrm{POS\ X},i]}{2} \qquad (26)$$

RP[E/C,i] for the pellet pair is similarly determined according to section 4.1.1., whereupon

$$P[\mathrm{FUEL,POS\ Y}] = \frac{H[X]}{H[C]} \cdot P[\mathrm{FISS},C] \cdot RP[E/C] + P[\gamma,\mathrm{FUEL}] \cdot \mathrm{CPF} \qquad (27)$$

and

$$BU[POS\ Y,i] = \sum_i BUI[POS\ X,i] \cdot RP[E/C,i] \qquad (28)$$

4.1.4. Calculation of pin average values

Calculation of pin average values may be carried out easily, provided that

$$\frac{P[E,PIN\ X]}{P[E,PIN\ Y]} = \frac{P[C,PIN\ X]}{P[C,PIN\ Y]} \qquad (29)$$

at any period in time during irradiation.

This is usually the case when the pins of an assembly are identical and the flux profile is flat.

The ratio $\frac{A[Z,C]}{A[C]}$ between the gamma scan are corresponding to the enriched pellets of pin Z and the sum of these areas for all pins in the assembly may then be used to distribute Q[FISS] among the pins to give P[FUEL,PIN]:

$$P[FUEL,PIN\ Z] = \frac{Q[FISS] \cdot A[Z,C]}{L[FUEL,PIN\ Z] \cdot A[C]} \qquad (30)$$

and for BU[PIN.Z]

$$BU[PIN\ Z] = \sum_i \frac{Q[FISS,i] \cdot IT[i]}{W[FUEL,PIN\ Z]} \cdot \frac{A[Z,C]}{A[C]} \qquad (31)$$

If

$$\frac{P[E,PIN\ X]}{P[E,PIN\ Y]} \neq \frac{P[C,PIN\ X]}{P[C,PIN\ Y]} \qquad (32)$$

the calculation of pin average values requires the calculation of Q[FISS,E,PIN Z] from

$$Q[FISS,E,PIN\ Z] = P[FISS,E1,PIN\ Z] \cdot L[FUEL,E1,PIN\ Z] + P[FISS,E2,PIN\ Z] \cdot L[FUEL,E2,PIN\ Z] \qquad (33)$$

and Q[FISS,C,PIN Z] from

$$Q[FISS,C,PIN\ Z] = Q[FISS,C] \cdot \frac{A[Z,C]}{A[C]} \tag{34}$$

$$P[FUEL,PIN\ Z] = \frac{Q[FISS,C,PIN\ Z] + Q[FISS,E,PIN\ Z]}{L[FUEL,PIN\ Z]} + P[\gamma,FUEL] \cdot CPF \tag{35}$$

Similarly for the burn-up

$$BU[PIN\ Z] = \sum_i \frac{(Q[FISS,C,PIN\ Z,i] + Q[FISS,E,PIN\ Z,i]) \cdot IT[i]}{W[FUEL,PIN\ Z]} \tag{36}$$

4.2. Fast flux

The fast flux in a fuel element position depends on the distance of the position from the core center and on the U235-content of the fuel element in the position. Actual flux levels are obtained from activation of a Ni-wire, which is used as a monitor for neutrons with energies in the range 0.016-1.160 pJ (0.1-10 MeV) and with a maximum at 0.320 pJ (2 MeV), see ref. 3.

The actual fast flux levels in the fueled length of the cladding are obained ass follows:
From Figure 5 (lower part) the flux level for a given position is found taking into account the U235 content in the driver element; this flux level applies to the driver element position without a rig, and is independent of the total U235-content in the core.

It must be corrected by the flux contribution from fissions in the fuel inside the cladding (ref. 11).
For pins with 12.7 mm fuel diameter and 1.5% enrichment:

$$\phi_f = \phi_{fig.5} \times 1.64 \tag{37}$$

For pins with 9.3 mm fuel diameter and 3.2% enrichment:

$$\phi_f = \phi_{fig.5} \times 2.02 \tag{38}$$

4.3. Cladding surface temperature

Subcooled boiling on the cladding surface begins at surface heat fluxes around 500 kW/m^2 (slightly different for BWR and PWR pressures, in both cases corresponding to about 20 kW/m). Above this level, the cladding surface temperature is given as the coolant saturation temperature minus a depression (estimate: 4 K) due to dissolved He, plus the wall superheat obtained from the Jens and Lottes equation (ref. 4). The 4 K estimate of depression corresponds to an assumed 50% saturation of He. The equations used at BWR (7.2 MPa) and PWR (15.3 MPa) system pressures are as follows:

At 7.2 MPa:

$$T = 554 + 1.45 \times Q^{0.25} \tag{39}$$

and at 15.3 MPa:

$$T = 610 + 0.40 \times Q^{0.25} , \tag{40}$$

giving temperatures in K, when Q, the surface heat flux, is given in kW/m^2.

4.4. Comparison of calorimetric and radiochemical burn-up determination

The accuracy of the calorimetrically determined local burn-up may be found by comparison with the chemically determined burn-up values. Burn-up values are given in refs. 6 and 7, where corrections are applied - except for one case - for difference in energy per fission and for difference in fission yield of the burn-up monitor Nd148 for fissions of U and Pu.

The standard deviation of the experimental burn-up is 2.2% (ref. 5).

The results are compared in Table 4, where an excellent agreement is found for four of the five cases. The relatively large deviation for the last case may be attributed to the absence of the necessary data for making the above mentioned corrections.

5. CONCLUSION

The calculation of linear heat rating and burn-up of test fuel pins in HP1 rigs, irradiated in DR3 at Risø, has been described in detail together with the calculation of fast flux in the cladding and the cladding surface temperature.

A good agreement between measured and calculated local burn-up values has been found. This gives confidence to the detailed treatment of the data.

6. ACKNOWLEDGEMENT

The authors thank Messrs. Per Knudsen and Niels Rhod Larsen for contributing to several stimulating discussions.

7. REFERENCES

1) WIIG, P. (September 1970). "Thermisk og hurtig neutronflux i DR3 med MkIV kerne". Internal Risø Report, 00/n-104.
2) WIIG, P. (April 1971). Appendix 2 to Internal Risø Report 18/N-104.
3) HAACK, K. (1972). "Nuclear Heat Dose Rate Separation by Calculation and Measurement". Risø Report No. 119.
4) JENS, W.H. and LOTTES, P.A. (1951). "Analysis of Heat Transfer Burn-Out, Pressure Drop and Density Data for High Pressure Water". ANL-4627.
5) LARSEN, N.R. and LARSEN, E. (1979). "Isotope Analysis, Radiochemical Burn-up Determination". Risø-M-2183.
6) KNUDSEN, P. and BAGGER, C. (1978). "Power Ramp and Fission Gas Performance of Fuel Pins M20-1B, M2-2B and T9-3B". Risø-M-2151.
7) BAGGER, C., CARLSEN, H. and KNUDSEN, P. (1978). "Details of Design, Irradiation and Fission Gas Release for the Danish UO_2-Zr Irradiation Test 022". Risø-M-2152.
8) ANDERSON, A.R. and WAITE, R.J. (1961). "Energy Absorption from Reactor Radiation in the Heavy Water Moderated Reactors DIDO and PLUTO". AERE-C/R-2713.
9) LETH, J-A. (1967). "Beskrivelse af forsøg nr. 108. Højtryksrig for Mk-III element". Reaktorforsøgssektionen, Intern rapport nr. 21, appendix 2.
10) HANSEN, K. (1975). "Måling af varmebidraget hidrørende fra afsætning af γ-varme samt nedbremsning af hurtige neutroner i HP1-riggene". Forsøgssektionen, Intern rapport F-6-108-6.
11) KRISTIANSEN, G.K. (April 1975). "Calculation of Fast Flux in HP-rigs". Reactor Physics Section, Internal Report RP-2-75.

Table 1. γ-Capture Constants.

Values at 10 MW Core Power								
Irradiation position	Irradiation group position	Q[γ,RIG] a)		Q[γ,H_2O] b)		Q[γ,LIFT.ROD] c)	G[γ,METAL] c)	G[γ,FUEL] c)
		HP1,Mk II kW	HP1,Mk III kW	BWR kW	PWR kW	kW	W/g	W/g
C3 C4	1	6.8	8.4				1.49	1.93
B3 B4 D3 D4	2	6.1	7.7				1.30	1.68
C2 C5	3	5.6	7.1				1.16	1.49
B2 B5 D2 D5	4	5.1	6.5	0.20	0.11	0.05	1.03	1.32
A2 A3 E2 E3	5	4.7	6.1				0.94	1.22
A1 A4 C1 C6 E1 E4	6	4.2	5.3				0.80	1.03
B1 B6 D1 D6	7	3.9	4.9				0.74	0.95

a) Ref. 10

b) Ref. 9

c) Based on refs. 3 and 8

Table 2. Weight of assembly details.

	W[COUPLING PIECE] g	W[NOSE PIECE] g
14 mm O.D. (BWR)	15.3	5.0
10.7 mm O.D. (PWR)	4.8	3.8

Table 3. Numerical constants and burn-up limits for RP[E/C] = a • BU[C]^b + c.

E/C	a	b	c	BU[C] limits GJ/kg U
0.71%/1.5%	$1.12 \cdot 10^{-3}$ 0.3743 1	1 0.1242 0	0.552 0 0	0 - 70 70 - 3013 3013 - ∞
0.71%/2.28%	$8.80 \cdot 10^{-4}$ 0.1916 1	1 0.2033 0	0.401 0 0	0 - 92 92 - 3387 3387 - ∞
2.28%/3.16%	$1.70 \cdot 10^{-4}$ 0.504 1	1 0.0818 0	0.750 0 0	0 - 121 121 - 4342 4342 - ∞

Table 4. Comparison of local burn-up values.

Pin/sample	Location	Exp.BU,GJ/kg U	Calc.BU,GJ/kg U	Rel.diff,%
PA29-4/3	central	4011	3997	+0.4
PA29-4/6	end	3388	3402	-0.4
M2-2C/4	central	3591	3586	+0.1
M2-2B/5	central	2327	2335	-0.3
M20-1E/4	central	2797	2659	+5.2

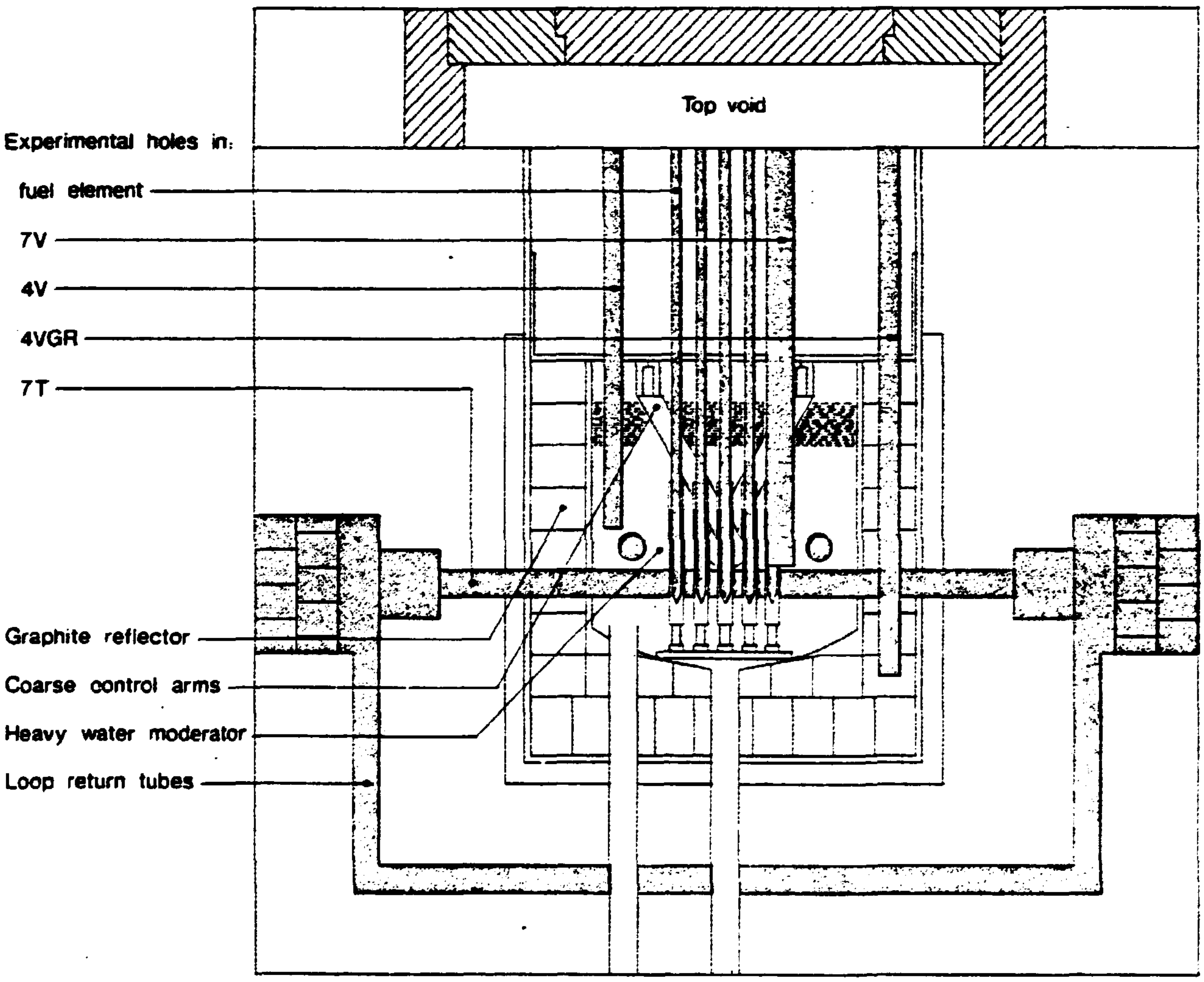


Fig. 1 A. Vertical cross section of DR3.

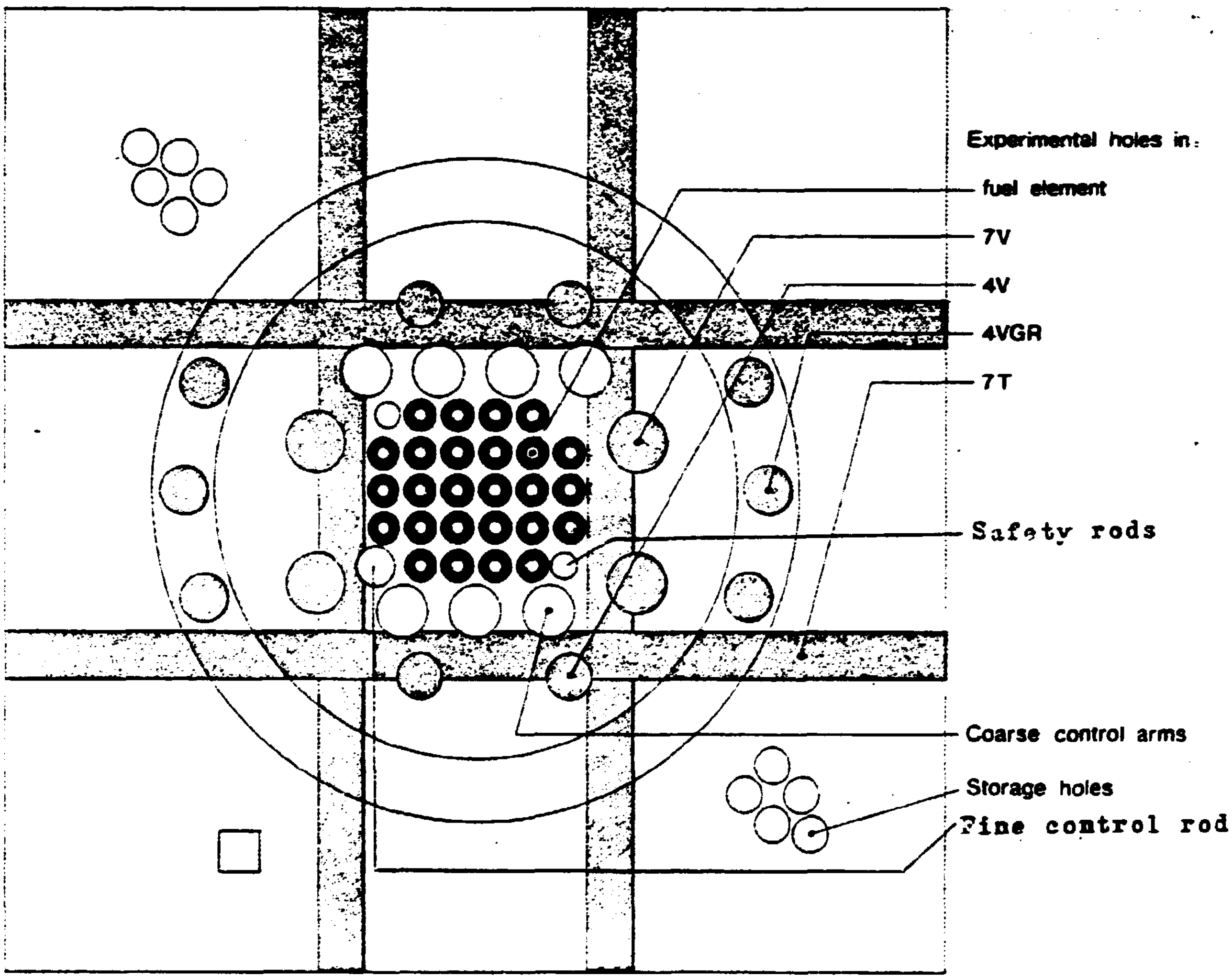


Fig. 1 B. Horizontal cross section of DR3.

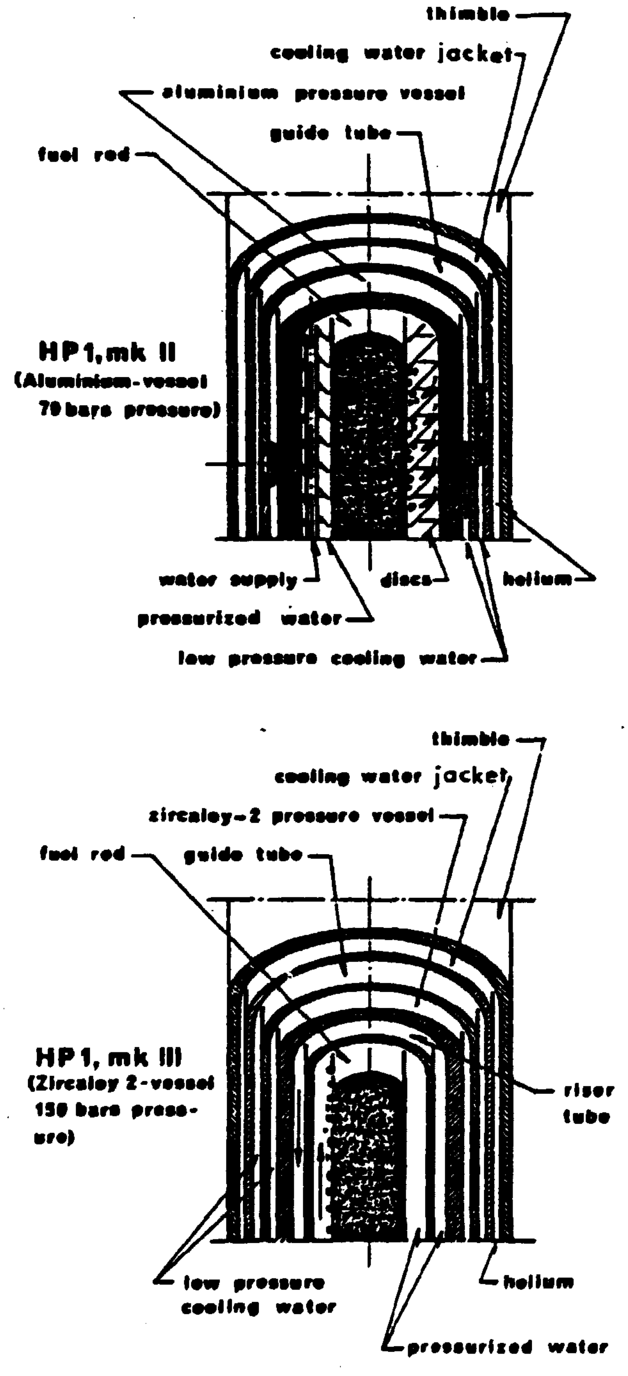
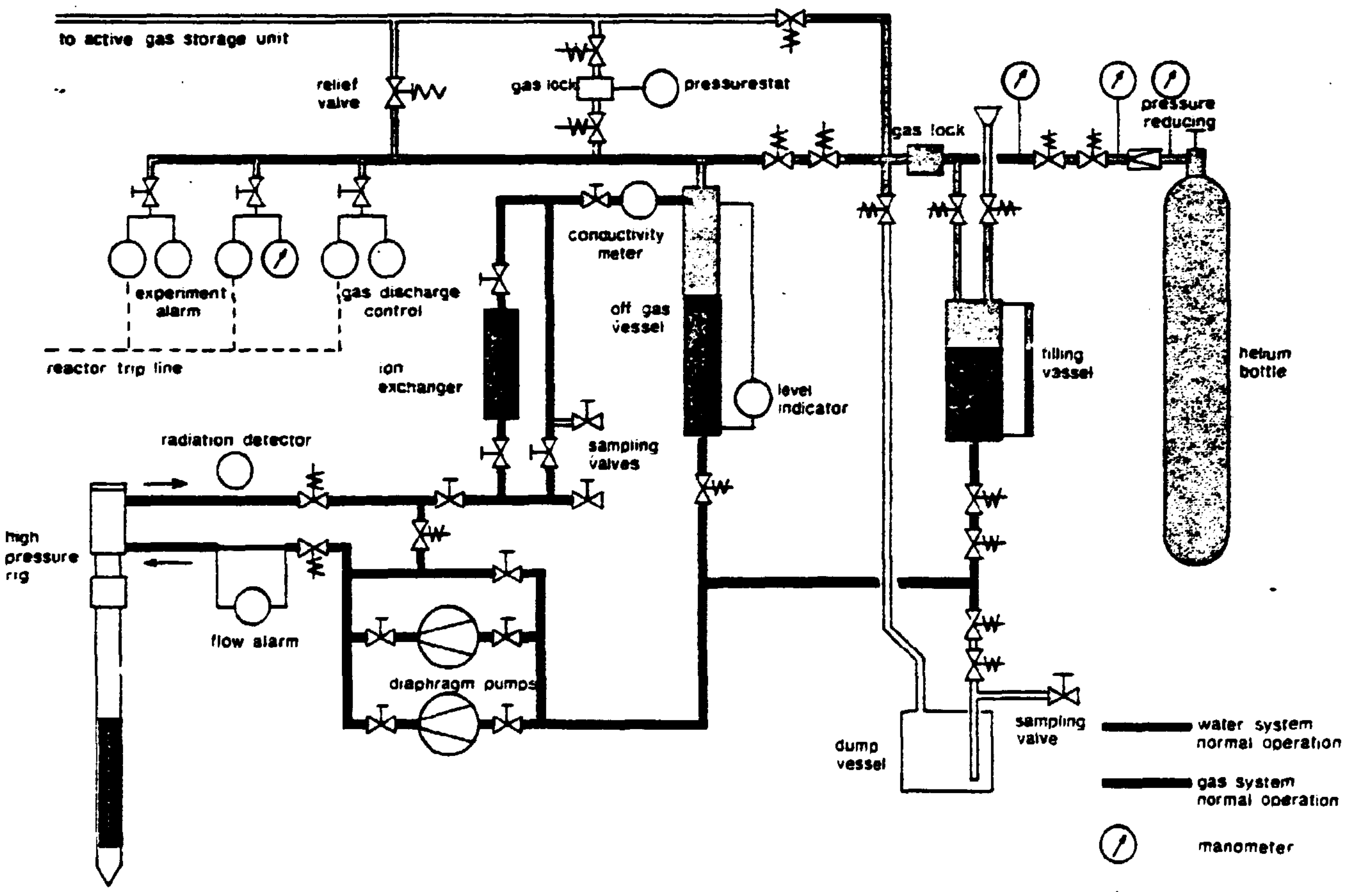


Fig. 2. HP1 high pressure rigs.



<u>Fig. 3.</u> High pressure circuit for HP1 rigs.

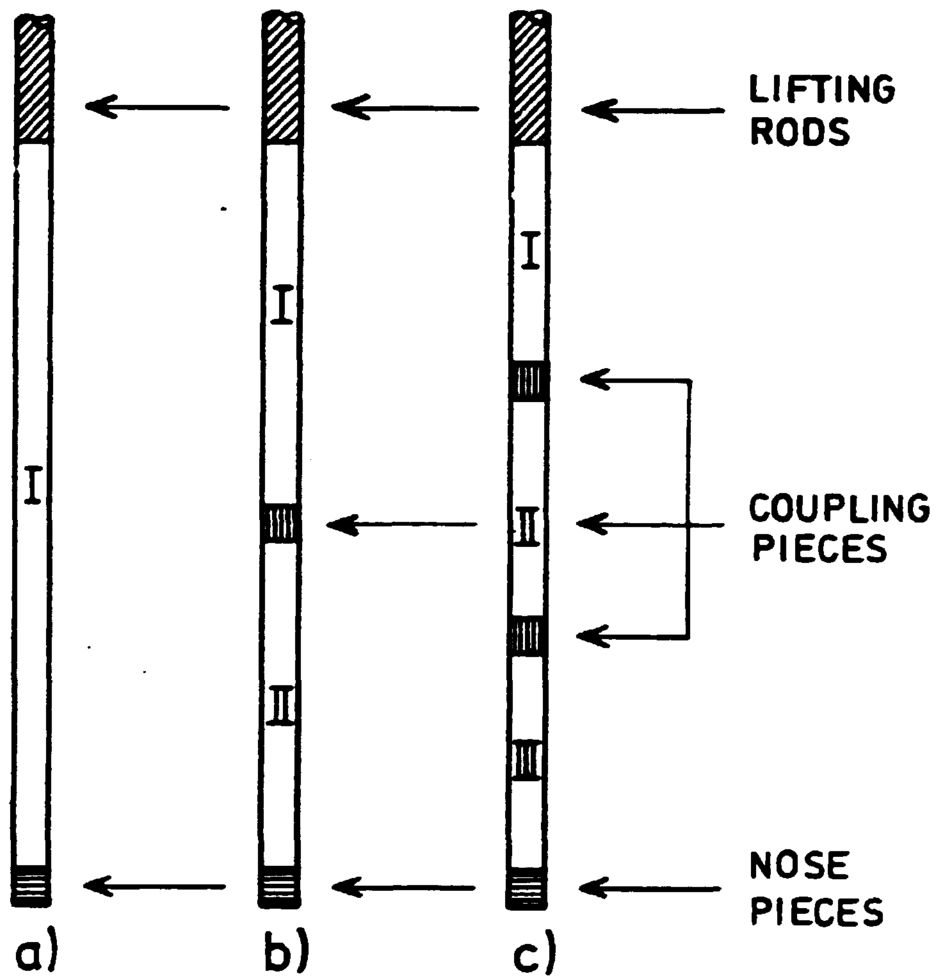


Fig. 4. Drawing of fuel pin assemblies with
a) 1 maxi, b) 2 midi and c) 3 mini fuel pins

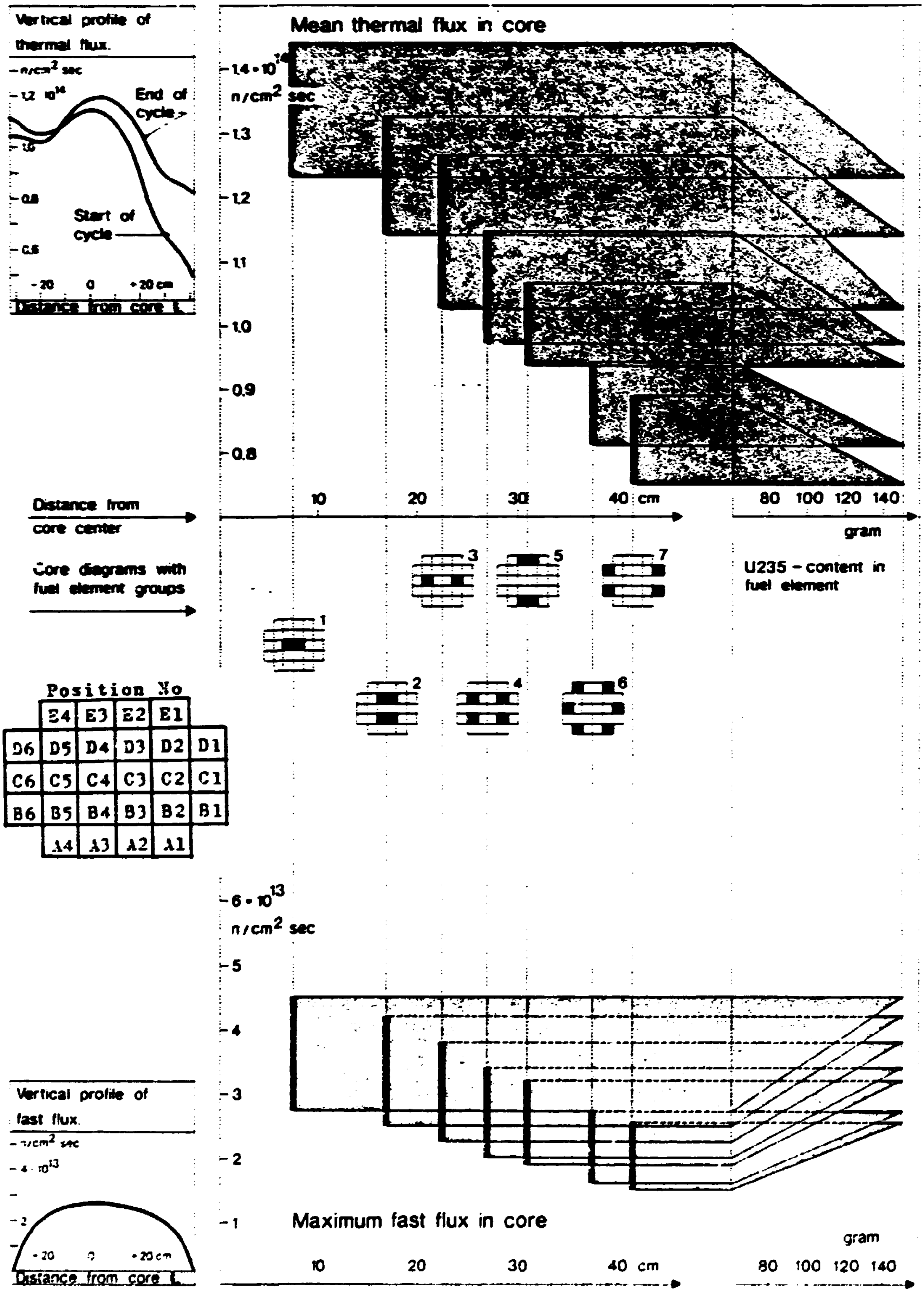


Fig. 5. Thermal and fast flux levels in DR3.

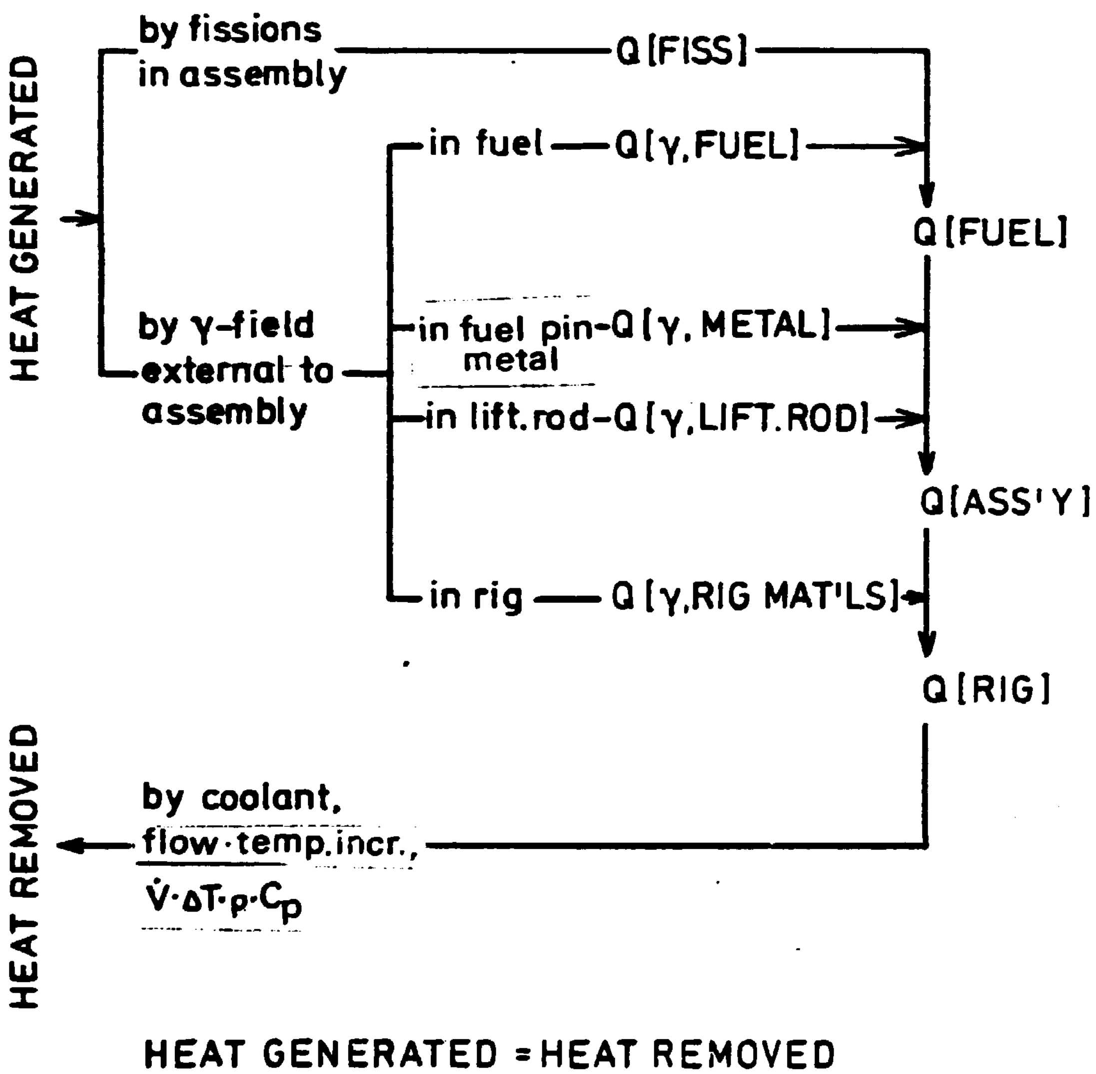


Fig. 6. Heat balance of rig with fuel assembly

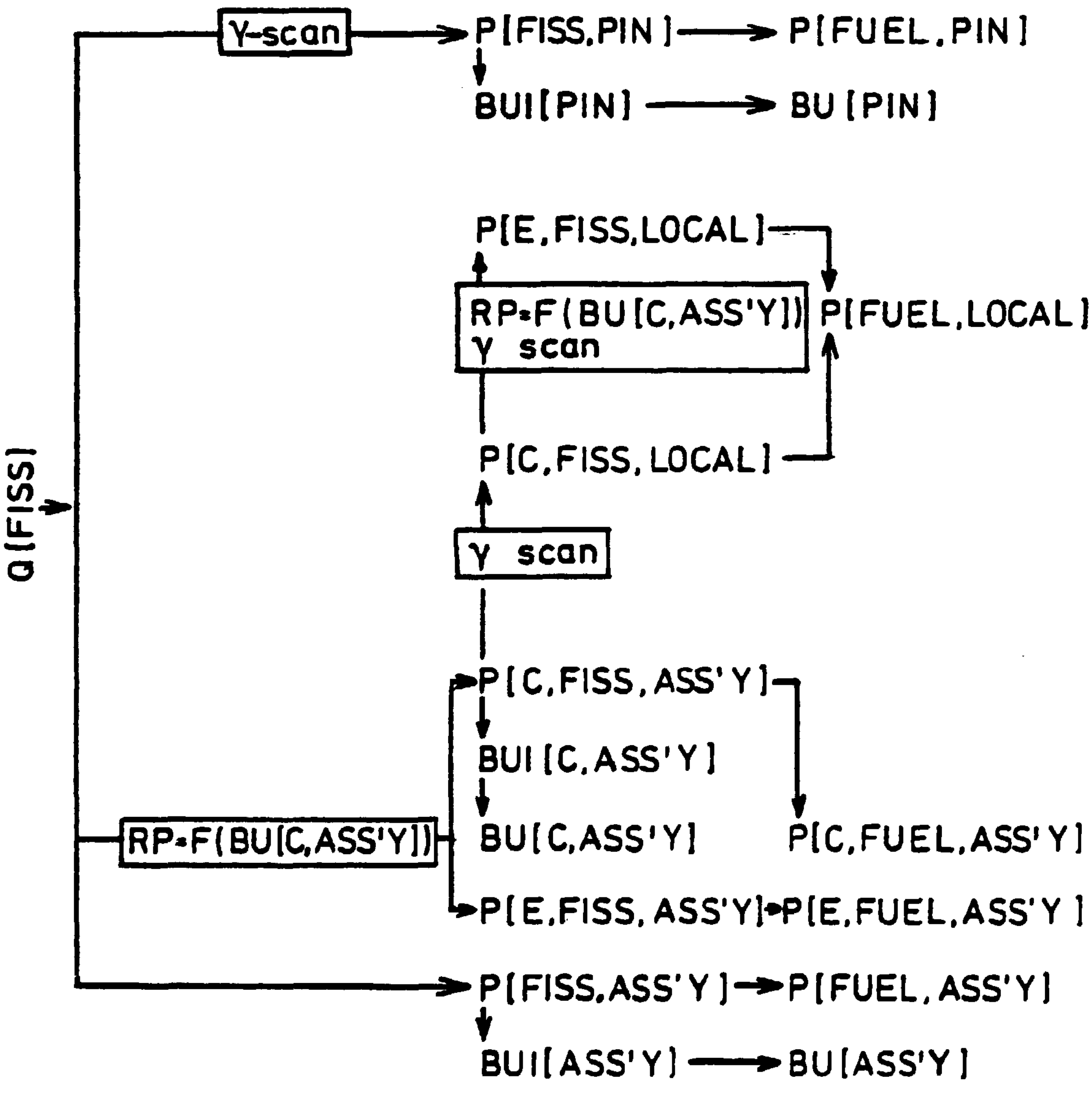


Fig. 7. Calculation of test values

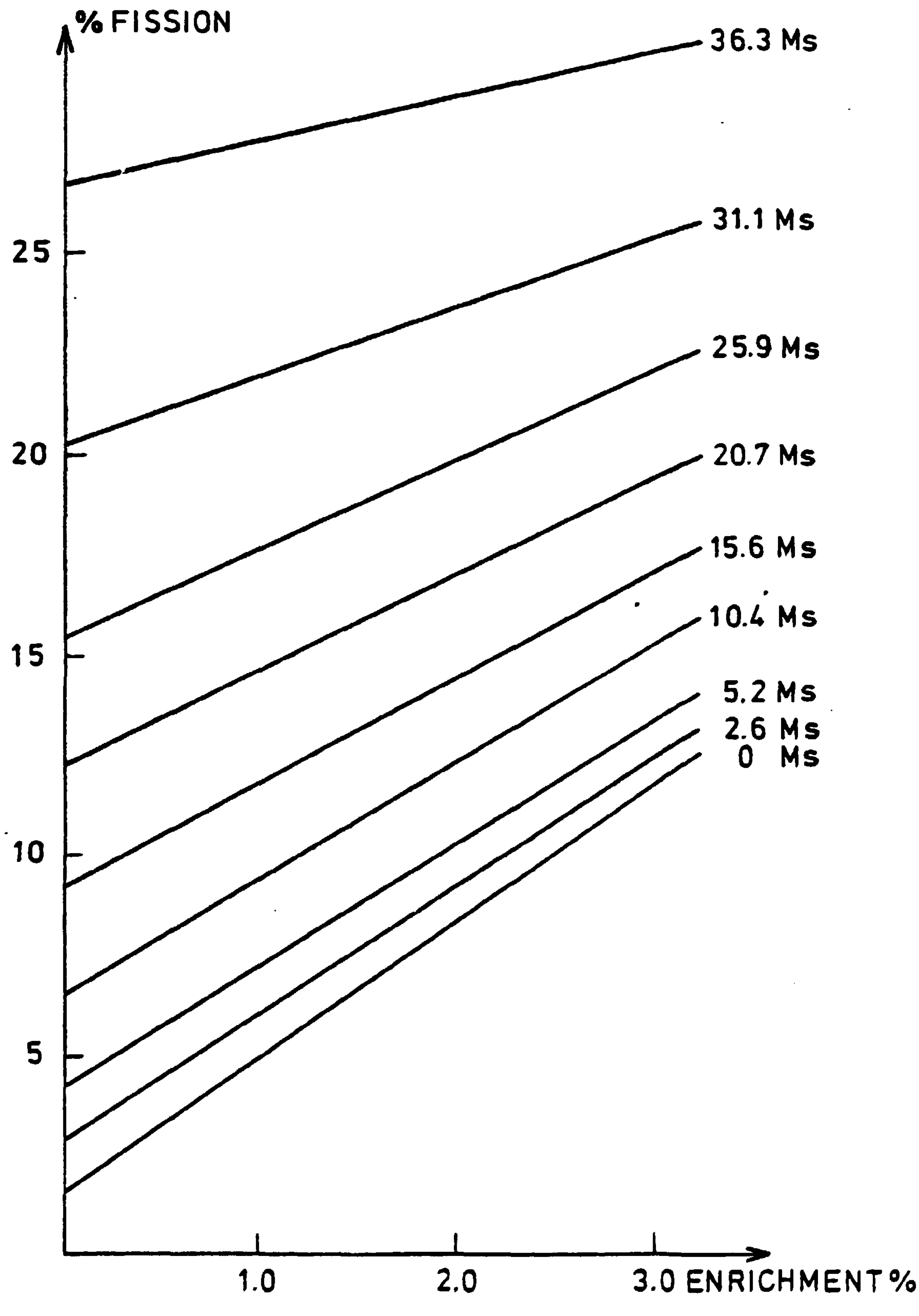


Fig. 8. % fission = F (enrichment, irr. time)

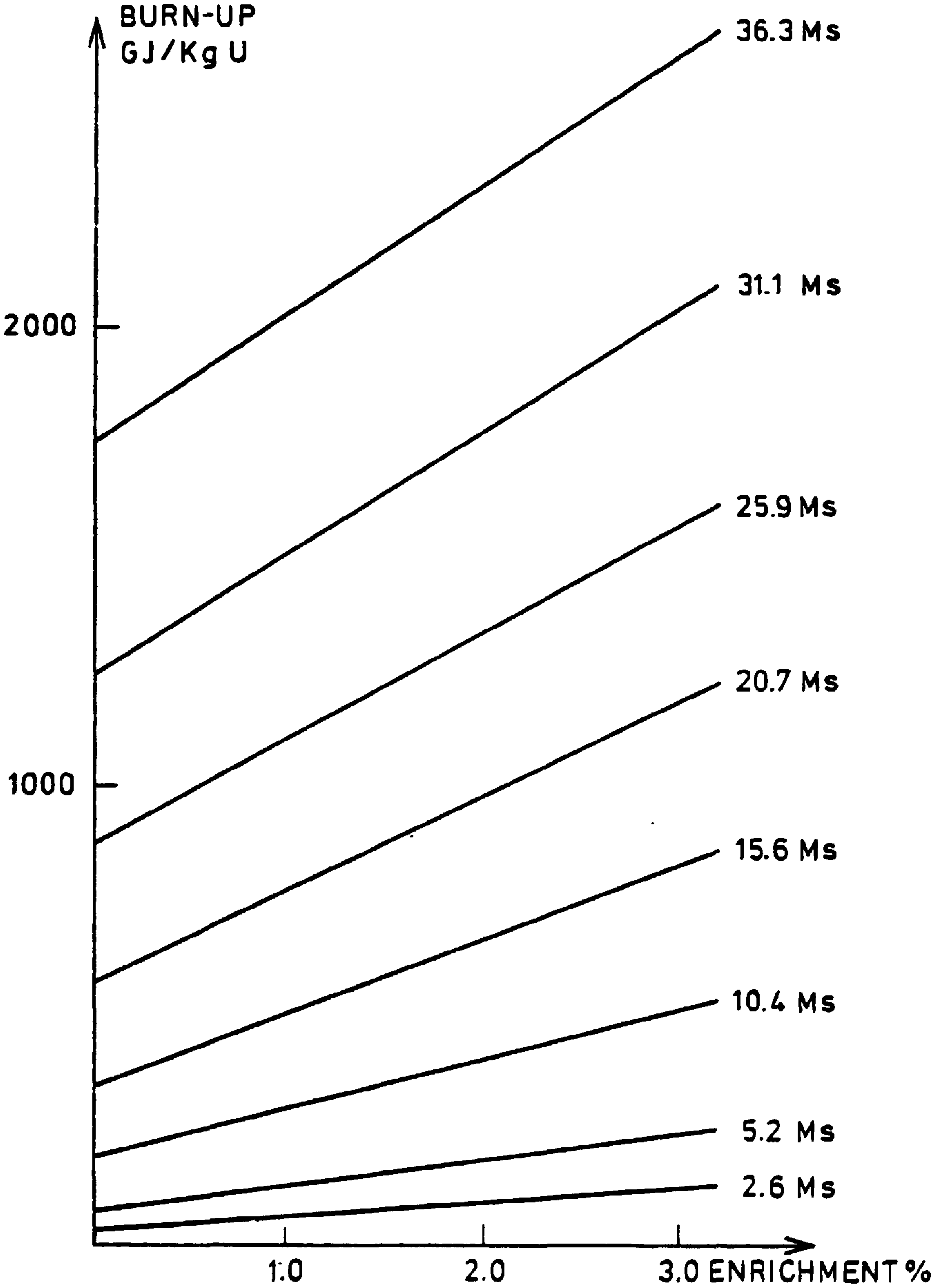


Fig. 9. Burn-up = F (enrichment, irr. time)

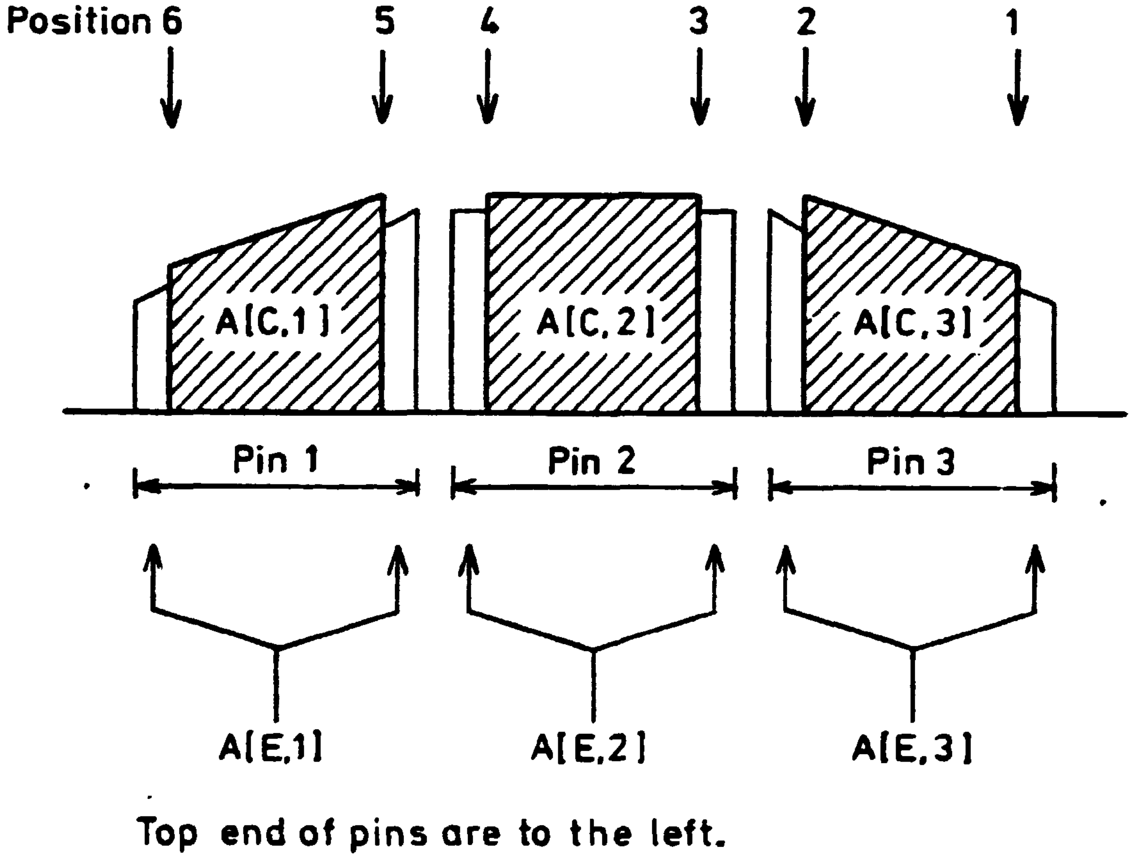


Fig. 10. Typical schematic gammascan shape for 3 pins irradiated together.

APPENDIX A

POWER RATIOS FOR DIFFERENT ENRICHMENTS AND PELLET DIAMETERS

F. Højerup, T. Petersen
Reactor Technology Department, Risø

When a fuel pin with low enrichment is placed in the centre of a highly enriched fuel element it may be assumed that the neutron current into the fuel pin is determined essential by the properties of the highly enriched fuel element. Thus the current into the fuel pin will be constant along the entire length of the pin even if it consists of fuel pellets with different enrichments. The approximation may be expressed as $\Sigma_a \phi$ = constant, where Σ_a is the average absorption cross section and ϕ the average flux of the fuel pin.

According to the approximation there is no axial coupling between zones of different enrichment in the fuel pin. The power/cm of a zone in the fuel pin may be calculated as the power/cm in an infinite cylinder having the same geometric cross section as the fuel pin and fuel element. The ratio between the power of zones of different enrichments in the fuel pin may therefore be found as the ratio between the pin powers determined in two such calculations where the pin enrichments have the appropriate values.

The ratio between the power density in fuel pellets made from natural UO_2 and 1.5% enriched UO_2 has been calculated. The result is $P_{nat}/P_{1.5\%} = 0.55$. This is in good agreement with the result obtained from a γ-scan on a fuel pin made from 1.5% enriched pellets and end pellets made from natural uranium. The burn-up of the pin was small: ~ 0.7 GJ/kg U (8 MWd/tU).

Since $\Sigma_a\phi$ will scarcely be affected by changing the radius of the fuel pin within reasonable limits, the results may also be used for other pin radii.

By the use of the described method for calculating the power density the ratios $P_{nat}/P_{1.5\%}$ and $P_{nat}/P_{2.28\%}$ are calculated. The results are given in Table A1.

Table A1

Time		Burn-up of nat. U		Burn-up of 1.5% enrich. U		Burn-up of 2.28% enrich. U		% Fissions in nat.U	% Fissions in 1.5% enr.U	% Fissions in 2.28% enr.U	$\frac{p_{nat}}{p_{1.5\%}}$	$\frac{p_{nat}}{p_{2.28\%}}$
Ms	days	GJ/kg U	MWd/tU	GJ/kg U	MWd/tU	GJ/kg U	MWd/tU					
0	0	0	0	0	0	0	0	3.69	6.68	9.20	0.55	0.40
2.6	30	45	517	70	813	92	1062	4.87	7.69	10.06	0.63	0.48
5.2	60	101	1170	151	1746	193	2230	6.13	8.79	11.03	0.70	0.56
10.4	120	258	2991	352	4078	431	4993	8.49	10.93	12.99	0.78	0.65
15.6	180	462	5348	596	6901	710	8213	10.97	13.20	15.11	0.83	0.73
20.7	240	718	8305	888	10278	1033	11957	13.74	15.77	17.53	0.87	0.78
25.9	300	1034	11972	1237	14321	1411	16329	17.08	18.86	20.45	0.91	0.84
31.1	360	1432	16578	1660	19216	1857	21498	21.38	22.82	24.17	0.94	0.88
36.3	420	1942	22472	2184	25278	2397	27741	27.44	28.28	29.19	0.97	0.94

APPENDIX B

ENERGY RELEASED PER FISSION

F. Højerup, T. Petersen

The energy released in a fission process comprises the kinetic energies of the fission fragments, fissions neutrons, β-particles, and neutrinos as well as the γ-radiation emitted during the fission or from the decay of fission products. Furthermore, energy is released by capture of neutrons (as γ-radiation).

For this work the following values which have been taken from ref. (B1) are used:

Table B1

	Energy per fission			
	U^{235} fission		Pu^{239} fission	
	PJ	MeV	PJ	MeV
Fission fragments	26.63±0.21	166.2±1.3	27.68±0.30	172.8±1.9
β-particles	1.12 ±0.05	7.0 ±0.3	0.98 ±0.10	6.1 ±0.6
neutrons	0.77 ±0.02	4.8 ±0.1	0.95 ±0.02	5.9 ±0.1
prompt γ	1.28 ±0.13	8.0 ±0.8	1.23 ±0.22	7.7 ±1.4
decay γ	1.15 ±0.18	7.2 ±1.1	0.98 ±0.21	6.1 ±1.3
(n,γ)*	1.28 ±0.16	8 ±1	1.28 ±0.16	8 ±1

* these numbers are not from ref. B1; they are evaluated in the following text.

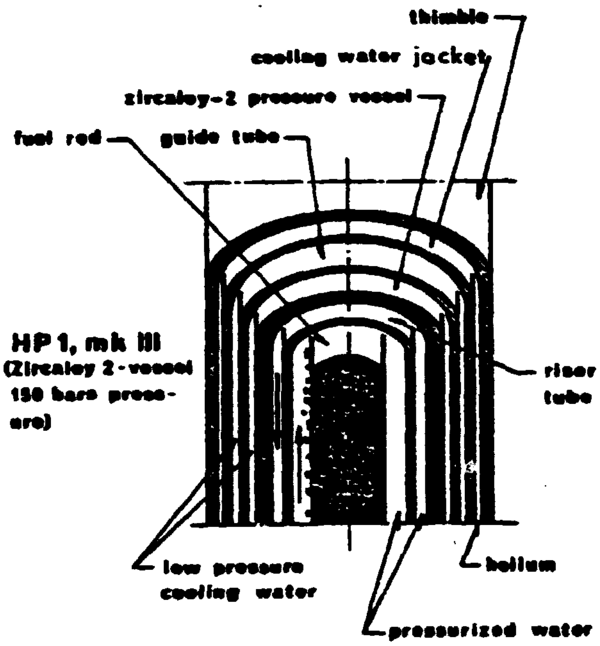


Fig. B1.

In these experiments the fuel rods are placed in a calorimeter.

In Figure B1 a cross section of the calorimeter is shown. All of the energy absorbed inside the He-gap will contribute to the energy measured by the calorimeter. In order to calculate the burn-up of the fuel pin from the calorimeter measurements the energy absorbed inside the He-gap per fission must be known.

The energy of the fission fragments and β-particles will be absorbed directly in the fuel. Because of the large amount of water in the calorimeter it is assumed that the kinetic energy of the fission neutrons will be absorbed inside the He-gap. Thus only the contributions from γ-radiation must be estimated.

Reactor physics calculations show that the neutrons captured in the materials in the calorimeter are shared approximately equally between water-zirconium and UO_2. The energies liberated per capture in some important nuclides are:

Nuclide	Q-value	
	pJ/capture	MeV/capture
H	0.35	2.2
Zr	1.30	8.1
U235	1.04	6.5
U238	0.91	5.7
Pu239	1.04	6.5

The rate of capture in the different nuclides will vary during burn-up, but, as can be seen from the table, the average energy liberated per capture can not deviate much from ~ 0.96 pJ (6 MeV), considering the given $Zr-H_2O$ mixture.

To get the energy liberated by neutron capture per fission we must multiply the energy liberated per capture with the effective number of captures per fission. In the beginning, when all fissions are U235-fissions, this number must be $(\nu_{235}-1) \times f_{leak} \sim 1.2$, with $\nu_{235} = 2.47$ and the leakage probability, $f_{leak} \sim 0.8$.

If all fissions were Pu239 fissions the corresponding number would be ~ 1.5, with $\nu_{239} = 2.89$.

These are the limits. 1.2 is the starting value and 1.5 will never be reached.

Thus the energy liberated by capture per fission will be in the range 1.15-1.44 pJ (7.2-9 MeV).

To calculate the γ-absorption of the calorimeter the latter is treated as a homogeneous cylinder having density as UO_2 and the same mass per unit length as the calorimeter. This equivalent cylinder turns out to have a radius of 13.0 mm.

The problem is now reduced to that of calculating the γ-absorption in a 13.0 mm ϕ cylinder in two cases:

1) When the γ-source is uniformly distributed across the cylinder (corresponding to capture γ's)
2) When the γ-source is confined to the inner 6.3 mm radius cylinder (corresponding to prompt γ's and fission product decay γ's).

Earlier calculations by means of a Monte Carlo method have shown that 55% of the γ-energy generated uniformly within a UO_2 cylinder with radius 6.3 mm escapes from the cylinder. This result is scaled to a UO_2 cylinder with radius 13.0 mm by the following method.

It is assumed that γ-quanta are absorbed at their first collision. From collision probability theory it is known that given an infinite cylinder in which particles are emitted uniformly the fraction of particles that escapes the cylinder without colliding with the atoms of the cylinder is given approximately by

$$\frac{4}{2+\ell} - \frac{3}{3+\ell} \qquad \text{(B1)}$$

where ℓ is the mean cord length and ℓ is a dimensionsless number. For an infinite cylinder ℓ is equal to the diameter given in units of mean free paths. By means of (B1) the mean cord length for a UO_2 cylinder with radius 6.3 mm may be calculated. The result is $\ell_o = 1.14$. The mean free path $\bar{r}$ for γ in UO_2 is then $\bar{r} = 1.26/1.14 = 1.11$ cm.

The mean cord length for the 1.30 cm cylinder is

$$\ell_1 = 2 \times 1.30/\bar{r} = 2.35 \qquad \text{(B2)}$$

By means of (B1) it is seen that ~ 65% of the energy from the (n,γ) reactions is absorbed in the calorimeter.

As stated earlier Monte Carlo calculations show that ~ 55% of the energy from prompt γ's and γ's from decay of fission products escape from the zone of radius 6.3 mm. Some of this energy is absorbed in the annulus with radii 6.3 mm and 13.0 mm. This absorption is calculated as if the annulus were an infinite slab of the same thickness as the annulus. The fraction of energy absorbed in the slab is

$$1 - e^{-2t/\bar{r}} = 0.70 \qquad \text{(B3)}$$

where t is the thickness of the slab.

The total fraction of energy from prompt γ's and γ's from the fission products absorbed in the calorimeter is then ~ 84%. (0.45 + 0.55 · 0.70) ~ 84%.

The energy released per fission to the calorimeter is given in Table B2.

Table B2. Energy per fission released to calorimeter.

	U^{235} fissions		Pu^{239} fissions	
	pJ	MeV	pJ	MeV
Fission fragments	26.63±0.21	166.2±1.3	27.68±0.30	172.8±1.9
β-particles	1.12 ±0.05	7.0 ±0.3	0.98 ±0.10	6.1 ±0.6
neutrons	0.77 ±0.02	4.8 ±0.1	0.95 ±0.02	5.9 ±0.1
prompt γ	1.07 ±0.13	6.7 ±0.8	1.04 ±0.22	6.5 ±1.4
decay γ	0.96 ±0.18	6.0 ±1.1	0.82 ±0.21	5.1 ±1.3
(n,γ)	0.88 ±0.16	5.5 ±1	0.88 ±0.16	5.5 ±1
Total	31.4 ±0.3	196 ±2	32.4 ±0.3	202 ±2

REFERENCE

(B1) JAMES, M.F., "Energy Released in Fission", AEEW-M863, 1969.

APPENDIX C

EXAMPLE OF CALCULATION OF POWER AND BURN-UP DISTRIBUTION

The procedures for calculating assembly values (section 4.1.2), local values (section 4.1.3) and pin average values (section 4.1.4) have been implemented on a PDP 11/10 minicomputer (ref. C1). The input required for a given calculation is given on the following page, while the corresponding output is seen on the continuing page.

REFERENCE

1) BAGGER, C. (1979). Program HP, Version PDP/1, til effekt- og udbrændingsfordelingsberegning på HP-1 stave". Internal Report B472.

CALCULATION OF HEAT RATING AND BURN UP

INPUT TO PROGRAM HP/PDP/1

HP - test no.: 069 Pins (max. 3) 3 Irr. per's 3

PIN INFORMATION	FUEL LENGTH			WEIGHTS		
Pin no. Bottom pin first	End pellets Bottom	End pellets Top	Total stack	End pellets	Central pellets	Pin total
	cm	cm	cm	g	g	g
AG 17-4	1.05	1.05	12.49	14.0	75.0	124.0
AG 17-7	1.04	1.04	12.40	14.0	75.0	124.0
AG 17-6	1.02	1.02	12.46	14.0	75.5	124.0

Rig type MK II/III: III BWR/PWR: PWR Weight couplings: 13.4 g

Enr. init., end pellets: 2.28 % Enr. init., central pellets: 3.16 %

RP - calculation parameters according to $RP = A \cdot BU^{B} + C$, $D = BU_{max}$, GJ/kgU				
A	B	C	D	
1.7 E-4	1.0	0.750	121.	Linear eq. at low BU
0.404	0.0818	0.0	4342	Power eq. at higher BU

<u>Gamma scan information.</u> Isotope: 95 Zr/Nb

Curve heights in pos. 1-6 (bottom first): 59.0 | 59.5 | 59.5 | 59.5 | 58.7 | 55.8

Average height of central pellet scans: 58.7

Central pellet scan curve areas (bottom pin first): 6399 | 6486 | 6240

Irradiation parameters

DR 3 per. No.	DR 3 pos.	Core power MW	Irr. time h	Rig power kW
185	A1	10.0	650	22.5
186	A1	10.0	526	21.3
187	A1	10.0	559	20.9

— continued

DR 3 per. No.	DR 3 pos.	Core power MW	Irr. time h	Rig power kW

May be continued up to 74 entries

HP-TEST 69 RIG TYPE MARK 3 PIN TYPE PWR

TABLE C1
IRRADIATION PARAMETERS AND ASSEMBLY RESULTS

DR3.PER NO	POS	CORE POWER MW	IRRT HRS	Q(RIG) KW	HEATRATING KW/M	BURN-UP GJ/KG U	BURN-UP MWD/TUO2
185	A1	10.0	650	22.5	44.4	167.6	1710
186	A1	10.0	526	21.3	41.3	293.6	2996
187	A1	10.0	559	20.9	40.2	424.1	4327

TABLE C2
PIN AVERAGE VALUES

AG17-6 TOP-PIN			AG17-7 CENTER-PIN			AG17-4 BOTTOM-PIN		
POWER	BURN UP		POWER	BURN UP		POWER	BURN UP	
KW/M	GJ/KGU	MWD/TUO2	KW/M	GJ/KGU	MWD/TUO2	KW/M	GJ/KGU	MWD/TUO2
43.5	163.9	1672.	45.1	170.9	1743.	44.7	169.0	1724.
40.5	287.1	2929.	42.0	299.3	3053.	41.5	296.0	3020.
39.5	414.7	4230.	40.9	432.3	4411.	40.5	427.6	4362.

TABLE C3
LOCAL VALUES
AG17-6

POSITION E6		POSITION C6		POSITION C5		POSITION E5	
POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U
33.7	126.0	43.8	164.9	46.1	173.4	35.4	132.6
32.1	222.9	40.6	288.3	42.7	303.3	33.8	235.0
32.2	326.5	39.4	415.5	41.4	437.1	34.0	344.5

TABLE C4
LOCAL VALUES
AG17-7

POSITION E4		POSITION C4		POSITION C3		POSITION E3	
POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U
35.9	134.5	46.7	175.8	46.7	175.8	35.9	134.5
34.3	238.4	43.3	307.4	43.3	307.4	34.3	238.4
34.5	349.4	42.0	443.1	42.0	443.1	34.5	349.4

TABLE C5
LOCAL VALUES
AG17-4

POSITION E2		POSITION C2		POSITION C1		POSITION E1	
POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U	POWER KW/M	BURN-UP GJ/KG U
35.9	134.5	46.7	175.8	46.3	174.3	35.6	133.3
34.3	238.4	43.3	307.4	42.9	304.8	34.0	236.3
34.5	349.4	42.0	443.1	41.6	439.4	34.2	346.3

<u>Note</u>: Position numbers in Tables C3, C4 and C5 are defined in Figure 10 in the main text.
The letter "E" or "C" in front of the position number refers to end or central pellets, respectively.

APPENDIX D

NOMENCLATURE AND UNITS

Main parameter

		SI-unit
BU[-]	Accumulated Burn-Up	GJ/kg U
BUI[-]	Burn-Up Increment	GJ/kg U
CPF	Core Power Factor Actual DR3 core power in the irradiation period in question, MW, divided by the nominal core power = 10 MW	-
G[-]	Material specific gamma power	W/g
H[-]	Height of a curve	mm
IT	Irradiation Time (often one DR3 operating period)	s
L[-]	Length of material	mm
P[-]	Power, linear heat rating	kW/m
Q[-]	Heat, power	kW
RP[-]	Ratio of Power	-
ΔT	Temperature increase	K
$\dot{V}$	Volume of flow per time	m^3/s
W[-]	Weight of material	kg

Text in bracket

This text specifies in general the main parameters with regard to origin, time and location. If unspecified, the main parameter refers to the whole assembly.

A list of non-self-explanatory text is given below.

AVG	Average
ASS'Y	Assembly
C	Central pellets
E	End pellets
E1	First of two end pellets
E2	Second " " " "
FISS	Due to fissions only
FUEL	for Q[FUEL]: heat power due to fissions and γ-absorption
GAMMA,γ	Due to γ-absorption
POS X	Position of central pellets
POS Y	Position at end pellets

TABLE OF CONVERSION FACTORS

1 MPa = 1 MN/m^2	=	9.869 kp/cm^2 = 9.869 at.
1 kW/m^2	=	0.1 W/cm^2
1 kW/m	=	10. W/cm
1 GJ/kg U	=	11.57 MWd/tU 10.20 MWd/tUO_2
1 pJ	=	6.24 MeV
1 s	=	$3.169\cdot10^{-8}$ year $1.157\cdot10^{-5}$ day $2.778\cdot10^{-4}$ hour $1.667\cdot10^{-2}$ minute

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Title and author(s)	Date
CALCULATION OF HEAT RATING AND BURN-UP FOR TEST FUEL PINS IRRADIATED IN DR3. C. Bagger, H. Carlsen and K. Hansen	January 1980
	Department or group
	Group's own registration number(s)
50 pages + tables + illustrations	

Abstract

A summary of the DR3 reactor and HP1 rig design is given followed by a detailed description of the calculation procedure for obtaining linear heat rating and burn-up values of fuel pins irradiated in HP1 rigs. The calculations are carried out rather detailed, especially regarding features like end pellet contribution to power as a function of burn-up, gamma heat contributions, and evaluation of local values of heat rating and burn-up. Included in the report is also a description of the fast flux- and cladding temperature calculation techniques currently used.

A good agreement between measured and calculated local burn-up values is found. This gives confidence to the detailed treatment of the data.

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