KERNFORSCHUNGSZENTRUM

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Institut für Angewandte Reaktorphysik

Program, Pin Design and Specifications for Fuel Irradiation Experiments in the Enrico Fermi Fast Breeder Reactor

G. Karsten, K. Kummerer
with Contributions of
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+) Work performed within the association in the field of fast reactors between the European Atomic Energy Community and Gesellschaft für Kernforschung mbH., Karlsruhe.

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PREFACE

This report contains the revised version of the irradiation proposal delivered to Atomic Power Development Associates, Inc., Detroit (APDA) in March 1966. This proposal was principally accepted by APDA in April 1966.

In addition some introductory remarks and comments are included in order to clarify the background and boundary conditions.

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1. Introduction

For the fuel element development of fast breeder reactors, it is essential to perform test irradiations in fast neutron environment. Therefore a fuel irradiation agreement was established between the Euratom community and Atomic Power Development Associates Inc., Detroit, USA (APDA) with the aim to perform fuel irradiations in the Enrico Fermi Fast Breeder Reactor (EFFBR) for the different fast breeder projects associated to Euratom. In this framework a high flux core position of the EFFBR was tentatively allocated to the Karlsruhe fast breeder project. The envisaged position is of the category 1900 (which means the daily irradiation charge in US-\$). Fig.1 outlines the potentially available positions in the EFFBR core cross section. There was designed by APDA a standard fuel test subassembly with four thermally insulated thimbles, Fig.2. The nominal power condition of the EFFBR is scheduled to be 110 Megawatt (thermal) leading to a maximum neutron flux in the core center of 2.6×10^{15} n/cm^{2}sec. The radial and axial distribution of the total flux is shown in Fig.3 and 4.

In the following sections of this report all details and calculations are composed as requested by APDA in the course of the preparational work.

2. Program for the Irradiations

2.1 Basic Characteristics

We assume to have one position available in the Fermi-core. The irradiations are planned for a position of the 1900 \$/day-category near the core center. Our pin design is further based on the assumption, that we shall use the standard fuel test subassembly, as it was designed by APDA, containing 24 single-clad pins in 4 thimbles, 6 pins per thimble, each thimble being equipped with a coolant orifice adapted to the power output.

2.2 Irradiation Schedule

We propose to irradiate a total of 36 pins in the Charges I, II and III. Each charge is composed by two batches, containing 6 pins each. Charge I shall be irradiated the full time of the contract, Charge II will be irradiated up to about 10 000 MWd/t [+)], being replaced by Charge III.

2.3 Pin Variants and Identification

The batches in Charge I are identified by the letters A and B, in Charge II by C and D, in Charge III by E and F. The couples of batches are identical in the different charges, but are different in their own composition. They differ by cladding material, fuel type and fuel density. By the following table the differences are shown and the signature of each of the 36 pins to be irradiated is given.

Table I Composition of the Irradiation Charges

<table>
<tr><th colspan="4">Pin Identification</th><th rowspan="2">Cladding Material</th><th rowspan="2">Fuel Type</th><th rowspan="2">Smeared density (th.d.)</th><th rowspan="2">fuel density (th.d.)</th></tr>
<tr><th>Charge I</th><th>II</th><th>III</th><th>Reserve Pins</th></tr>
<tr><td>A 1
A 2
A 3</td><td>C 1
C 2
C 3</td><td>E 1
E 2
E 3</td><td>X 1</td><td rowspan="2">X8CrNiMoVNb 1613</td><td>pellets</td><td>83
83
88</td><td>88
88
93</td></tr>
<tr><td>A 4
A 5
A 6</td><td>C 4
C 5
C 6</td><td>E 4
E 5
E 6</td><td>X 2</td><td>vibrational compacted</td><td>85
85
85</td><td>85
85
85</td></tr>
<tr><td>B 7
B 8
B 9</td><td>D 7
D 8
D 9</td><td>F 7
F 8
F 9</td><td>X 3</td><td rowspan="2">Inconel 625</td><td>pellets</td><td>83
83
88</td><td>88
88
93</td></tr>
<tr><td>B10
B11
B12</td><td>D10
D11
D12</td><td>F10
F11
F12</td><td>X 4</td><td>vibrational compacted</td><td>85
85
85</td><td>85
85
85</td></tr>
</table>

Each batch shall be loaded into a separate thimble within the subassembly. Hence the grouping of the first subassembly loading is:

Thimble No.	1	2	3	4
Pin No.	A 1 - A 6	B 7 - B 12	C 1 - C 6	D 7 - D 12

+) burnup always in megawatt-days per metric ton of heavy elements contained in the fuel

After replacing Charge II by Charge III the grouping is proposed to be as follows:

Thimble No.	1	2	3	4
Pin No.	A 1 - A 6	B 7 - B 12	E 1 - E 6	F 7 - F 12

2.4 Reserve Pins

For all the pins actually to be irradiated in the subassembly four reserve pins with the designation X 1 to X 4 are fabricated. In case of pin failure at the first subassembly loading the following replacements shall be applied:

At	failure	of	A1, A2, C1 or C2	replace	by	X 1
"	"	"	A3 or C3	"	"	E 3
"	"	"	A4, A5, A6, C4, C5 or C6	"	"	X 2
"	"	"	B7, B8, D7 or D8	"	"	F 7
"	"	"	B9 or D9	"	"	X 3
"	"	"	B10, B11, B12, D10, D11 or D12	"	"	X 4

In case of replacements at the first subassembly loading also a modified arrangement at change from Charge II to Charge III will become necessary. This will be specified later according to the still available reserve pins and to the other circumstances.

3. Design and Performance Data

3.1 Pin Design

a) The conceptional design of the pins is:

fuel composition: UO_2-PuO_2 mixed oxide with 15 % PuO_2, the U fully enriched

fuel density: 88 or 93 % th.d. for pellets and 85 % th.d. for vibrational compacted powder

fuel fabrication type: pellet and powder packed oxide, respectively fuel diameter: 5.4 mm with pellets and 5.55 mm with vibrational compacted powder

diametral clearance at pellet fuel: 0.15 mm

blanket: natural UO_2, density same as corresponding fuel

insulator pellet: natural UO_2, 88 % th.d.

cladding thickness: 0.4 mm

outer diameter: 6.35 mm

over-all length (with end caps): 1286.3 mm

inner length (without end caps): 1241.4 mm

b) The pin length is divided into the following sections (mm):

	Pin Type I X8CrNiMoVNb 1613 cladding	Pin Type II Inconel 625 cladding
upper end cap	16,1	16,1
upper gas plenum	188,4	63,7
blanket	150,0	200,0
fuel zone	700,0	774,7
insulator pellet	8,0	8,0
lower gas plenum	195,0	195,0
lower end cap	28,8	28,8
total length	1286,3	1286,3

c) The pin design is represented by a general drawing and 2 special technical drawings as follows:

Designation	Drawing No.
Irradiation Test Pins, Type I + II	TA 2K-3 -07-02-1755
Brennelement-Prüfstab, Type I	TA 2K-16-07- 2-1701
Brennelement-Prüfstab, Type II	TA 2K-16-07- 2-1682

They are included here in Appendix C.

The survey drawing TA-2K-3-07-02-1755 shall be explained as follows: The outer dimensions have been taken from APDA design and will fit into the APDA Support System. We are presenting the pin types I and II, clad by X8CrNiMoVNb1613 or Inconel 625, respectively.

d) Special description of design details

- The upper end cap has a hole for evacuating after completed welding works of the as-fabricated pin and filling with helium. It will be closed then by a small plug and welded.

- The upper gas space is containing an inserted tube (Type I) or a plug only (Type II), fixed by a tight fit at its upper end and serving as a limitation for the spring (Part 3 is a SS-filter pellet for gas passage. It is described below.). The spring only has to avoid fuel shifting during transport. It can be compressed however by a movable slide when fuel expansion should occur. The upper gas spaces are of different volumes, because of different cladding properties. The lower mechanical cladding properties in Type I are furthermore aided by smaller blanket and fuel quantities compared to Type II.

- Fission gas can move into the lower gas space, equal in both types, by a CrNi SS-filter pellet only. This filter pellet is a special product for such purposes. It closes the inserted supporting tube, which fits into the cladding tube very tightly by a tolerance of +0 and -8 microns in order to avoid fuel to fall into the lower gas space during fuel loading, especially during vibrating. The sinter filter is protected by a natural UO_2-insulator pellet against fuel heat.

- After irradiation the lower end cap must be cut off at the neck for the clips in order to release the fission gas. For this purpose the end cap has a axial hole inside.

3.2 Power Data and Target Burnup

These data are given in the following table for a 110 MW reactor power condition.

Table II   Operational Data

Core Position ($/day)	Max.Total Neutron Flux ($n/cm^2 sec$)	Power Density (W/g)	Average Burnup per 225 days at 100% Load Factor (MWd/t)	Fuel Smeared Density (% th.d.)	Max. Linear Rod Power (W/cm)
1900	$2.53 \cdot 10^{15}$	268 +)	50 000	83 85 88	600 615 635

Explanation of table columns:

- Core position: About 10 cm from core centerline
- Smeared density: Fuel density, including diametral clearance volume
- Target burnup: Burnup, averaged over fuel length, after 225 full power operating days
- Rod power: Maximum linear rod power at core midplane

3.3 Sodium Coolant Temperatures

The sodium coolant temperatures are given for nominal and 130 % of power. They have been calculated with the following formula:

$$T(z) = T_{inlet} + \frac{\Delta T}{2} \left[1 + \frac{\sin(\frac{\pi}{2} \cdot \frac{z}{L/2})}{\sin(\frac{\pi}{2} \cdot \frac{l}{L})} \right] \qquad (1)$$

This expression is derived from the basic equations

$$\chi(z) = \chi_{max} \cdot \cos(\pi \frac{z}{L}) \qquad (2)$$

and

$$\chi(z) = Q \cdot c_p \cdot \frac{d}{dz} T(z) \qquad (3)$$

by integration and introduction of the boundary conditions $z = -\ell/2$, $T(z) = T_{inlet}$; $z = +\ell/2$, $T(z) = T_{outlet}$.

+) calculated with $\bar{\sigma}_F = 1.56$ barn

The designations are:

$T(z)$ = local axial coolant temperature at distance z

$\Delta T(z)$ = $T(z) - T_{inlet}$

T_{inlet} = $232^{o}C$

ΔT = $T_{outlet} - T_{inlet}$ (^{o}C)

z = distance from fuel midplane (cm)

L = extrapolated length of power distribution $\hat{=}$ 110,5 cm

l = fuel length $\hat{=}$ 70 cm for pin type I and 77,47 cm for pin type II

χ = linear rod power (W/cm)

Q = coolant quantity (g/sec)

c_p = heat capacity (W sec/g ^{o}C)

Coolant temperatures are calculated for the SS clad and for the Inconel 625 clad pins at nominal power and at 130 % of power with formula (1). The results are given in the next tables.

Table III Sodium Temperatures for SS-clad Pins

z (cm)	Fuel height h (cm)	$\Delta T(z)$ at nominal power (^{o}C)	$\Delta T(z)$ at 130% power (^{o}C)	$T(z)$ at nominal power (^{o}C)	$T(z)$ at 130% power(^{o}C)
35	70	336	435	568	667
30	65	320	415	552	647
20	55	276	358	508	590
10	45	226	294	458	526
0	35	168	218	400	450
-10	25	111	144	343	376
-20	15	60	78	292	310
-30	5	17	22	249	254
-35	0	0	0	232	232

Table IV Sodium Temperatures for Inconel 625-clad Pins

z (cm)	fuel height h (cm)	ΔT(z) at nominal power (°C)	ΔT(z) at 130% power (°C)	T(z) at nominal power (°C)	T(z) at 130% power (°C)
38.7	77,5	370	480	602	712
30	68,7	342	445	574	677
20	58,7	296	385	528	617
10	48,7	244	317	476	549
0	38,7	185	240	417	472
-10	28,7	127	165	359	397
-20	18,7	73	95	305	327
-30	8,7	28	36	260	268
-38.7	0	0	0	232	232

The coolant temperatures are plotted in Fig.5. The maximum temperatures are 712°C (1330°F) for the case of Inconel clad pins and 667°C (1240°F) for SS-clad pins at 130 % of nominal power. At nominal power the corresponding data are 602°C (1110°F) and 568°C (1060°F).

In connection with sodium temperatures, it has to be mentioned that the cooling rates for the averaged rod power of 500 W/cm for the 6 pin containing thimble are 1810 kg/h for Inconel-clad pins and 1800 kg/h for SS-clad pins. These quantities of sodium which are needed for cooling the Inconel- and SS-clad pins were calculated for nominal power condition with the formula

$$Q = \frac{\overline{\chi} \cdot l}{\Delta T \cdot c_p} \qquad (4)$$

with c_p = 1.25 W sec/g °C and an average linear rod power $\overline{\chi}$ = 500 W/cm.

The data ΔT and l used were for	ΔT (°C)	l (cm)
Inconel-clad pins	370	77,5
SS-clad pins	336	70,0

Though the sodium temperature rise is smaller in the latter case the coolant rate is somewhat lower because the fuel zone in SS-clad pins is shorter than in the Inconel-clad pins.

3.4 Average Cladding Temperatures

The cladding temperatures T_{av} are given for the middle of the wall thickness over the fuel length both for 130 % and nominal power.
The temperatures have been calculated in the following way for SS- and Inconel 625-clad pins. 15°C as temperature difference between sodium and cladding surface have been added to the sodium temperatures leading to the surface temperature T_e of the cladding. Half of the temperature rise ΔT_i in the cladding was added to T_e for getting T_{av}. For ΔT_i we have

$$\Delta T_i = \frac{\chi(z)}{2\pi k_c(T)} \cdot \ln\left(\frac{r_a}{r_i}\right) \tag{5}$$

with k_c = heat conductivity of the cladding (W/cm°C)

$2r_a$ = 6.35 mm outer diameter of cladding

$2r_i$ = 5.55 mm inner diameter of cladding

The results are compiled in the tables V to VIII and demonstrated in Fig.6. The maximum temperatures at 130 % of power are 746°C (1380°F) for Inconel 625 and 706°C (1300°F) for the SS. At nominal power the corresponding data are 632°C (1160°F) and 602°C (1110°F). The differences are induced by the different sodium outlet temperatures.

Table V Average Cladding Temperatures for SS-clad Pins at Nominal Power

z (cm)	Fuel height h (cm)	T_e (°C)	k_c (W/cm°C)	$\chi^{+)}$ (W/cm)	ΔT_i (°C)	T_{av} (°C)
35	70	583	0,22	390	38	602
30	65	567	0,21	450	46	590
20	55	523	0,20	550	59	553
10	45	473	0,195	610	67	507
0	35	415	0,19	610	69	450
-10	25	358	0,185	565	65	390
-20	15	307	0,18	480	57	336
-30	5	264	0,16	350	47	288
-35	0	247	0,15	280	40	267

+) Rod power for 85 % th.d. (smeared density) calculated with total flux of $\phi = 2.53 \cdot 10^{15}$ n/cm²sec and $\bar{\sigma}_{fission}$ $(UO_2/PuO_2) = 1,56 \cdot 10^{-24}$ cm².

Table VI Average Cladding Temperatures for Inconel 625-clad Pins at Nominal Power

z (cm)	Fuel height h (cm)	T_e (°C)	k_c (W/cm°C)	χ (W/cm)	ΔT_i (°C)	T_{av} (°C)
38.7	77,5	617	0,19	280	31	632
30	68,7	589	0,18	405	48	613
20	58,7	543	0,175	515	63	575
10	48,7	491	0,17	590	74	528
0	38,7	432	0,16	615	82	473
-10	28,7	374	0,155	590	82	415
-20	18,7	320	0,15	515	73	356
-30	8,7	275	0,14	405	62	306
-38.7	0	247	0,13	280	46	270

Table VII Average Cladding Temperatures for SS-clad Pins at 130 % of Power

z (cm)	Fuel height h (cm)	T_e (°C)	k_c (W/cm°C)	χ (W/cm)	ΔT_i (°C)	T_{av} (°C)
35	70	682	0,23	510	48	706
30	65	662	0,225	585	56	690
20	55	605	0,22	715	70	640
10	45	541	0,215	795	79	580
0	35	465	0,19	795	90	510
-10	25	391	0,185	735	85	434
-20	15	325	0,18	625	74	362
-30	5	269	0,16	455	61	300
-35	0	247	0,15	365	52	273

Tabele VIII   Average Cladding Temperatures for Inconel 625-clad Pins at 130 % of Power

z (cm)	Fuel height h (cm)	T_e (°C)	k_c (W/cm°C)	χ (W/cm)	ΔT_i (°C)	T_{av} (°C)
38,7	77,5	727	0,21	365	37	746
30	68,7	692	0,20	525	56	720
20	58,7	632	0,19	670	75	670
10	48,7	564	0,18	765	91	610
0	38,7	487	0,17	800	100	537
-10	28,7	412	0,16	765	102	463
-20	18,7	342	0,15	670	95	390
-30	8,7	283	0,14	525	80	323
-38,7	0	247	0,13	365	60	277

3.5 Integral of Heat Conductivity

This integral is given for a density of 90 % th.d., Fig.7. It is based on a literature review (see below) assuming that the thermal conductivity is the same both with UO_2 and UO_2-PuO_2 of the chosen composition. The melting point is chosen to be about 2730°C for UO_2-15w/o PuO_2.

Literature

for Heat Conductivity:

J.A.L. Robertson, J.Nuc.Mat. 7 (1962)

M.F. Lyons, GEAP 4624

D.H. Choplin et al., Trans. ANS 8 (1965)

W.D. Kingery, J.Am.Ceram.Soc. 42 (1959)

for Melting point:

T.D. Chikalla, HW-69832 (1961)

W.L. Lyon et al., Trans ANS 8 (1965)

3.6 Fuel Surface and Central Temperatures

The fuel surface temperatures were obtained by adding the temperature differences in the gap between the inner surface of the cladding and the fuel surface to the cladding temperatures at the inner surface. The cladding temperatures are calculated from values of 3.4. The temperature differences in the gap were calculated by the formula

$$\Delta T_g = \frac{\chi_{(z)}}{2\pi r_i \alpha} \qquad (6)$$

$\chi(z)$ = linear rod power (W/cm)
r_i = inner cladding radius (cm)
α = heat transfer coefficient

The data in the following tables are given for nominal power and for 130% power for the Inconel-clad pins, which is the case with higher fuel temperatures. The value for heat transfer between cladding and fuel surface was taken as an average of data published by different authors. It seems right to take 1 W/cm^2 ^{o}C (1760 Btu/hr ft^2 ^{o}C).

Literature for gap heat transfer:

M. Bogaievski et al., Proc. Vienna Conf. 1963, 1
J.A.L. Robertson, J.Nuc.Mat. 7 (1962)
Westinghouse WCAP-4059 (1961)
I. Cohen et al., J.Nuc.Mat. 3 (1961)
A.M. Ross, R.L. Stoute, AECL-1552 (1962)
G. Höppner, Atomkernenergie (1965) 5/6

Furthermore in the sum of heat conductivity integrals

$$\int_o^{T_c} kdt = \int_o^{T_s} kdt + \int_{T_s}^{T_c} kdt \qquad (7)$$

($T_s \hat{=}$ fuel surface temperature, $T_c \hat{=}$ fuel central temperature)

the first term is evaluated by introducing the fuel surface temperature T_s. Then at a given rod power the second term is specified leading to the

fuel central temperature by Fig.7.
In the tables IX and X all the data for fuel surface and central temperatures for Inconel-clad pins only - the worst case - are compiled. The internal surface temperature of the cladding $T_i = T_e + \Delta T_i$ is taken out of Tables VI and VIII. In the further treatment the temperature figures are approximated to the nearest 5^oC. Figures 8 and 9 are the corresponding graphs.

Table IX   Fuel Temperatures at Nominal Power (Inconel-clad)

z (cm)	fuel height h (cm)	T_i (^oC)	ΔT_g (^oC)	T_s (^oC)	$\int_0^{T_c} kdt$ (W/cm)	T_c (^oC)
38,7	77,5	648	160	810	60	1760
30	68,7	637	230	870	72	2160
20	58,7	606	295	900	82	2440
10	48,7	565	340	905	88	2580
0	38,7	512	350	860	89	2600
-10	28,7	456	340	795	85	2510
-20	18,7	393	295	690	76	2280
-30	8,7	337	230	570	63	1870
-38,7	0	293	160	455	49	1250

Table X   Fuel Temperatures at 130 % of Power (Inconel-clad)

z (cm)	fuel height h (cm)	T_i (^oC)	ΔT_g (^oC)	T_s (^oC)	$\int_0^{T_c} kdt$ (W/cm)	T_c (^oC)
38,7	77,5	764	220	985	72	2170
30	68,7	748	300	1050	87	2550
20	58,7	707	385	1090	99	2840
10	48,7	655	440	1095	107	3000
0	38,7	587	460	1045	109	3040
-10	28,7	514	440	955	104	2940
-20	18,7	437	385	820	93	2700
-30	8,7	363	300	665	76	2280
-38,7	0	307	220	525	58	1680

At nominal power the fuel will remain below melting point, 2600°C max. The temperature will rise up to 3040°C max. at 130 % of power. This means that the fuel will be molten in a channel of about 2.5 mm in diameter and 45 cm in length (about 10 % of fuel volume) if power excursion happens at the beginning of the irradiation. After some short time, however, there will probably be no melting, even at 130 % of power. Most of the porous volume will have migrated to the center having formed a void of just that size described above because the smeared density is more than 10 % below theoretical density.

Failure on fuel melting is not to be expected since fuel density is as low as to equal the fuel volume increase of 10 % at melting.

4. Design and Fabrication Conditions and Irradiation Criteria for Euratom Fuel Specimens

This chapter brings the full wording of the criteria for design, fabrication and irradiation, which was established by APDA in the final form by May 25, 1966.

The drawings

6XN-5385 E
6XN-5824 D
5XN-5413 B

which are mentioned in the text, are fully taken into account in the pin design. In this report the copies of these drawings are not included.

I. Final Criteria for the Design and Fabrication of Fuel Specimens

A. Test Vehicle

All Euratom irradiations of fissile materials must be compatible with, and conducted in, the standard Fuel Test Subassembly being developed by APDA for Detroit Edison. Specifically, the fuel specimens will be placed in insulated tubes of 0.944-inch inside diameter. There are four such tubes (thimbles) in each Fuel Test Subassembly, as shown in Drawing 6XN-5385E previously transmitted to Euratom.

Design and procurement of filler pieces which may be desired to reduce the coolant channel from a 0.944-inch circle to any other shape are the responsibility of Euratom. Acceptance is conditional upon APDA design review.

Design and procurement of specimen support devices are the responsibility of Euratom. Acceptance is conditional upon APDA design review. The only exception to this is the option of Euratom to accept the design of a support system being developed by APDA for Detroit Edison to hold six 0.250-inch diameter pins in each thimble. The pins are held at 60° intervals on a 0.610-inch circle (6.3 or 6.4 mm is considered to be equivalent to 0.250 inch for this purpose.).

All components of the experiment (i.e., everything inside the 0.944-inch thimble inside diameter) must be loaded from the top and held down from the top. The hold-down contact is the thermal shield assembly at the discharge end of the thimble as shown in Drawing 6XN-5824D previously transmitted to Euratom.

Responsibility for items whose design and procurement are specified above as Euratom's responsibility may be, at Euratom's option, delegated to APDA. All other hardware associated with the Fuel Test Subassembly will be procured by APDA. All fuel specimens must be furnished by Euratom. Acceptance is conditional upon APDA design review and upon adherence to the criteria given herein.

B. Specimen Description

1. General

The fuel specimens shall be round and cylindrical, no greater than 3/4 inch in diameter and no greater than 50-41/64 inches in length. The Fermi core falls in the span from 9-1/8 inches to 39-5/8 inches from the bottom of this maximum specimen length.

2. For Use With APDA-Designed Support System

The test specimens shall be round and cylindrical with a 0.250-inch nominal diameter (6.3 or 6.4 mm nominal diameter) and no greater than 50-41/64 inches in length. The Fermi core falls in the span from 9-1/8 inches to 39-5/8 inches from the bottom of this maximum specimen length. Each specimen must be provided with a lower end cap as detailed in Drawing 5XN-5413B previously transmitted to Euratom.

C. Maximum Sodium Temperature

The maximum local sodium temperature in the Fuel Test Subassembly shall not exceed 1200 F with hot channel factors to 3 σ confidence.

The hot channel factors include all uncertainties which affect specimen power and thimble flow rate. They also include coolant maldistribution within the thimble and the inherent nonstatistical uncertainty of the orientation of the four faces of the Fuel Test Subassembly within the lattice position. To clarify this latter point, the inability to orient the faces of the subassembly requires the assumption that each of the four thimbles is the one closest to the reactor centerline.

It is APDA's responsibility to evaluate the magnitude of the hot channel factor and to determine the nominal thimble outlet temperature equivalent to a 1200 F maximum sodium temperature with 3 σ confidence. It is APDA's responsibility to perform calculations and hydraulic tests to properly orifice each thimble for either this temperature or the nominal thimble discharge temperature desired by Euratom, whichever temperature is lower.

It shall be Euratom's responsibility to provide APDA with the following information required for the hot channel factor analysis:

1. Nominal cladding diameter and tolerance.
2. Nominal over-all length and fueled length with tolerances.
3. Anticipated diametral swelling during the irradiation period.
4. Reactor lattice zone requested (ϕ/day from Fig.2 of Euratom-APDA Fuel Irradiation Services Agreement).
5. Isotopic composition of specimens and possible variations from the nominal composition for the following isotopes: U-235, U-238, Pu-239, Pu-240, Pu-241. If the composition is not uniform over the fueled length of the pin, so indicate.
6. If the pin support structure is not the APDA design, furnish complete details. Indicate nominal location of fuel specimen and deviation from the nominal locations permitted by the design.

D. <u>Exit Sodium Temperature Matching</u>

The nominal sodium outlet temperature of a Fuel Test Subassembly, after full mixing of the bypass sodium with the sodium leaving the four thimbles, must be within $\pm$ 100 F of the nominal outlet sodium temperature of any adjacent core or blanket subassembly. It is APDA's responsibility to size nozzle orifices and make hydraulic tests required to meet this specification.

If, during the initial approach to full power of a newly loaded Fuel Test Subassembly, it becomes clear from the sodium outlet thermocouple reading that at full power either:

1. the mixed outlet sodium temperature will not fall within $\pm$ 100 F of the nominal sodium outlet temperature for an adjacent lattice position,
2. a thimble discharge temperature will exceed 1200 F,

the Subassembly must be removed from the reactor and the specimen bundles transferred to new thimbles (and possibly a new subassembly) with resized orifices. To avoid the possibility of discharging the Subassembly as a result of a faulty thermocouple reading, the Subassembly will be placed in an alternate core position which is under a funtioning thermocouple to confirm the observation that sodium temperature criteria will be violated if the Subassembly is operated at power.

E. Mismanagement

The following protection is required to guard against the possibility of improper positioning of the Fuel Test Subassembly in the reactor:

> Any Fuel Test Subassembly must be orificed in such a way that if the Subassembly is inadvertently placed in the central lattice position, the maximum local sodium temperature in that Subassembly shall not exceed 1475 F with hot channel factors to 3 σ confidence.
>
> It is APDA's responsibility to evaluate this factor and, should this consideration call for higher thimble flow rates than Item I.C., to orifice the Subassembly for the higher thimble flow rates.

F. Test Subassembly Reactivity Limitations

To qualify for a given lattice position, the reactivity of a Fuel Test Subassembly, evaluated at that position, must fall within limits as defined below:

1. Upper Limit

   The reactivity of an individual Fuel Test Subassembly in the core must not exceed that of the core or blanket subassembly it replaces by more than 10 per cent or 10 cents ($\Delta k/k = 0.00073$), whichever is greater. The reactivity of a Fuel Test Subassembly in the inner radial blanket zone (outer row of high pressure plenum) must not exceed that of a core subassembly at the same location.

2. Lower Limit
   a. The reactivity of an individual Fuel Test Subassembly shall be sufficiently high that when the core subassembly it replaces is transferred to the highest worth inner radial blanket position available, there is not net decrease in reactivity, or
   b. The collective reactivity of all in-reactor Fuel Test Subassemblies from Euratom shall be sufficiently high that when the core subassemblies they replace are transferred to the highest worth inner radial blanket positions available, there is no net decrease in reactivity.

Significant additions of moderating material such as would cause high local power peaking in adjacent core subassemblies will not be accepted.

Final judgment of acceptability, under this criterion, of a Fuel Test Subassembly loading for a given lattice position will be based on APDA calculations. Subsequent nuclear measurements which demonstrate that the Fuel Test Subassembly does not, in fact, meet the lower reactivity limit will not require the repositioning of the subassembly.

G. Nuclear Safety

No Fuel Test Subassembly will be accepted for irradiation in the Fermi reactor unless it meets the following criteria of nuclear safety:

1. Complete loss of flow to the specimens in a Fuel Test Subassembly, resulting in gross melting and movement of all fissile material to the center plane of the core, shall not cause a reactivity increase of more than 50 cents.
2. Reactivity effects from fuel shifting or slumping in the fuel specimens, or from other mechanisms that could cause contraction or motion toward the core center plane, shall not exceed

3.3 cents per Fuel Test Subassembly. At APDA's discretion, this specification may be relaxed for selected Fuel Test Subassemblies as long as the total reactivity insertion from maximum credible contractions simultaneously occurring in all Fuel Test Subassemblies in the reactor does not exceed 50 cents. It is Euratom's responsibility to furnish APDA with a model which in Euratom's opinion constitutes the maximum credible contraction model for each specimen type within its Fuel Test Subassembly loadings.

3. The effect of the Fuel Test Subassembly on an adjacent core subassembly must be such that neither the thermal output of the core unit nor the power to flow ratio is increased by more than 5 per cent.

4. The value of k_{eff} of a Fuel Test Subassembly in an infinite water pool shall not exceed 0.6.

H. Lattice Positions Available

Only lattice positions whose outlet sodium temperature can be monitored by an operable thermocouple in the hold-down mechanism are available for receiving a newly charged Fuel Test Subassembly.

After startup and a sufficient operating period to confirm predicted temperature behavior, the Fuel Test Subassembly may at the option of APDA be moved to an instrumented or uninstrumented position in the same flux group or at the option of APDA may remain in the same lattice position if the thermocouple has failed.

J. Double Containment +)

To reduce to a minimum the possibility of fission product release to the primary sodium and/or cover gas systems during irradiation, double containment will be required for specimens which fall in any of the following categories:

+) Double containment, should it be required, implies two distinctly separate walls. The walls may be in physical contact (but not metallurgically bonded), or they may be separated by a gap filled entirely with a gas or with a liquid metal.

1. Sodium Compatibility

   Double containment will be required if the specimen cladding is of such material operating at such temperatures that its properties would be adversely affected by the coolant sodium or its impurities. Appropriate sodium compatibility data for specimen claddings used in tests must be provided.

2. Steam and Water Compatibility

   Double containment will be required to protect any specimen cladding considered vulnerable to steam cleaning or water pool storage conditions following the specimen irradiation period.

3. Risk of Fission Product Release

   Under "Irradiation Conditions Criteria," the allowable burnup for singly-clad fuel specimens is defined. If these limits are to be exceeded, double containment will be required.

Euratom must furnish appropriate data of compatibility of the outer jacket material with the internal material in direct contact with it. Euratom must also establish that the outer jacket can withstand the impact of a sudden burst failure of the contained specimen if such a failure mechanism is plausible.

Results of eddy current (or similar) inspection for liquid metal nonbonds between fuel specimen and outer jacket must be supplied along with an analysis of the extent of temperature peaking caused by the largest detected bond void or, if non are found, by the largest bond void undetectable by the sensitivity of the testing apparatus.

K. Vented Elements

Assured containment will be required on any specimen designed to vent fission products. That is, there must be one barrier between the vented fission gas and the Fermi primary coolant.

L. Radioactivity of As-Received Materials

Materials received at the Fermi plant must pass a smear test for alpha activity. The test consists of wiping the entire fuel specimen with soft paper tissue and surveying the tissue with an alpha monitor. The acceptance level shall be less than 100 d/min on the tissue.

The sum of gamma and/or neutron activity from an assembled test loading of one Fuel Test Subassembly shall not exceed 50 mrem/hour at 1 foot.

M. Quality Control

Destructive and nondestructive test results and the quality control practices used in the fabrication of irradiation fuel specimens shall be supplied to APDA for review. Of particular interest are:

1. Cladding and Structural Materials
   a. Certification of material including chemical analyses and mechanical properties.
   b. Pressure test results on samples of tubing used for cladding, and sample end closures.
   c. Eddy current and/or ultrasonic test results.
   d. Dimensional inspection.
2. Fissionable Materials
   a. Chemical analysis.
   b. Isotopic analysis.
   c. Metal/oxygen ratio of oxide materials.
   d. Density and density uniformity and precision of measurements.
3. Assembled Fuel Element
   a. Inspection of seal integrity by dye penetrant and helium leak testing and precision of test.
   b. Weight, dimensions and other gross physical characteristics.
   c. Gamma scanning to check location and variation of fissile material content and axial density uniformity.
   d. Alpha contamination smear test.
   e. Eddy current testing of sodium bond if applicable.

N. Identification and Loading Instruction

It is Euratom's responsibility to furnish each specimen with a permanent legible identification on the upper end cap. Written instructions for loading of the specimens must accompany the shipment. The required instructions consist of specifying, by their end cap identifications, the specimen groupings in each thimble of the Fuel Test Subassembly, and the lattice zone for the Subassembly. All correspondence relating to specimen descriptions and quality control testing should refer to the individual specimens by their end cap identity.

II. Irradiation Conditions Criteria

A. Operating conditions

The operating conditions of the reactor, during the Euratom contract period, will be as follows:

1. Power

The "equilibrium" 105-subassembly core will be operated at a power level of 110 Mwt. The operational limits for controllability, with best current knowledge, are ± 3.0 per cent. It is anticipated that the power will be known to ± 5 per cent based on heat balance data. This accuracy will be established with greater confidence after the heat balances are run at 200 Mwt.

During reactor operation at power for fuel testing, the temperatures, sodium flow and power will be maintained as constant as is practical. It is intended that the irradiation environment of the specimens will be held reasonably constant. If the number of subassemblies in the core increase with the addition of tests, every reasonable effort will be made to maintain constant power densities in the test subassemblies. A reasonable effort will also be made to move fuel test subassemblies in the reactor core to compensate for burnup in the test elements if this is desired. The introduction during the period of this agreement of any fuel tests into the Plant may not change the power generation or the sodium temperature rise along the length of the Euratom specimens by more than 10 per cent from that calculated from these criteria; that is, 110 Mwt in the 105-subassembly core.

2. Sodium Inlet

Nominal 450 F. The operational limits for controllability, with best current knowledge, are ± 5 F. In addition, the absolute temperature is believed to have an accuracy of ± 2 F, which will be confirmed during 200-Mwt operation.

Before the orifices for the Euratom test thimbles are selected, the reactor inlet temperature may be increased up to 85 F to improve test conditions.

B. Allowable Burnup

The residence time of a fuel specimen in the reactor will be limited to reduce the possibility of fuel failure and fission product release. It is recognized that no criteria can presently be written which would guarantee that fuel would not fail. However, examination of prior irradiation experience, consideration of solid fission product swelling and consideration of fission gas pressure can all be used as guides to limit the allowable burnup of Euratom fuel specimens. Therefore, the burnup allowed will be the lowest value of the three following criteria:

1. Prior Irradiation Experience

A thorough literature search containing abstracted results, favorable or unfavorable, of previous irradiations pertinent to the proposed test in Fermi must be made. In the final analysis, the containment integrity at the allowable burnup of Euratom's proposed test must be shown to be a tenable extrapolation of previous testing.

2. Solid Fission Product Swelling

Double containment will be required if the burnup is greater than the burnup obtained by the following formulae, which take into account the swelling of the fuel material within the cladding.

$$\text{Burnup} = \frac{100-p_s}{A} \times 10^{20} \text{ fissions/cc of fuel assuming the fuel occupies area up to inside diameter of cladding}$$

$$= \frac{100-p_s}{B} \times 10^{4} \text{ MWD/metric ton of oxide}$$

where: p_s = smeared density in per cent of theoretical.

$A \cong 0.7$ for oxide fuel.

$B = 1.81$ for oxide fuel.

3. Gaseous Fission Products

The gaseous fission products act to produce a pressure stress in the clad. This stress and consequently the burnup will be limited by the following clad stress criteria which must be applied along the entire length of the specimens.

a. $1/4\ \sigma_t + \sigma_p < \sigma_y$

b. $\sigma_t < 2\ \sigma_y$

c. $\sigma_p < 3/4\ \sigma_y$

d. Secondary creep $<$ 0.1 per cent.

where: σ_t = thermal stress calculated elastically for conditions of 130 per cent of full power.

σ_p = pressure stress calculated elastically for fission gas released from the fuel to the gas storage area+) at nominal removal burnup with the gas at a temperature ++) characteristic of 130 per cent of full power.

σ_y = 0.2 per cent offset yield point for unirradiated fully annealed material at a hot channel temperature characteristic of 130 per cent of full power.

Secondary creep calculated using unirradiated creep strength at hot channel temperatures using the load created by 100 per cent fission gas release (for oxide materials) integrated over the period from zero burnup to nominal removal burnup.

It is Euratom's responsibility to furnish calculational demonstration of adherence to a, b, c, and d above. The effective cladding thickness which enters the structural analyses will not be the preirradiation nominal cladding

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+) For oxide materials assume 100 per cent gas release.

++) For APDA six-pin support system design use nominal temperature. For all others use hot channel temperature unless a relaxation is specifically approved by APDA.

thickness but a value which makes justifiable allowance for fabrication tolerance on wall thickness and material wastage due to sodium corrosion and mass transport at hot channel temperatures over the life of the test.

4. Double Containment

It is Euratom's responsibility when supplying a fuel specimen in an outer containment jacket to provide calculational demonstration that the design of the outer jacket meets specifications II.A.3.a, b, c, and d above. For this calculation, the temperatures and heat fluxes corresponding to 130 per cent of full power must be assumed. The pressure stress on the outer jacket must assume fission gas release from the fuel +) and direct communication between the gas reservoirs of the outer jacket and the fuel specimen.

C. Specimen Failure

It is believed that in a relatively clean Fermi cover gas system a failure of an oxide test element, releasing its accumulation of stored fission gas, would be detectable by existing monitoring instruments. Should this occur and the continuation of reactor operation be deemed hazardous, the reactor will be shut down until the Fuel Test Subassembly containing the failed element is located and removed from the reactor.

In the effort required to locate the subassembly with the leaking element, APDA reserves the right to discharge any Fuel Test Subassemblies suspected of containing the leaking element until the actual leaker is located. Since it is not feasible technically to reinsert sodium bonded capsules, Fuel Test Subassemblies containing these will be given special consideration when searching for the leaking subassembly and will be removed from the reactor only if all other reasonable possibilities have been explored. Reinsertion of non-sodium bonded pins or capsules will be accomplished if feasible and advisable.

+) For oxide materials assume 100 per cent gas release.

D. Sodium Compatibility

Proper evidence must be presented that demonstrates that the cladding will not be adversely affected over the time required to achieve the burnup determined in II.A. above; specifically cladding considered vulnerable to carburization must be operated at less than 850 F unless double jacketed or unless specific tests conducted in the Fermi primary system sodium show that the material will not, in fact, carburize significantly during the irradiation period.

5. Special Requirements by the APDA Criteria

The contents of this chapter especially refer to the requirements given by APDA Criteria for Euratom Fuel Specimens (see chapter 4). Such references are marked in the headlines.

5.1 Test Vehicle (I.A)

The irradiations shall be conducted in the standard Fuel Test Subassembly with four thimbles. For the support system we take the APDA design provided for 6 test pins with a diameter of 6.35 mm in each thimble.

5.2 Specimen Description (I.B.1 and I.B.2)

The data proposed there have been accepted and are the basis for our design. It is represented by the general drawing TA-2K-3-07-02-1755 giving a survey, and a set of special technical drawings from which detailed data can be picked up, see chapter 3.1 and Appendix C.

5.3 Maximum Sodium Temperature (I.C)

The maximum local sodium temperature at the specified sodium inlet temperature of 450°F (= 232°C) will be 1110°F (= 600°C, see 3.3). The following information is given for the hot channel factor analysis:

C.1 The nominal cladding outer diameter is 6.35 ± 0.065 mm. This tolerance is necessary because the inner diameter has to be of lowest tolerances for the required fits.

C.2 The nominal over-all length is at both pin types 1286.3 $\pm$ 1.0 mm. The fueled length at Type I is 700.0 $\pm$ 0.5 mm, at Type II 774.7 $\pm$ 0.5 mm. For further details see drawings.

C.3 The anticipated swelling is expected not to be greater than 150 microns with pellet fuel. This clearance will be consumed mainly by thermal expansion. Fission product induced swelling will be very low, because of the available porosity in the fuel. The behaviour of vibratory compacted fuel is expected to be similar.
The diametral swelling of the clad will be not more than 1 %, caused only by thermal expansion of the cladding material.

C.4 Reactor lattice zone: 1900 to 1950 Ø/day category according to Fig.1.

C.5 Isotopic composition see Specifications, 6.1.2.

C.6 APDA design is selected.

5.4 Exit Sodium Temperature Matching (I.D)

This point has to be met by APDA responsibility.

5.5 Mismanagement (I.E)

All four thimble orifices can be the same, as coolant flow rates are practically identical, i.e. 1800 and 1810 kg Na per hour and thimble, respectively, see 3.3.

This condition concerning an inadvertent mismanagement is fulfilled, because the local power difference between the chosen 1900 Ø test position and the central lattice position is very low.

5.6 Test Subassembly Reactivity Limitations (I.F)

The reactivity of an EFFBR core subassembly at 1900 - 1950 Ø-position is about $\Delta k/k = 63 \cdot 10^{-4}$ = 86 cents, at an inner radial blanket position about $\Delta k/k = 28 \cdot 10^{-4}$ = 39 cents. The test subassembly with 24 test pins as proposed has a reactivity value at 1900 - 1950 Ø-position of about $\Delta k/k = 60.8 \cdot 10^{-4}$ = 83 cents (see Appendix A.1).

F.1 Upper Limit: Thus the reactivity of the test subassembly does not exceed that of the core subassembly, which it replaces.

F.2 Lower Limit: Thus the reactivity of the test subassembly is sufficiently high, that no net decrease takes place.

5.7 Nuclear Safety (I.G)

G.1 The gross melting and movement of all the fuel of the test subassembly to the center plane of the core causes a reactivity increase of 36 cents; for details see Appendix A.1.

G.2 According to the shift model, established and calculated in Appendix A.1, the reactivity increase for maximum credible fuel contractions is not more than 3.0 cents.

G.3 This condition must be checked by APDA.

G.4 The value of k_{eff} of the fuel test subassembly in an infinite water pool is supposed to meet this condition. As anounced the definite evaluation will be provided by APDA.

5.8 Lattice Position Available (I.H)

The APDA proposal is accepted.

5.9 Double Containment (I.J)

Compatibility Problems

To prove that double containment is not necessary, the compatibility problems are evaluated as follows:

Sodium Compatibility (I.J.1)

The corrosion behaviour of X8CrNiMoVNb1613 and Inconel 625 apparently is similar to each other in sodium below 700°C as was published on Geneva Conference 1964 in papers 28/P/244 (USA) and 28/P/343 (USSR). The American paper gives 40 microns/year corrosion for 316 S.S. at 650°C in a loop and furthermore says, that very little corrosion of any kind was found below 700°C with Ni-base alloys. The Russian paper (on page 4) says, that all such materials behave in the same way even up to 750°C. Another paper given in Trans ANS 8 (1965) (by GE) reports corrosion to a depth of 175 microns at 650°C in 25 000 h for 316 S.S.

From the Geneva Conference papers and this GE-publication it is concluded, that the corrosion will not be more than 50 microns in the proposed irradiation time at temperatures up to 600°C, which is the maximum sodium temperature.

Concerning the mass transport problem we state that as a result of our own corrosion experiments in a sodium loop with Interatom, Bensberg, we found a corrosion rate of less than 0.01 mg/cm^2 month for both types of material at 525°C. So we conclude that both 16/13 CrNi S.S. and Inconel 625 should be admissible for a long stay in sodium under these conditions.

Steam and Water Compatibility (I.J.2)
Both Inconel 625 and S.S. 16/13 CrNi are rather new materials which have not yet been tested for all the qualifications being required in their practical use. So we have to answer questions for steam and water compatibility somewhat in general:

<u>Inconel 625</u>

a) Steam and water compatibility
Inconel 625 is regarded to be compatible with steam up to 800°C according to the following experiences and publications:

GfK experiments
J. Electrochem. Soc. 111 (1964), 1116

b) NaOH-compatibility
The NaOH-compatibility is regarded to be better than of pure Ni which is a material for concentrating NaOH in industry. Reports are given in the brochure of Intern. Ni-Comp., Germany:

"Alkaline-resistivity of Ni and Ni-alloys".

<u>X8CrNiMoVNb1613</u>

a) Steam and water compatibility
No experimental data are available yet.

b) NaOH-compatibility
16/13 CrNi-steel is regarded to be invulnerable to NaOH because of its Ni-contents, reported also by Mannesmann AG:

"ABC of Steel Corrosion".

5.10 <u>Radioactivity of As-Received Materials (I.L)</u>

The specified conditions are accepted.

5.11 <u>Quality Control (I.M)</u>

The quality control will be performed according to all items requested. All the controls are included in the Specifications, Chapter 6 below.

5.12 <u>Identification and Loading Instruction (I.N)</u>

The identification is explained in 2.3 and in the specifications, Chapter 6. Each thimble of the test subassembly has to be loaded with the 6 pins of one batch as identified in 2.3. Additional loading instructions will be supplied together with the shipment of the test pins.

5.13 <u>Operating Conditions (II.A)</u>

We agree upon the given operating condition as specified in 1. (Power) and 2. (Sodium Inlet).

5.14 <u>Prior Irradiation Experience (II.B.1)</u>

From the compilation of all appropriate data available, which was produced in cooperation with CEA/France and Transuranium Institute/ Karlsruhe (see Appendix A.2) we conclude that there can be expected a life time at the proposed rod power (635 watt/cm max.) of at least 50 000 MWd/t without failure.

5.15 <u>Solid Fission Product Swelling (II.B.2)</u>

As the highest smear density is 88 % th.d. we get out of the given formula an allowable burnup of 63 000 MWd/ton which is more than the scheduled burnup.

5.16 <u>Gaseous Fission Products (II.B.3)</u>

Cladding Stress Analysis

The conditions to avoid double containment are

a) $1/4\ \sigma_t + \sigma_p < \sigma_y$

b) $\sigma_t < 2\ \sigma_y$ (8)

c) $\sigma_p < 3/4\ \sigma_y$

d) Secondary creep $<$ 0.1 %.

For the first 3 conditions the necessary stresses σ_t and σ_p are calculated in Appendix A.3, while the creep calculation is carried out in Appendix A.4.

Basic data - like σ_y, E, ß and r - for the calculations were taken from literature mentioned below. The data of cladding temperatures are to be found in Fig.6.

The results for the S.S. show, that the conditions a, b and c are well fulfilled. σ_p was calculated at 580°C average gas temperature, 100 % gas release and reduced wall thickness of 0.35 mm caused by assumed sodium corrosion (see also Chapter 5.9).

The σ_y-data for Inconel 625 are rather constant at 31 kp/mm^2 over the total temperature range. The maximum value of $\sigma_p + 1/4\ \sigma_t$ is 14.4 kp/mm^2 with σ_p of 9.45 kp/mm^2.

The creep analysis data are uncritical for both materials. The total secondary creep for 225 full power operations days is at both claddings about 0.0003 %.

These data are very conservative, because they have been calculated under radial stress aspects only. In the actual case, biaxial calculations would meet reality more and reduce the given results here by a factor of about 0.4.

Literature for physical and mechanical data:

X8CrNiMoVNb1613: Landolt-Börnstein, Zahlenwerte und Funktionen, Band Technik, Teil 2a, Springer-Verlag Berlin/Göttingen/Heidelberg, 1963, S.518

Mannesmann AG: Druckschrift "Erzeugnisse aus warmfesten Stählen", Ausgabe 1966

Inconel 625: Inconel alloy 625, Preliminary Data + Supplementary Data, Huntington Alloy Products Division, Huntington 17, West Virginia

5.17 Sodium Compatibility (II.D)

Carburization Problem

Specific carburization tests conducted in the EFFBR primary system sodium have shown that both cladding material do not carburize significantly. The 1200 F - 1000 h - exposure of the X8CrNiMoVNb 1613 and Inconel 625 showed a carbon pickup within acceptable limits (smaller than the nominal guidelines of 0.5 % C).

6. Specifications

A total of 40 fuel pins are to be produced according to these specifications. Of this total, 36 are to be irradiated in batches of 12 each while 4 are reserve pins. The fuel pins consist of 6.35 $\pm$ 0.065 mm O.D. rods either with 0.4 mm thick Inconel 625 or 0.4 mm thick X8CrNiMoVNb 1613 stainless steel cladding. The fuel, UO_2 - 15 weight % PuO_2, will be provided either in the form of fuel pellets with densities of 88 and 93 % of theoretical, or in the form of powder, vibratory compacted to 85 % theoretical density.

6.1 Fuel Pellets and Vibratory Powder

The fuel shall be uranium-plutonium mixed oxide prepared from ceramic grade UO_2 and oxalate precipitated PuO_2 powders, either in the form of pellets or in the form of sintered powder particles for vibratory compaction. The chemical composition of the UO_2 and PuO_2 powders shall be such as to meet the specified chemical purity of the fuel pellets.

6.1.1 Chemical Composition: 15.0 $\pm$ 0.1 weight % PuO_2
85.0 $\pm$ 0.1 weight % UO_2

6.1.2 Isotopic Composition:

in the Pu-element:	Pu 239	90.86 $\pm$ 0.15 w/o
	Pu 240	8.22 $\pm$ 0.15 w/o
	Pu 241	0.88 $\pm$ 0.15 w/o
	Pu 242	0.04 $\pm$ 0.01 w/o
in the U-element:	U 235	93.00 $\pm$ 0.20 w/o
	U 238+)	7.00 $\pm$ 0.20 w/o

+) This U 238 figure contains also the unknown U 234 content. The supplier of the source material is the USAEC.

6.1.3 Fabrication of pellets: Pressed and sintered pellets prepared from mechanically mixed powder.

6.1.4 Fabrication of vibratory powder: Rounded, sintered; powder particles prepared by pressing mechanically mixed powder, crushing and then milling.

6.1.5 Density:

Pellets for 7 fuel pins	-	93 $\pm$ 1 % of theoretical density
Pellets for 13 fuel pins	-	88 $\pm$ 1 % of theoretical density
Powder for 20 fuel pins	-	suitable for vibratory compaction to 85 $\pm$ 2 % theoretical density.

6.1.6 Pellet Diameter: 5.40 $\pm$ 0.01 mm

Average diametral clearance between cladding and pellets shall be 150 $\pm$ 10 µ. The entire cylindrical surface of all pellets shall - if necessary - be ground to produce the required fit.

6.1.7 Stoichiometry: The atomic oxygen to metal ratio shall be 2.0 $\pm$ 0.015.

6.1.8 Chemical purity: Total impurity content of the fuel pellets and vibratory powder shall not exceed 2500 ppm with the following specific limitations.

Impurity	Maximum Permissible Limit ppm
C	150
Ca	25
Cl	25
F	25
Mg	25
N	100
Moisture	100

The total gas content of the pellets shall be such, that the maximum volume of all gaseous products evolved from the pellets when heated to 1650°C shall not exceed 40×10^{-6} liters/gram.

6.1.9 Homogeneity: Ceramographic examination is to reveal no PuO_2 particles with a diameter greater than 50 µm.

6.1.10 Pellet Appearance: On the pellet surface the following defects are acceptable:

a) Chips on the pellet edges, at either end of the pellet, up to a maximum of 0.3 mm length in either radial or axial direction.

b) Hairline cracks in any number and direction up to a maximum of 0.1 mm in width, and 2 mm in length.

c) Surface pits in any number up to 1.0 mm diameter.

6.2 Blanket Pellets

Natural UO_2 pellets will be used to form a blanket and insulator pellet in both the vibratory-compacted and pellet-containing fuel pins. These will be prepared and be of such a quality as to meet the same specifications applied to the fuel pellets, except as to composition. The insulator pellets however will be of 88 % th.d. natural UO_2.

6.3 Cladding

Inconel 625 and S.S. X8CrNiMoVNb 1613 (Werkstoff-Nr. 4988) seamless tubing shall be supplied in the form required to fulfill the specifications of the final fuel pins given in Section 6.4 below. In addition it shall meet the following specifications:

6.3.1 Condition of Heat Treatment: As annealed

6.3.2 Maximum Grain Size: 2.5 grains/cm^2 at 100 X (ASTM Grain Size No.5)

6.3.3 Mechanical Properties: at room temperature

		Inconel 625	X8CrNiMoVNb 1613
Yield Strength, kg/mm^2			
	max	60	50
	min	35	25
Tensile Strength, kg/mm^2			
	max	105	80
	min	75	55
Reduction in Area, %			
	min	30	35

6.3.4 Chemical Analysis: The Inconel 625 and the S.S. shall conform to the following analyses:

	Inconel 625	X8CrNiMoVNb 1613
Nickel	Balance	12.5 - 14.5 %
Chromium	20 - 23 %	15.5 - 17.5 %
Molybdenum	8 - 10 %	1.1 - 1.5 %
Niobium	3.15 - 4.15 %	>10 X C <10 X C + 0.4 % max. 1.2 %
Vanadium	-	0.65 - 1.2 %
Iron	5.0 max.	Balance
Manganese	0.50 % max	1.0 - 1.5 %
Sulfur	0.015 % max	-
Carbon	0.10 % max	0.10 % max
Silicon	0.50 % max	0.30 - 0.60 %
Cobalt	0.10 % max	-
Aluminium	0.40 % max	-
Titanium	0.40 % max.	-
Nitrogen	-	0.10 %

6.3.5 Surface Roughness: Not to exceed 2 µm

6.3.6 Surface Grooves and Defects: Not to exceed 2 µm

6.3.7 Surface Cracks and Seams: None permitted

6.3.8 Inner Defects and Discontinuities:
Not to exceed 10 % of the wall thickness as determined by ultrasonic tests.

6.3.9 Inclusions and Stringers:
Microscopic examination of transverse cross sections of tubing shall reveal individual inclusions and stringers neither with any dimension larger than 0.05 mm nor with an area greater than 7×10^{-4} mm^2.
Furthermore, in any 0.25 mm^2 longitudinal or transverse area, inclusions and stringers shall constitute no more than 2 % of that area.

6.3.10 Ovality: Within the maximum and minimum inside diameters.

6.3.11 General Appearance: Inside and outer surface shall be free from discoloration, grease, dirt, metal particles, and other foreign matter.

6.4 Fuel Pins

The fuel pin design is detailed in the attached drawings. It varies with cladding material with respect to division of the fuel-pin inner length into gas plenum, blanket and active fuel regions. Only one blanket region, below the upper gas plenum, is provided. Overall pin lengths and diameters are identical.

6.4.1 Dimensions

a) Inner Diameter: 5.55 $\pm$ 0.025 mm
b) Wall Thickness: 0.4 $\pm$ 0.02 mm
c) Overall-Pin Length: 1286.3 $\pm$ 1.0 mm
d) Inner-Pin Length: 1241.4 $\pm$ 1.0 mm
e) Division of Inner-Pin Length:

	S.S. Clad Pins	Inconel 625 Clad Pins
Upper gas plenum	188.4 mm	63.7 mm
Blanket	150.0 mm	200.0 mm
Active Fuel Zone	700 mm	774.7 mm
Insulator Pellet	8.0 mm	8.0 mm
Lower gas plenum	195.0 mm	195.0 mm

f) Bowing: 1 : 1500 per each 300 mm length

g) Fuel Column Support: Support for the fuel column shall be provided within the gas-plenum regions in the form of 5.4 mm diameter cylindrical inserts made from the same material as the claddings. A spring is included in the upper gas-plenum region to minimize fuel movement during transportation.

6.4.2 Helium Filling

The fuel pins are to be filled with 1 atm of helium.

6.4.3 Welding

End caps are to be welded by means of non-consumable electrode, inert-gas arc welding or electron beam welding. A maximum helium leak rate of 10^{-6} torr liter/sec is permissible.

6.4.4 Identification

Permanent markings are to be made on the upper end plugs with the following designations:

7 SS clad pins, 88 % dense fuel pellets - A1, A2, C1, C2, E1, E2, X1

3 SS clad pins, 93 % dense fuel pellets - A3, C3, E3

10 SS clad pins, 85 % dense vibro fuel - A4, A5, A6, C4, C5, C6, E4, E5, E6, X2

6 Inconel clad pins, 88 % dense fuel pellets - B7, B8, D7, D8, F7, F8

4 Inconel clad pins, 93 % dense fuel pellets - B9, D9, F9, X3

10 Inconel clad pins, 85 % dense vibro fuel - B10, B11, B12, D10, D11, D12, F10, F11, F12, X4

6.5 Examination and Testing

Detailed examinations and tests are to be performed at each stage of fuel pin production. The following are required:

6.5.1 Starting Powder Material

a) Isotopic Analysis

b) BET-Surface Area Measurements

6.5.2 Fuel Pellets + Vibratory Particles

a) Density Measurements - every 10th pellet

b) Ceramographic Examination of Representative Pellets and Particles with Microphotography

c) Dimensional Measurements of Pellets for Loading Pins

d) Determination of Oxygen/Uranium Ratio

e) Chemical and Gas Content Analysis

f) Visual and Macroscopic Examination

6.5.3 Cladding Material

a) Dimensional Measurements

b) Tensile Test

c) Visual Examination

d) Chemical Analysis

e) Metallographic Examination with Microphotography - transverse and longitudinal section from end of each pin.

f) Ultrasonic Test - each pin

g) Pressure Test

6.5.4 Completed Fuel Pin

a) Dimensional Measurements

b) X-ray Examination of Weld Zone

c) Density Determination of Vibratory Compacted Pins

d) Leak Test

e) Weight Determination

6.5.5 Production and Testing Certification

A record shall be kept for the purpose of describing and certifying the production procedures employed in producing the fuel pellets, fuel particles for vibratory compaction, the cladding, and fuel pins and the procedures used and results obtained in examining and testing the pellets, particles, cladding, and fuel pins.

Appendix A 1

Reactivity Calculations for the Fuel Test Subassembly

For the "Test Subassembly Reactivity Limitations" and the conditions concerning "Nuclear Safety" in the Final Criteria of APDA the following calculations are supplied. To obtain the reactivity contribution of the different isotope oxides we use the diagrams of APDA, see Fig.10,11,12 and 13. In these diagrams the reactivity values of an oxide column of 100 °/o theoretical density and 1 cm^2 cross section is plotted versus the height of the column, each curve for a single isotope oxide. The steps of the calculations are:

a) Evaluate the geometrical cross section of the single isotope oxide in the given fuel columns. The sum of the values for the 4 different oxides (U 235, U 238, Pu 239, Pu 240) is the total geometrical cross section of all the fuel in the subassembly.

b) Recalculate these single geometrical cross sections to 100 °/o oxide density at unchanged height of column ($\hat{=}$ distance of core midplane).

c) Take the integrated $\Delta k/k$ values (per cm^2) out of the curves (Fig. 10-13) and multiply by the 100 °/o geometrical cross sections resulting in the 4 single reactivity contributions.

d) Sum up the 4 contributions to get the total reactivity value for the whole subassembly.

1. Fuel Composition and Density

Chemical composition: 85 w/o UO_2
15 w/o PuO_2

As isotopic composition we add the Pu 241 content to Pu 239 and take as average values for these calculations

in the Pu element :	92 w/o	Pu 239
	8 w/o	Pu 240
in the U element:	93 w/o	U 235
	7 w/o	U 238

With regard to the densities of the oxides we get as fuel composition in volume-$^{o}/o$:

79.6 v/o $U^{235}O_2$

6.0 v/o $U^{238}O_2$

13.2 v/o $Pu^{239}O_2$

1.2 v/o $Pu^{240}O_2$

As an average value for all 24 fuel pins we take 85 $^{o}/o$ of theoretical oxide density (smear density). Furthermore only pins of Type II with a fuel length of 77.5 cm are assumed being the most conservative case.

2. Geometrical Cross Section Distribution

Fuel diameter	5.55 mm
Fuel cross section of 1 pin	0.242 cm^2
of 24 pins	5.81 cm^2

This total fuel cross section is distributed to the single oxides according to the composition in v/o with the results:

$U^{235}O_2$	4.625 cm^2
$U^{238}O_2$	0.348 cm^2
$Pu^{239}O_2$	0.767 cm^2
$Pu^{240}O_2$	0.070 cm^2
total :	5.810 cm^2

3. Reactivity in the Original State

The calculational values are compiled in

Table XI   Reactivity Values in the Original State

Oxide	Cross Section at 100 °/o Density (cm^2)	$\Delta k/k$ per cm^2 for half height of 38.75 cm (cm^2)	$\Delta k/k$ per cm^2 for total height of 77.5 cm in 10^{-4}	$\Delta k/k$ for the given cross section in 10^{-4}
$U^{235}O_2$	3.930	+ 5.98	+ 11.96	+ 46.90
$U^{238}O_2$	0.296	- 0.11	- 0.22	- 0.07
$Pu^{239}O_2$	0.648	+ 10.54	+ 21.08	+ 13.70
$Pu^{240}O_2$	0.060	+ 2.32	+ 4.64	+ 0.28
total:	4.934			+ 60.81

Thus the reactivity value of the whole fuel test subassembly in the original state is

$$\Delta k/k = 60.8 \cdot 10^{-4} \mathrel{\hat{=}} 83 \text{ cents}$$

with the conversion 1 cent $\hat{=}$ $73 \cdot 10^{-6}$.

4. Melt Down State

All the fuel is assumed to be concentrated in the center of the core, distributed onto the whole cross section of the subassembly equally on both sides of the midplane. It is calculated with 100 °/o room temperature fuel density, i.e. no allowance was made for thermal expansion and expansion on melting. Therefore we have the most conservative case.

The total "room temperature" volume of the fuel is:

$$77.5 \text{ cm} \times 5.81 \text{ cm}^2 \times 0.85 = 383 \text{ cm}^3.$$

At a subassembly cross section of 46.6 cm^2 the fuel height in the molten condition is 8.22 cm, that means 4.11 cm on each side of the midplane.

The calculational values for this state is given in Table XII.

Table XII Reactivity Values in the Melt Down State

Oxide	Cross Section at 100 °/o Density (cm^2)	$\Delta k/k$ per cm^2 for half height of 4.11 cm in 10^{-4}	for total height of 8.22 cm in 10^{-4}	$\Delta k/k$ for the given cross section in 10^{-4}
$U^{235}O_2$	37.10	+ 0.90	+ 1.80	+ 66.80
$U^{238}O_2$	2.80	- 0.03	- 0.06	- 0.17
$Pu^{239}O_2$	6.15	+ 1.60	+ 3.20	+ 19.70
$Pu^{240}O_2$	0.56	+ 0.34	+ 0.68	+ 0.38
total:	46.61			86.71

Hence the reactivity of the subassembly in the melt down state is

$$\Delta k/k = 86.71 \cdot 10^{-4} \triangleq 119 \text{ cents.}$$

The reactivity difference between melt down and original state is then 36 cents.

5. Fuel Shift Model

For the deformation of the fuel we assume a model, in which the original fuel is at first slowly expanded by 1 °/o in length and then in a sudden power transient or coolant interruption densifies to 90 °/o of room temperature density.

The maximum densification takes into account, that all the fuel melts down while the cladding (hypothetically) is not affected. In the original state the fuel has a density of 85 °/o th.d. and a volume V. In the molten condition the porosity of 0.15 x V has disappeared, but the expansion on temperature rise and melting consumes also about 0.10 x V, see below. Hence the density in the molten state cannot be more than 89.5 °/o of th.d. of original state. Even if the volume change at melting is less, it is not necessary to assume in the shift model a higher density than 90 °/o th.d. For in realistic cases, where this model is applicable, only a part of the fuel is molten, because at total melting the whole pin including clad would be destroyed.

6. Expanded State in the Shift Model

The fuel length expanded by 1 °/o is 78.3 cm, density then 84.2 °/o th.d. The length increase is added to the upper half height of the test pins, the lower half height remains unchanged.

Table XIII Reactivity Values for the Expanded State in the Shift Model

Oxide	Cross Section at 100 °/o Density (cm^2)	$\Delta k/k$ per cm^2 for half height of 38.75 cm in 10^{-4}	$\Delta k/k$ per cm^2 for half height of 39.55 cm in 10^{-4}	$\Delta k/k$ per cm^2 for total height of 78.3 cm in 10^{-4}	$\Delta k/k$ for the given cross section in 10^{-4}
$U^{235}O_2$	3.885	+ 5.98	+ 6.05	+ 12.03	+ 46.81
$U^{238}O_2$	0.312	- 0.11	- 0.11	- 0.22	- 0.07
$Pu^{239}O_2$	0.648	+ 10.54	+ 10.65	+ 21.19	+ 13.71
$Pu^{240}O_2$	0.048	+ 2.32	+ 2.33	+ 4.65	+ 0.22
Total:	4.893				60.67

The reactivity of the subassembly in the expanded state is

$$\Delta k/k = 60.67 \cdot 10^{-4}.$$

7. Condensed State in the Shift Model

During densification the lower half height of the pins remains unchanged again, the upper half height is affected by the length decrease. The total fuel length is 73.2 cm corresponding to a density of 90 °/o th.d.

Table XIV Reactivity Values for the Condensed State in the Shift Model

Oxide	Cross Section at 100 °/o Density (cm^2)	$\Delta k/k$ per cm^2 for half height of 38.75 cm in 10^{-4}	$\Delta k/k$ per cm^2 for half height of 34.45 cm in 10^{-4}	$\Delta k/k$ per cm^2 for total height of 73.2 cm in 10^{-4}	$\Delta k/k$ for the given cross section in 10^{-4}
$U^{235}O_2$	4.155	+ 5.98	+ 5.66	+ 11.64	+ 48.34
$U^{238}O_2$	0.312	- 0.11	- 0.12	- 0.23	- 0.07
$Pu^{239}O_2$	0.696	+ 10.54	+ 9.90	+ 20.44	+ 14.23
$Pu^{240}O_2$	0.072	+ 2.32	+ 2.22	+ 4.54	+ 0.33
Total:	5.235				62.83

The reactivity in the condensed state is then

$$\Delta k/k = 62.83 \cdot 10^{-4}.$$

8. Reactivity Increase in the Shift Model

The reactivity difference between the condensed and expanded state is

$$(62.83 - 60.67) \cdot 10^{-4} = 2.16 \cdot 10^{-4} \mathrel{\hat{=}} \underline{\underline{2.96 \text{ cents.}}}$$

Literature:

J.A. Christensen
Thermal Expansion and Change in Vol. on Melting for UO_2
HW-75148

Appendix A 2

Prior Irradiation Experience

Before starting irradiation experiments in the Enrico Fermi reactor, it has become necessary to show that the test pins can be irradiated without any failure, which might be due to fuel element specifications. APDA has asked to provide a curve out of data from irradiation experiments,which already have been performed. This curve might give the upper limit for power and burnup, which have been reached up to the present.

In the following Tables XV and XVI all available data with the most characteristic irradiation conditions and results are compiled. From that a curve was drawn, which indicates the field for safe experiments conservatively. In Fig.14 this curve and the irradiation conditions of the single experiments are demonstrated.

Table XV Prior Irradiation Experience

Experiments without failure

No.	Reference/ Location	Fuel	Enrichment (°/o)	Fuel density (°/o theor.)	Fuel diameter (mm)	Fuel length (mm)	Cladding	Linear Rod Power (W/cm)	Burnup (MWd/t)	Remarks
1	Geneva Conf. 28/P/240 (1964) (GEAP 4264) GE-Vallecitos	UO_2- vibro comp.	1.3	83	~12		Zircaloy-2	2000 max.	10 000	70 °/o of volume, molten, not failed
2	Trans ANS 8, 42 (1965) GE-Vallecitos	UO_2 pellets + vibro comp.	3.8	85 vibr. 93 pellets	~12	750	Zircaloy-2	1500 - 1700 max.	11000-20000	no failure
3	Trans ANS 6, 350 (1963) GE-Hanford	UO_2/PuO_2 pellets + vibro comp.	2.57 PuO_2	91 pellets 65 vibr.	~12			1250, 880 max.	8800, 20000	no failure
4	GEAP 3811 (1962) GEAP 3833 (1962) GE-Vallecitos	UO_2/PuO_2 pellets + swaged	20 °/o Pu 40 °/o U235	90-95 pellets 75 swaged	3.8	40	Stainless Steel	590 max.	up to 90000	no failure
5	Geneva Conf. 28/P/155 (1964) UKAEA	UO_2 pellets	4.5	97	6.35		20/20 CrNi	590-710 max.	29000-44000	no failure
6	Trans ANS 7, 394 (1964) GE San Jose	UO_2/PuO_2 pellets	similar to No. 4	95	3.8	250		640 max.	70 000	no failure, ruption only at higher power rates than given here
7	London Conf. 4B/4 (1966) UKAEA Dounreay	UO_2/PuO_2	79 °/o U235 15 °/o Pu	79 - 85	5 - 7	710	SS M 316 L	400-800 max.	57000-67000	no failure

Table XVI Prior Irradiation Experience

Experiments with some single failure

No.	Reference/ Location	Fuel	Enrichment (°/o)	Fuel density (°/o th.)	Fuel diameter (mm)	Fuel length (mm)	Cladding	Linear rod power (W/cm)	Burnup (MWd/t)	Remarks
8	Trans ANS 6, 351 (1963) GETR	UO_2 molten	1.5 / 2.2	90.5	12	850	Zircaloy-2	850	10	4 rods, 1 failed at 0.8 mm diam. swelling
		vibro comp. and swaged	1.5 2.2	90.5 86	12	850	Zircaloy-2	980	12	2 rods, 1 failed at 2 mm diam. swelling
		UO_2 molten	1.5 / 2.2	87 - 88	12	850	Zircaloy-2	1150	1600	4 rods, no failure 0.4 mm diam. swelling
		vibro comp. only	3.0 / 3.8	83 - 85	12	750	Zircaloy-2	1750	2000	4 rods, no failure 0.2 mm diam. swelling
9	Trans ANS 8, 365 (1965) PRTR	UO_2-PuO_2 vibro comp.	2 PuO_2	85 - 87		1500 total length	Zircaloy-2	640-980	4000	currently operating, max. central temperature 2700°C
10	Trans ANS 8, 424 (1965) EBWR/ANL	UO_2-PuO_2 vibro comp.	1.5 PuO_2	77 - 84	9.5	short specim.	Zircaloy	750 average	20000	no failure at 28 specimens, 4 still operating up to 28000 MWd/T
11	Trans ANS 7, 388 (1964) PRTR	UO_2-PuO_2 vibro comp.	0.5 - 1 PuO_2	80 - 90	more than 4000 specimens of different types		Zircaloy	up to 980	6000 max.	36 specimens failed, by chemical impurities of fuel and material only
12	Private communication/ CEA, France (1966)	UO_2 - PuO_2	11 PuO_2	84 - 98	5.56-5.92	350	304 L 316 L	380-460	4000 - 28 000	6 rods, 2 failed at 22 000 and 27 000 MWd/t

Appendix A 3

Cladding Stress Conditions

For the stress analysis the pressure stress and the thermal stress have been calculated.

The pressure-stress σ_p is calculated with the formula

$$\sigma_p = p \cdot \frac{r_i}{s} \tag{9}$$

$p \mathrel{\hat{=}}$ fission gas pressure (kp/mm^2)
$r_i \mathrel{\hat{=}}$ inner radius (mm)
$s \mathrel{\hat{=}}$ wall thickness $\hat{=}$ 0.35 mm

The fission gas pressure is given by

$$p\,V = n\,R\,T$$

with $T = 850^\circ K$ (average fission gas temperature).

The fission gas amount n is calculated under the following assumptions and with values given in the table below. The fuel density is not more than 88 $^o/o$ th.d. (smeared density in both cases, pellet and powder packed fuel). The following Table XVII shows the moles of fission gas, released at an average rod power of 510 Watt/cm (maximum rod power 635 Watt/cm), 100 $^o/o$ fission gas release, 0.27 gas atoms per fission.

Table XVII Evaluation of Fission Gas Amount

	Pin Type I (S.S.-cladding)	Pin Type II (Inconel 625-cladding)
Fuel diameter, mm	5.55	5.55
Fuel length, mm	700	775
Fissionable atoms per cm^3	$2.056 \cdot 10^{22}$	$2.056 \cdot 10^{22}$
Fissionable atoms per pin	$3.48 \cdot 10^{23}$	$3.85 \cdot 10^{23}$
Total number of fissions in 225 day full power operation at average rod power	$2.14 \cdot 10^{22}$	$2.36 \cdot 10^{22}$
Fission gas amount n, moles	$0.96 \cdot 10^{-2}$	$1.06 \cdot 10^{-2}$

The gas volumes in the two pin types are:

Type I (S.S.-clad) = 8.91 cm^3

Type II (Inconel-clad) = 6.43 cm^3

The results are pressures of

77.5 kp/cm^2 with Type I and

119.0 kp/cm^2 with Type II.

That means stresses according to (9)

σ_p = 6.15 kp/mm^2 (Type I) and

σ_p = 9.45 kp/mm^2 (Type II).

The thermal stress σ_{th} is calculated with the formula

$$\sigma_{th} = \frac{E \cdot ß \cdot T_i}{2\,(1 - \nu)} \qquad (10)$$

E = Young's modulus (kp/mm^2)

ß = Thermal expansion coefficient ($^oC^{-1}$)

T_i = Temperature difference in the wall

ν = Poisson number

The values calculated are given in the following Tables XVIII and XIX, which include also the numerical details necessary for the stress conditions in 5.16. The results for pin Type I are plotted in Fig.15 and compared to the out-of-pile yield strength of the stainless steel.

Table XVIII Stress Evaluation for S.S.-clad pins at 130 $^{o}/o$ of power

z (cm)	Fuel height h (cm)	ΔT_i (^oC)	T_{av} (^oC)	E (kp/mm^2)	β $\times 10^{-6}$ (per ^oC)	ν	σ_{th} (kp/mm^2)	$1/4\ \sigma_{th}+\sigma_p$ (kp/mm^2)
35	70	48	705	15000	18		9,5	8,6
30	65	56	690	15200	18		11,3	9,0
20	55	70	640	16000	17,9		14,7	9,9
10	45	79	580	16500	17,7	0,32	17,0	10,5
0	35	90	510	17000	17,5		19,7	11,1
-10	25	85	430	18000	17,4		19,5	11,1
-20	15	74	360	18500	17,3		17,4	10,6
-30	5	61	300	19000	17,1		14,6	9,9
-35	0	52	275	19500	17,0		12,7	9,4

Table XIX Stress Evaluation +) for Inconel-clad pins at 130 $^{o}/o$ of power

z (cm)	Fuel height h (cm)	ΔT_i (^oC)	T_{av} (^oC)	E (kp/mm^2)	β $\times 10^{-6}$ (per ^oC)	ν	σ_{th} (kp/mm^2)	$1/4\ \sigma_{th}+\sigma_p$ (kp/mm^2)
-10	28.7	102	463	18500	14.3	0.32	19.9	14.4

+) It is sufficient to give the highest value only, because σ_y is nearly constant with temperature.

Appendix A 4

Creep Strain in Cladding

In order to calculate the expected secondary creep strain in the cladding, the creep law

$$\dot{\varepsilon} = k\,\sigma^n \tag{11}$$

is assumed applicable, where

$\dot{\varepsilon} \,\hat{=}\,$ creep rate, ($^o/o$/hr)
$k \,\hat{=}\,$ creep constant, $^o/o$/hr (kp/mm^2)^2
$\sigma \,\hat{=}\,$ stress, (kp/mm^2)^n
$n \,\hat{=}\,$ creep exponent

Furthermore it is assumed the 100$^o/o$ gas release accompanies burnup of the fuel, both occurring at a constant rate. Thus, the stress in the cladding at any time, t, based on a thin-wall treatment is

$$\sigma = \dot{p}\left(\frac{d}{2s}\right)t \tag{12}$$

where $\dot{p} \,\hat{=}\,$ rate of pressure buildup in cladding, kp/mm^2 per hr
$d \,\hat{=}\,$ cladding diameter
$s \,\hat{=}\,$ cladding thickness

To obtain the integral strain at any time, t, it is taken

$$\varepsilon = \int_o^t \dot{\varepsilon}\,dt = \int_o^t k\left[\dot{p}\left(\frac{d}{2s}\right)t\right]^n dt$$

with the result:

$$\varepsilon = \frac{k}{n+1}\left[\dot{p}\left(\frac{d}{2s}\right)\right]^n t^{n+1} \tag{13}$$

In order to calculate the creep constants and exponents to be used in this equation, published values of the stress required to produce a given strain in a specific time for both the stainless steel +) and Inconel 625 ++) were converted to creep rates of $^{o}/o$/hr and were plotted as shown in the figure 16. Since the strain values include primary as well as secondary creep, the actual secondary creep strain should be slightly lower than the total. The creep data for the stainless steel and for Inconel 625 are for temperatures of 600 and 650 ^{o}C, respectively.

Four sets of data were available for the stainless steel. When plotted, the calculated n values for the 3 lines shown in the figure were 8 $\pm$ 0.2. Three separate k values for the three lines also were calculated. While the intermediate value is regarded as the most probable appropriate value for annealed material, the lowest value represents a conservative choice. Thus, secondary creep can be represented on a conservative basis for this alloy as

$$\dot{\varepsilon} = 8.06 \cdot 10^{-14} \cdot \sigma^{8.2}$$

Three sets of data were available for the Inconel alloy as plotted in the figure. While the upper curve is considered to permit the best fit of the experimental points, the lower line, emphasizing a few lower points, is selected for purposes of calculation as being a conservative choice, giving

$$\dot{\varepsilon} = 1.824 \cdot 10^{-6} \cdot \sigma^{7.3}$$

For purposes of calculation, corrosion penetration during the time of irradiation is assumed to be 50 microns and again, in order to provide the most conservative estimate, the final cladding dimensions rather than the original are used in calculating the expected cladding strain. Therefore,

$$d_{O.D.} = 6.25 \text{ mm}$$

$$d_{I.D.} = 5.55 \text{ mm}$$

+) Landolt-Börnstein, "Zahlenwerte und Funktionen", Sechste Auflage, Springer-Verlag (1963), pg. 579

++) Huntington Alloy Products Division Circular, "Preliminary Data, Inconel Alloy 625".

$$s = 0.35$$

$$\frac{d_{av}}{2s} = 8.4$$

Also,

$$t = 225 \text{ day} = 5.4 \cdot 10^3 \text{ hr}$$

and for stainless steel pins

$$p_{final} = 78 \text{ kp/cm}^2$$

$$\dot{p} = 1.44 \cdot 10^{-4} \frac{\text{kp}}{\text{mm}^2\text{-hr}}$$

while for Incoloy 625 pins

$$p_{final} = 119 \text{ kp/cm}^2$$

$$\dot{p} = 2.20 \cdot 10^{-4} \frac{\text{kp}}{\text{mm}^2\text{-hr}}$$

Substituting these values in equation (13) we obtain for

S.S. $\underline{\underline{\varepsilon = 3.1 \cdot 10^{-4} \text{ \%}}}$

and for

Incoloy 625 $\underline{\underline{\varepsilon = 3.1 \cdot 10^{-4} \text{ \%}}}$

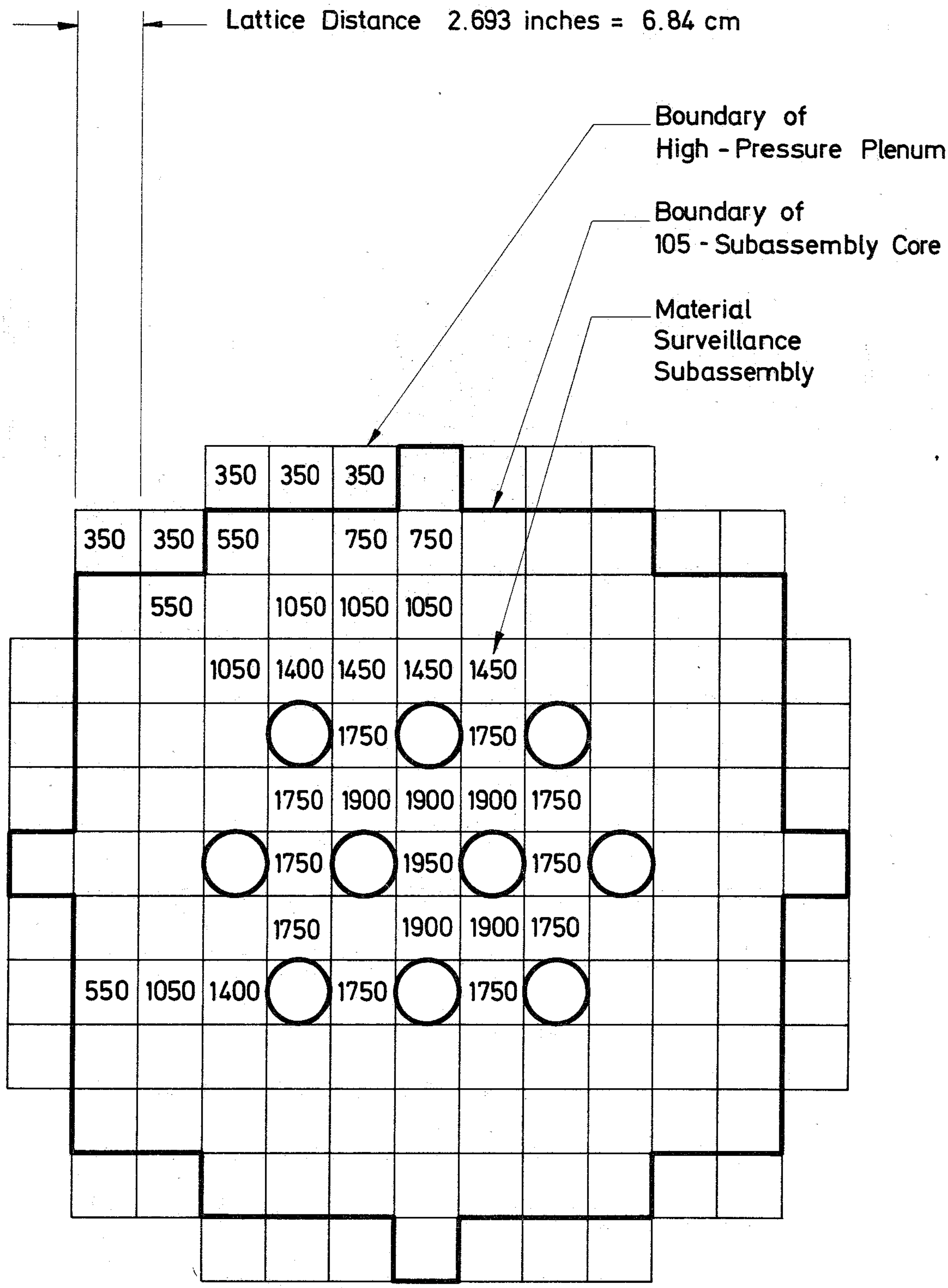


Fig.1 IRRADIATION POSITIONS IN EFFBR

Category Numbers Mean Irradiation Charges in $ per Full Power Operating Day at 110 Megawatt

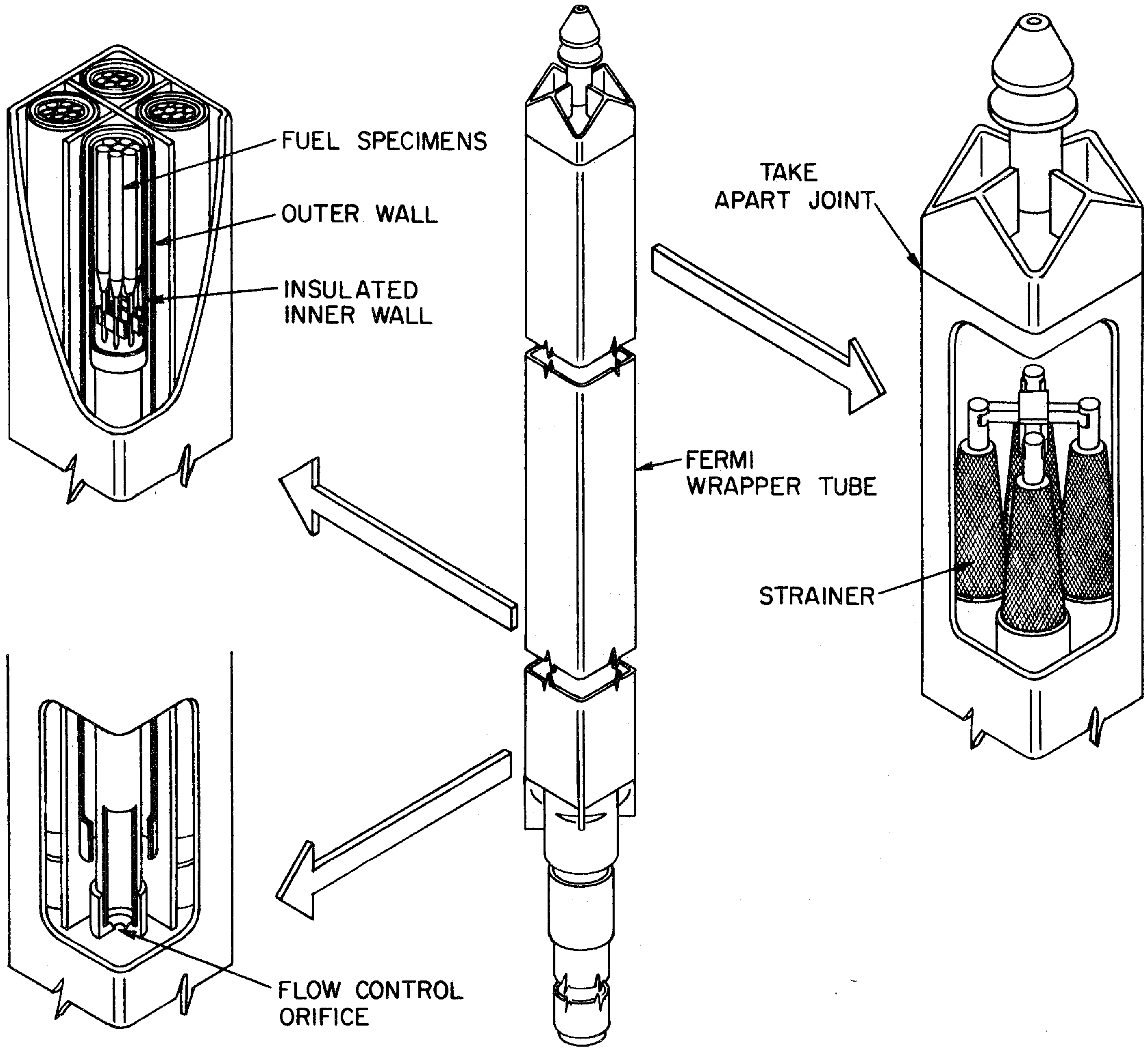


Fig. 2 **STANDARD FUEL TEST SUBASSEMBLY**

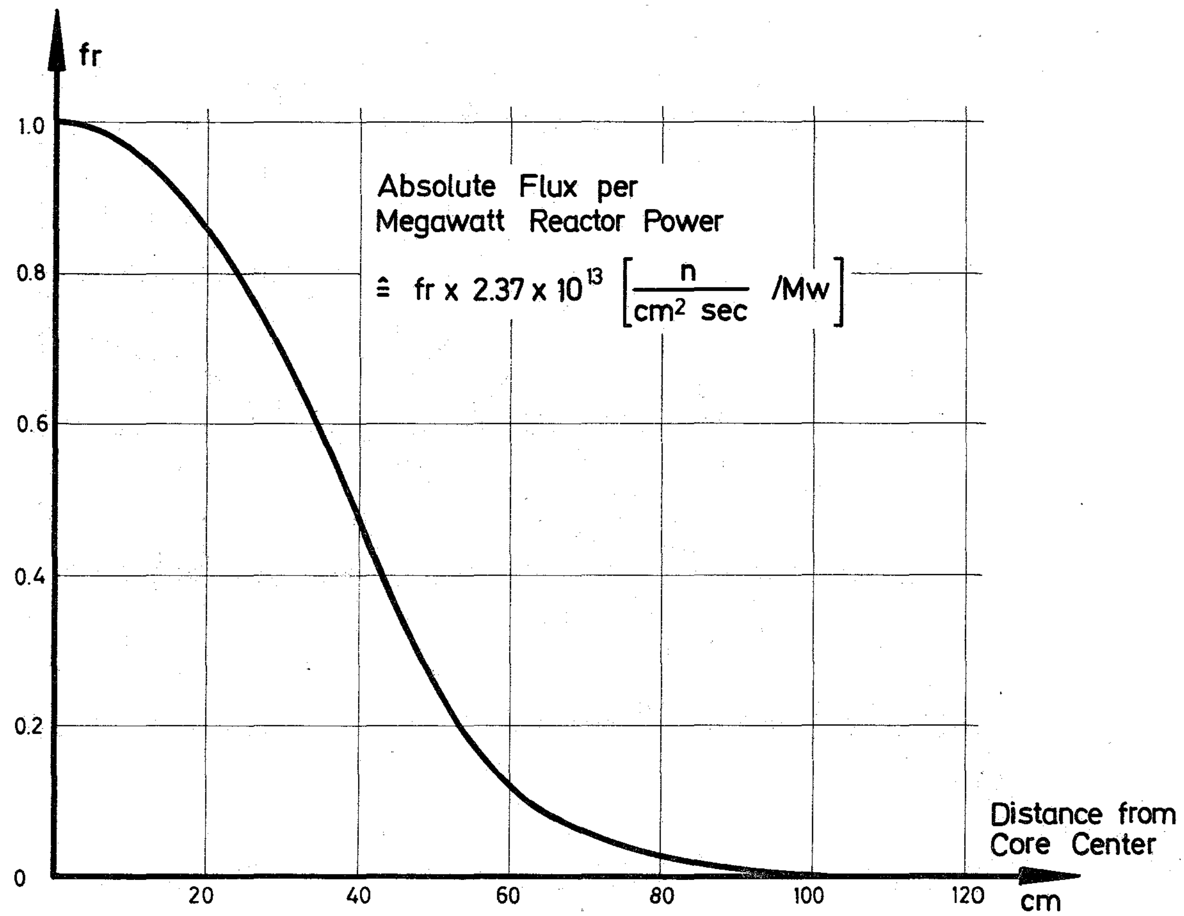


Fig. 3 RADIAL FLUX DISTRIBUTION

Total Flux of Central Plane

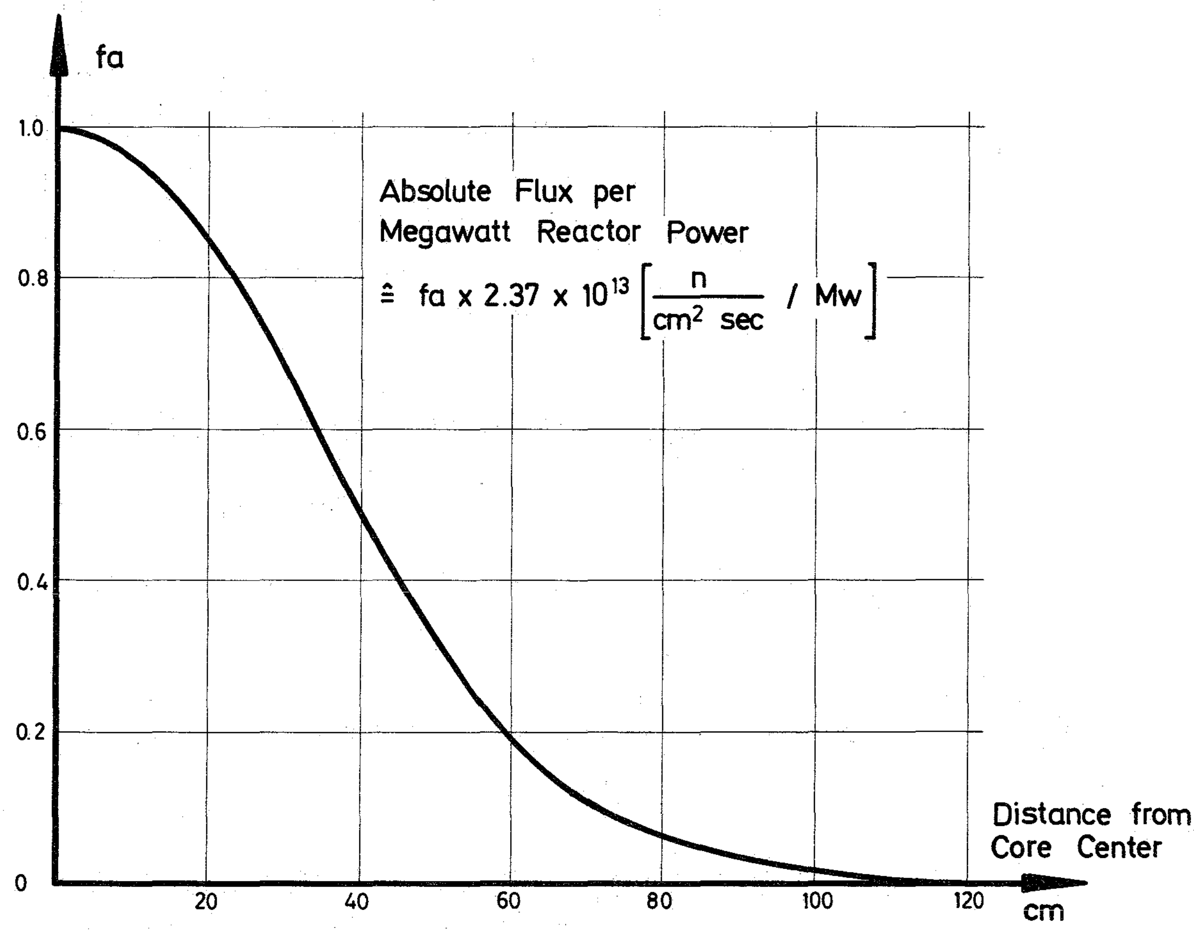


Fig. 4 AXIAL FLUX DISTRIBUTION

Total Flux along Core Centerline

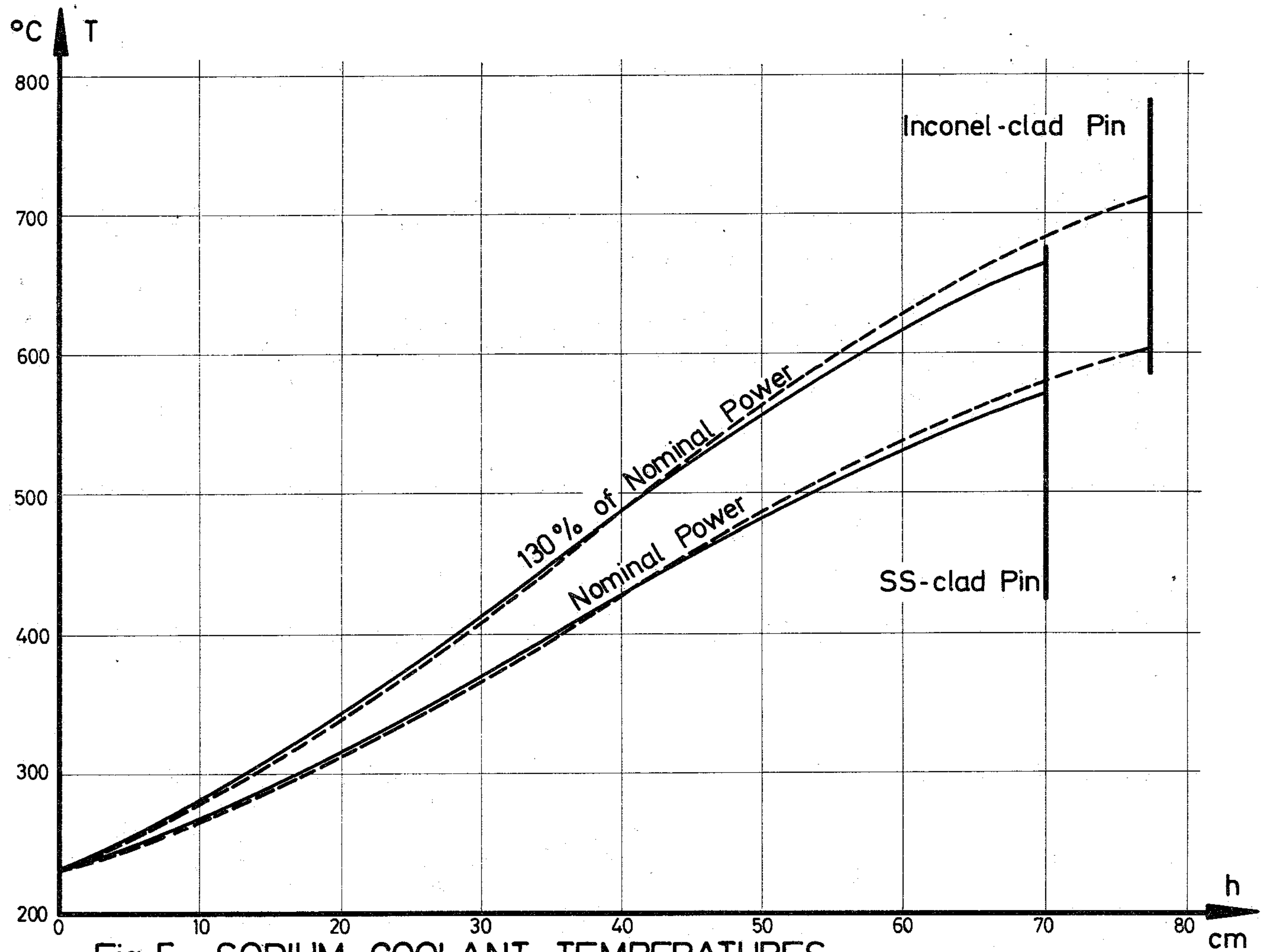


Fig. 5 SODIUM COOLANT TEMPERATURES

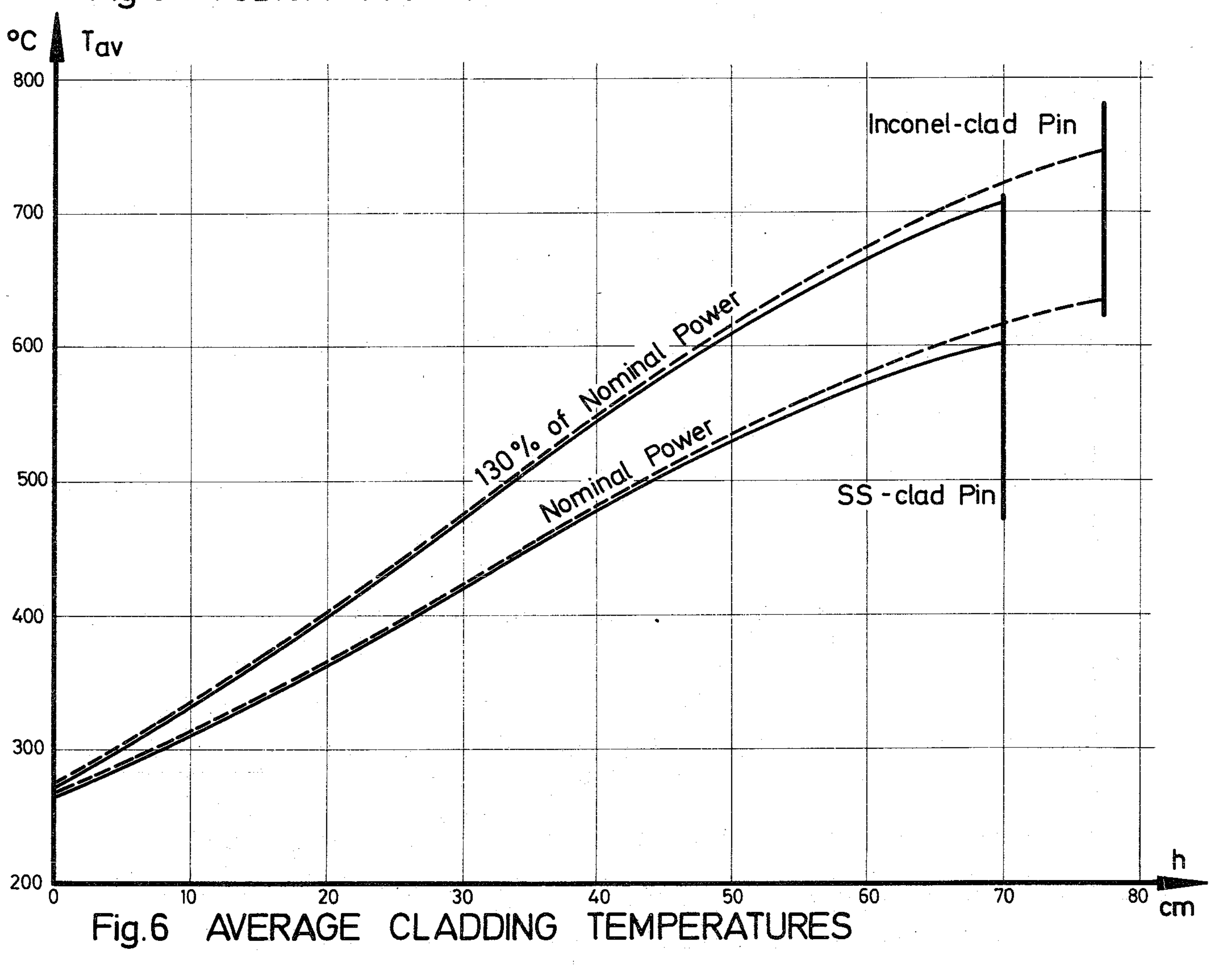


Fig. 6 AVERAGE CLADDING TEMPERATURES

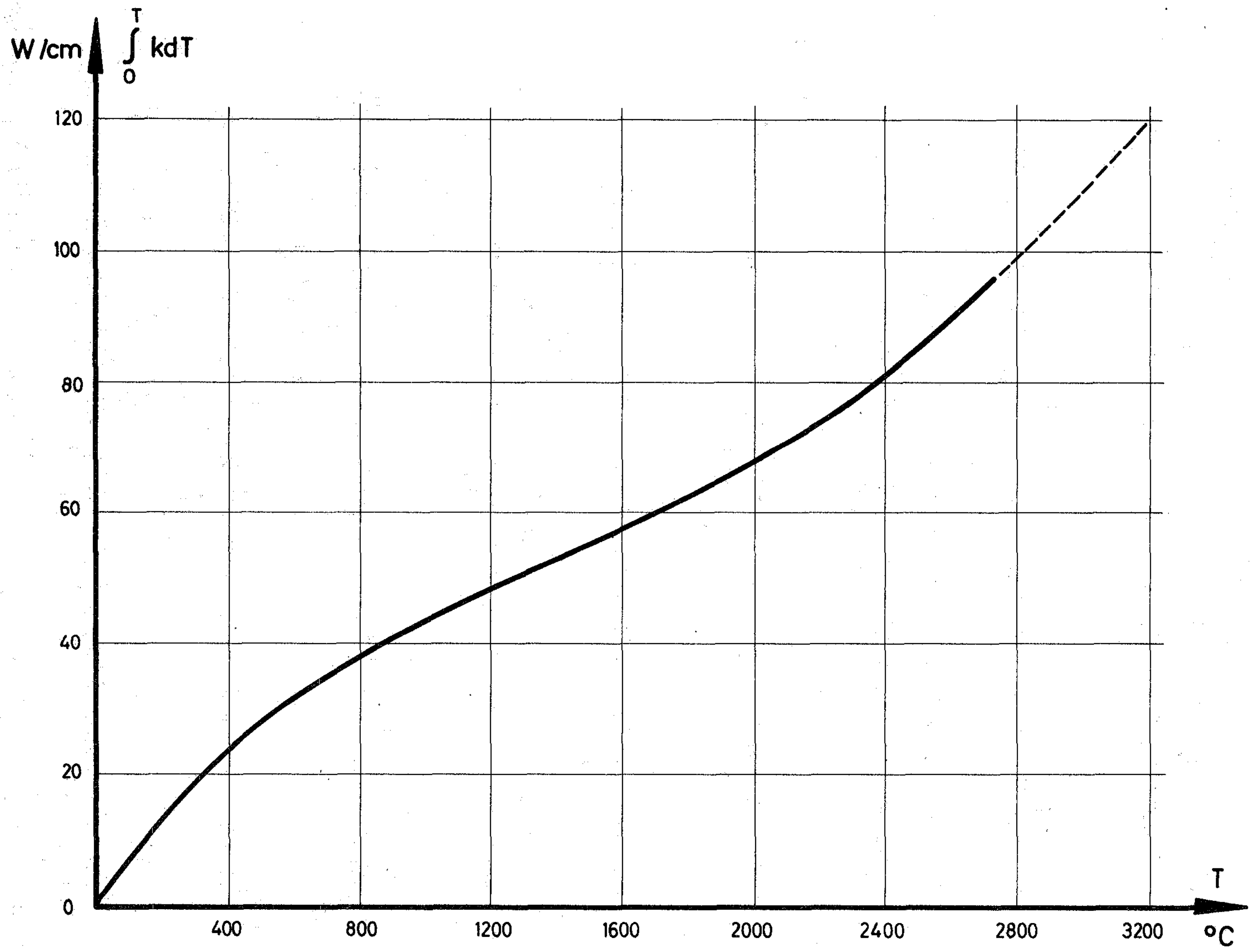


Fig. 7 INTEGRAL OF HEAT CONDUCTIVITY

UO_2 - PuO_2 with 15 w/o PuO_2 , 90% of th. d.

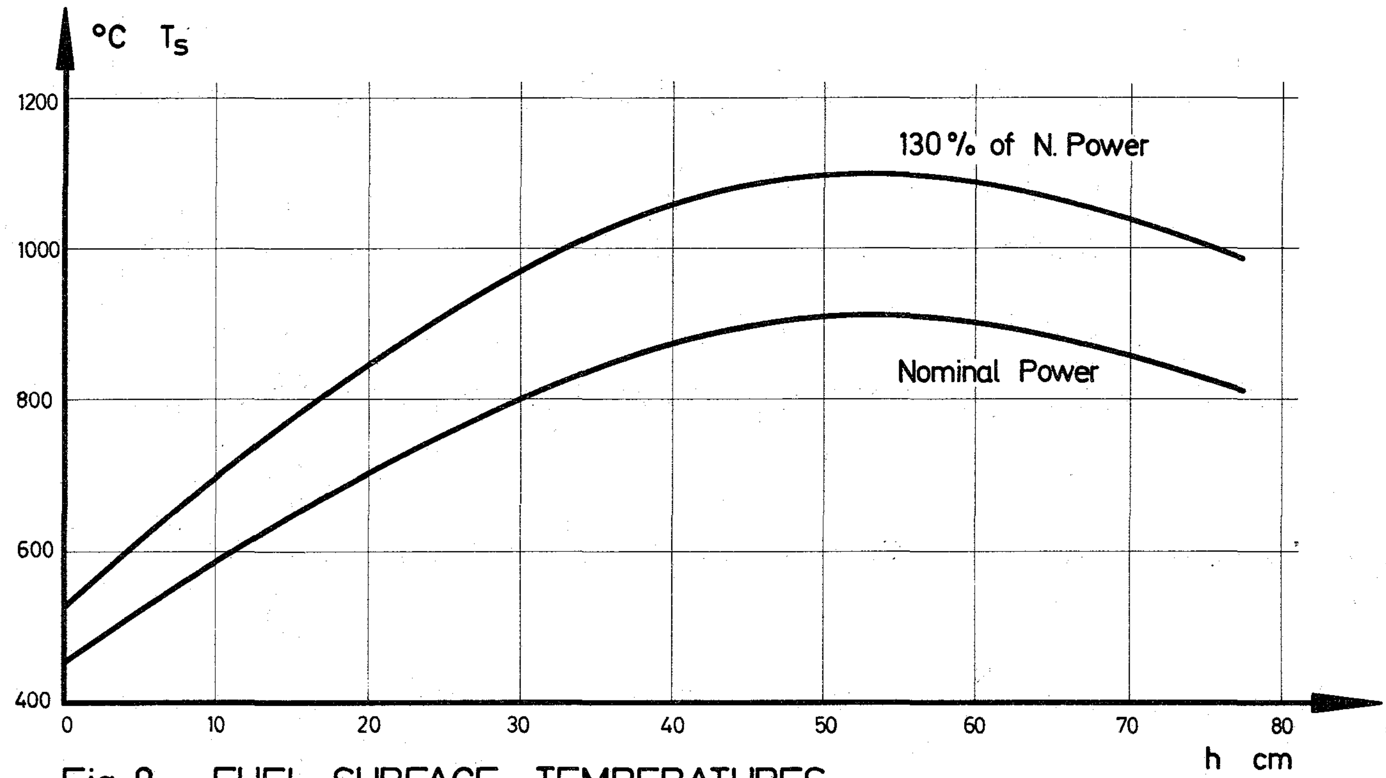


Fig. 8 FUEL SURFACE TEMPERATURES
Inconel - clad Pin

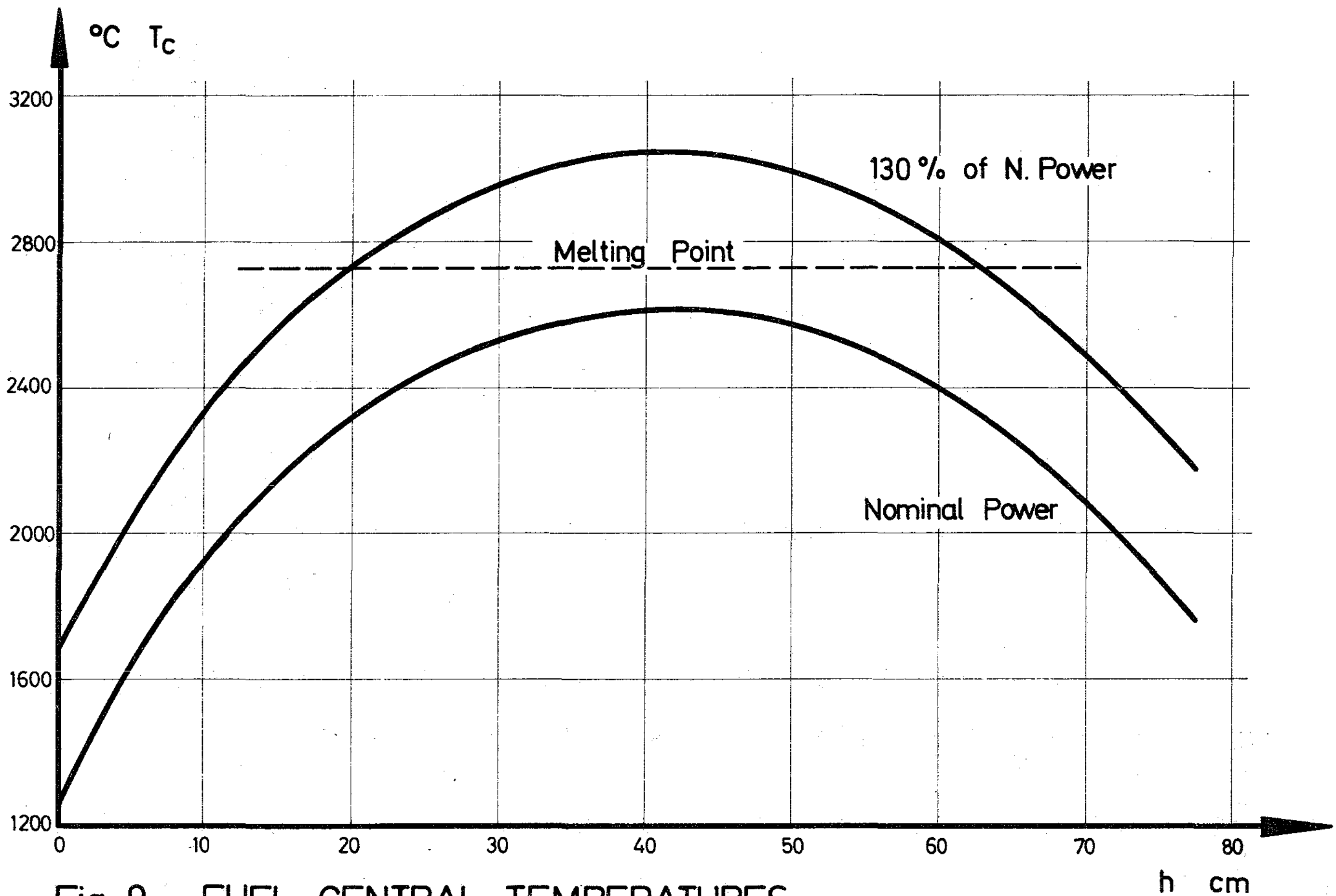


Fig. 9 FUEL CENTRAL TEMPERATURES
Inconel - clad Pin

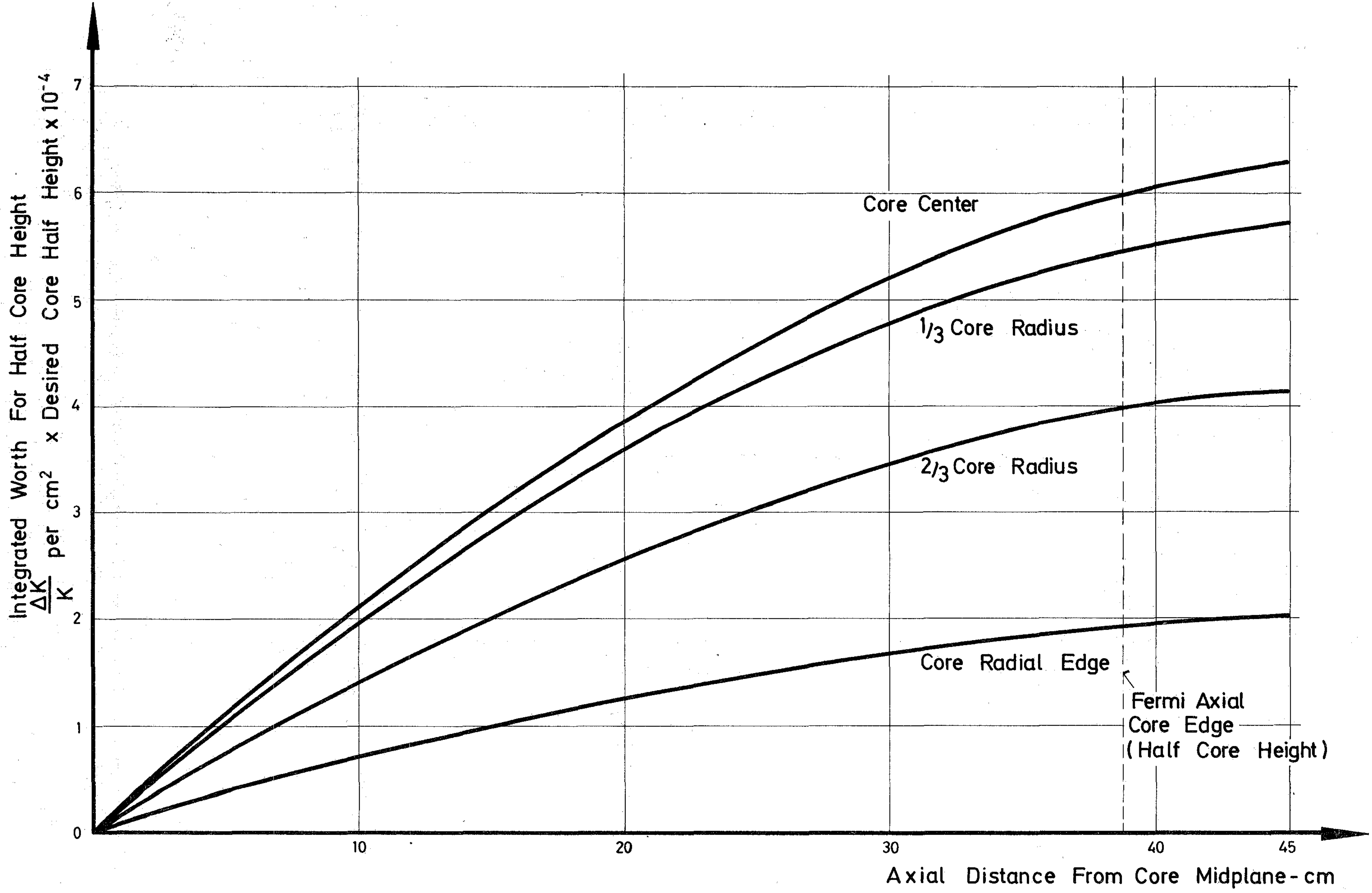


Fig.10 REACTIVITY VALUES FOR $U^{235}O_2$ (100% T. D.)

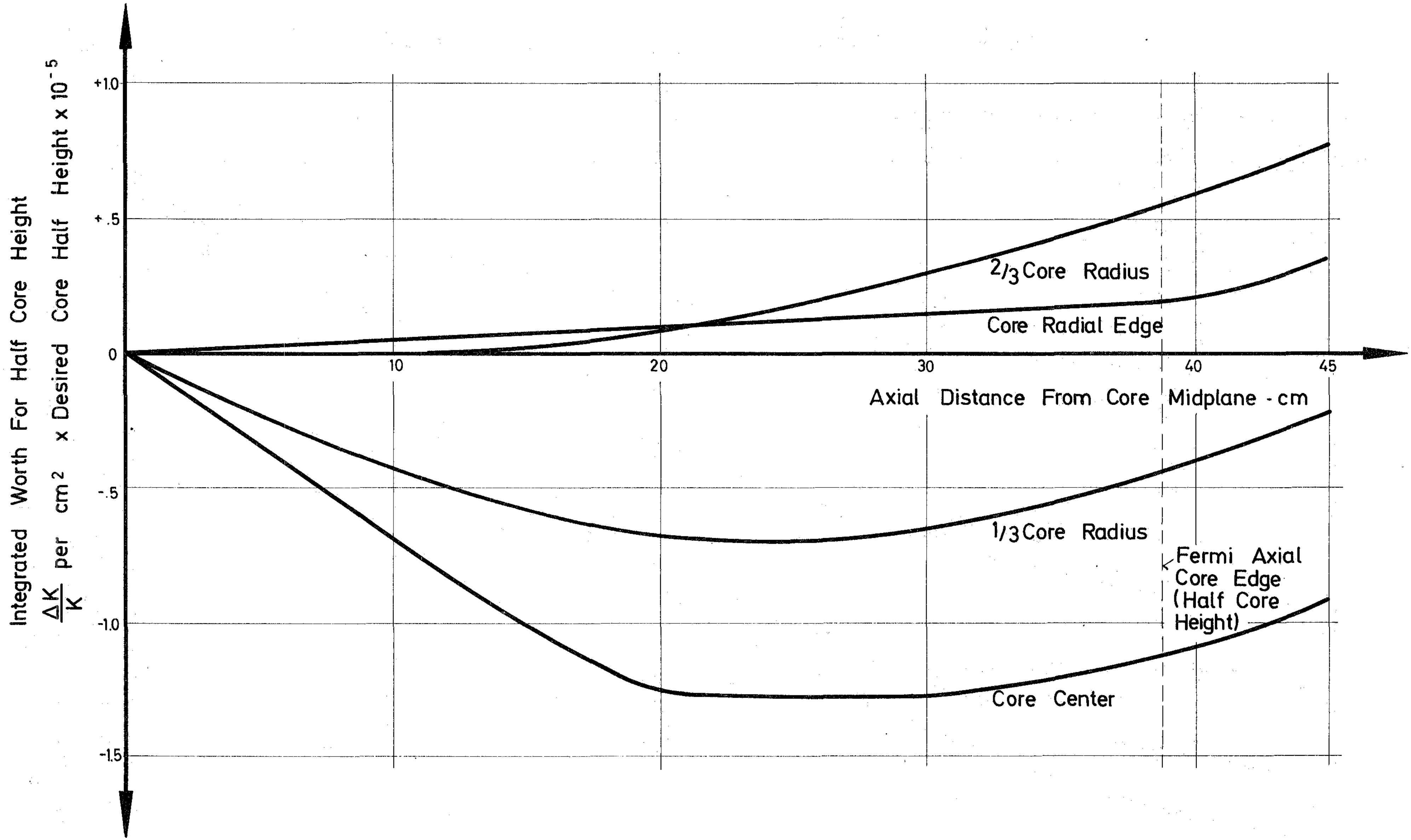


Fig. 11 REACTIVITY VALUES FOR $U^{238}O_2$ (100 % T. D.)

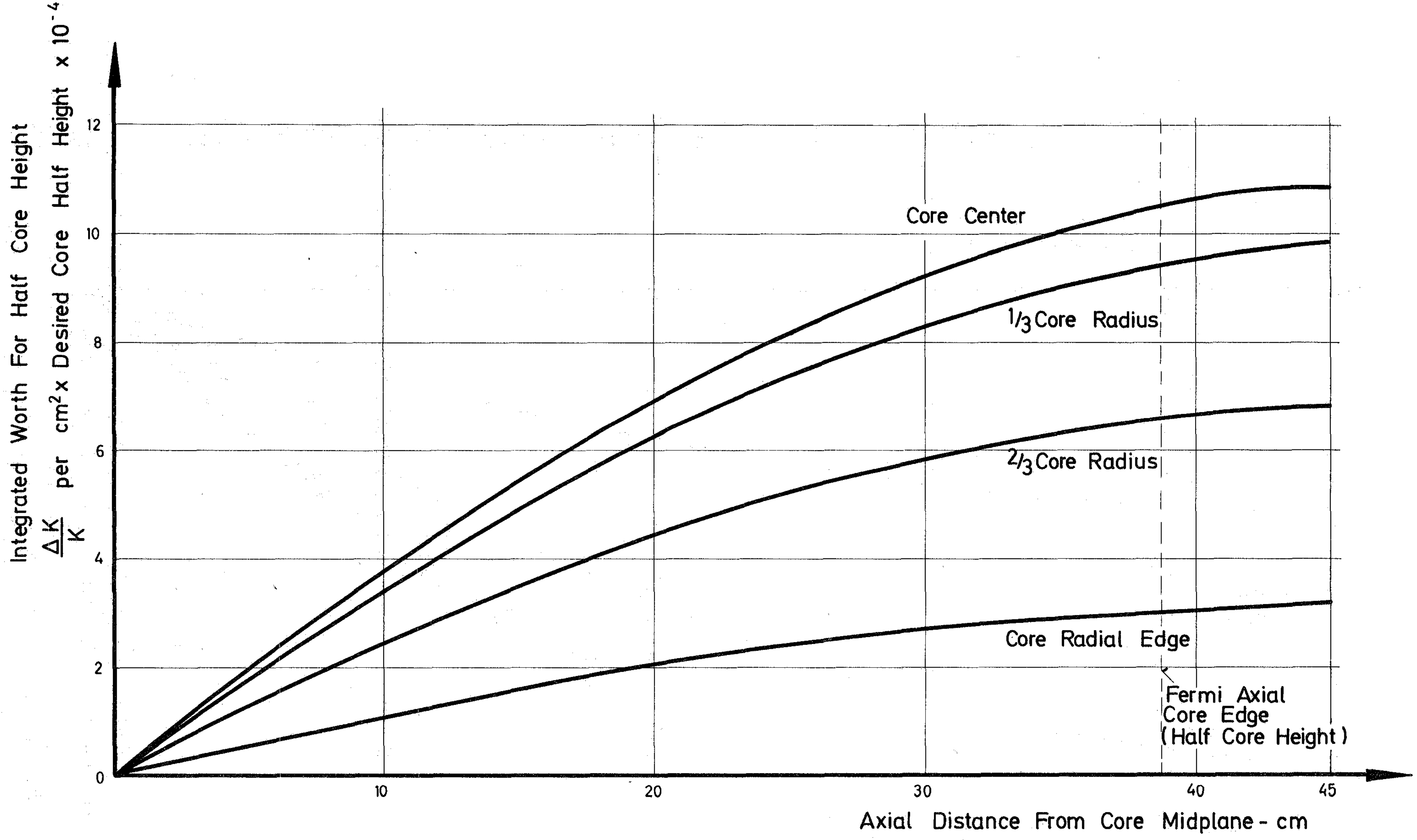


Fig. 12 REACTIVITY VALUES FOR $Pu^{239}O_2$ (100 % T.D.)

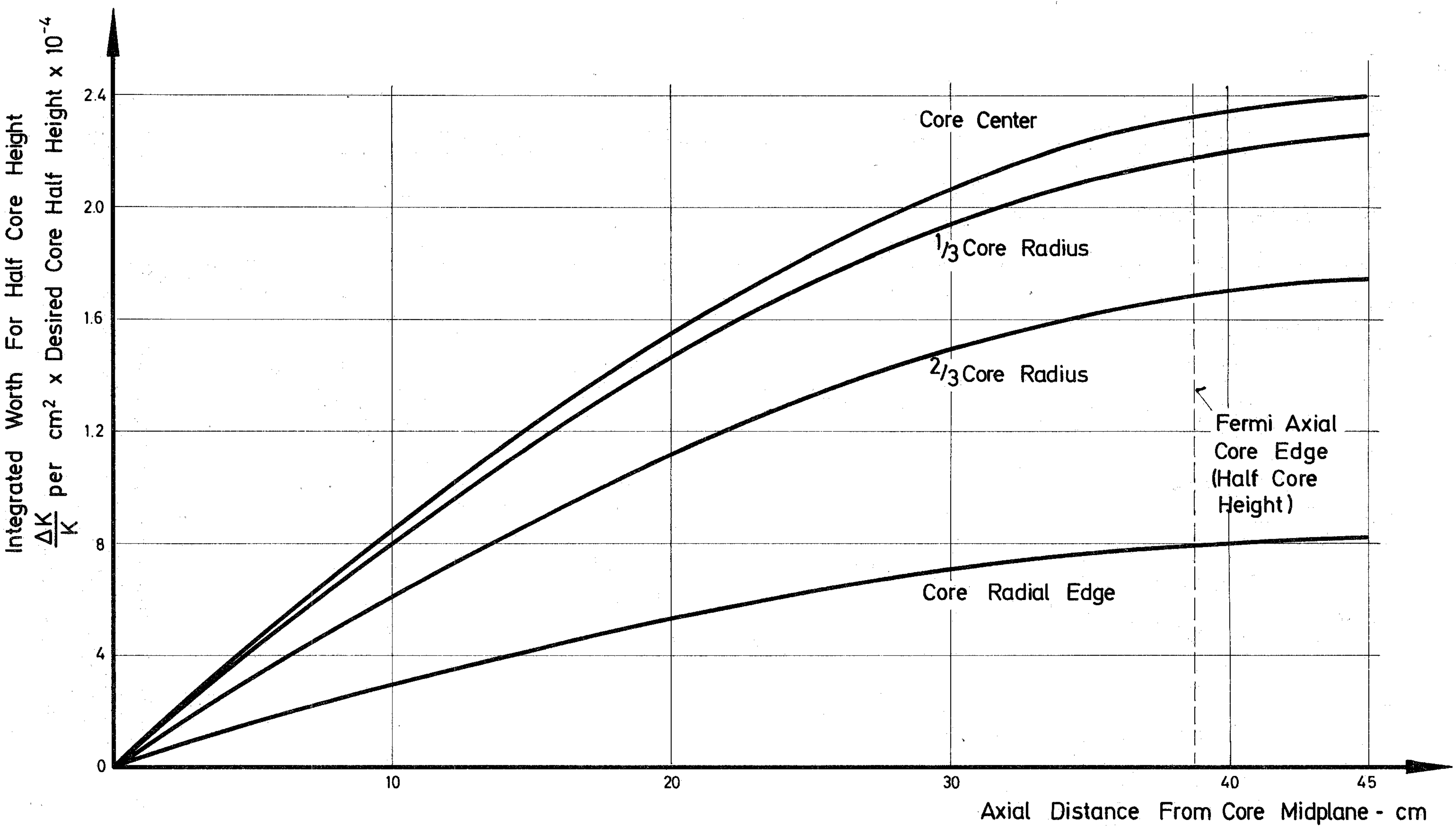


Fig. 13 REACTIVITY VALUES FOR $Pu^{240}O_2$ (100% T.D.)

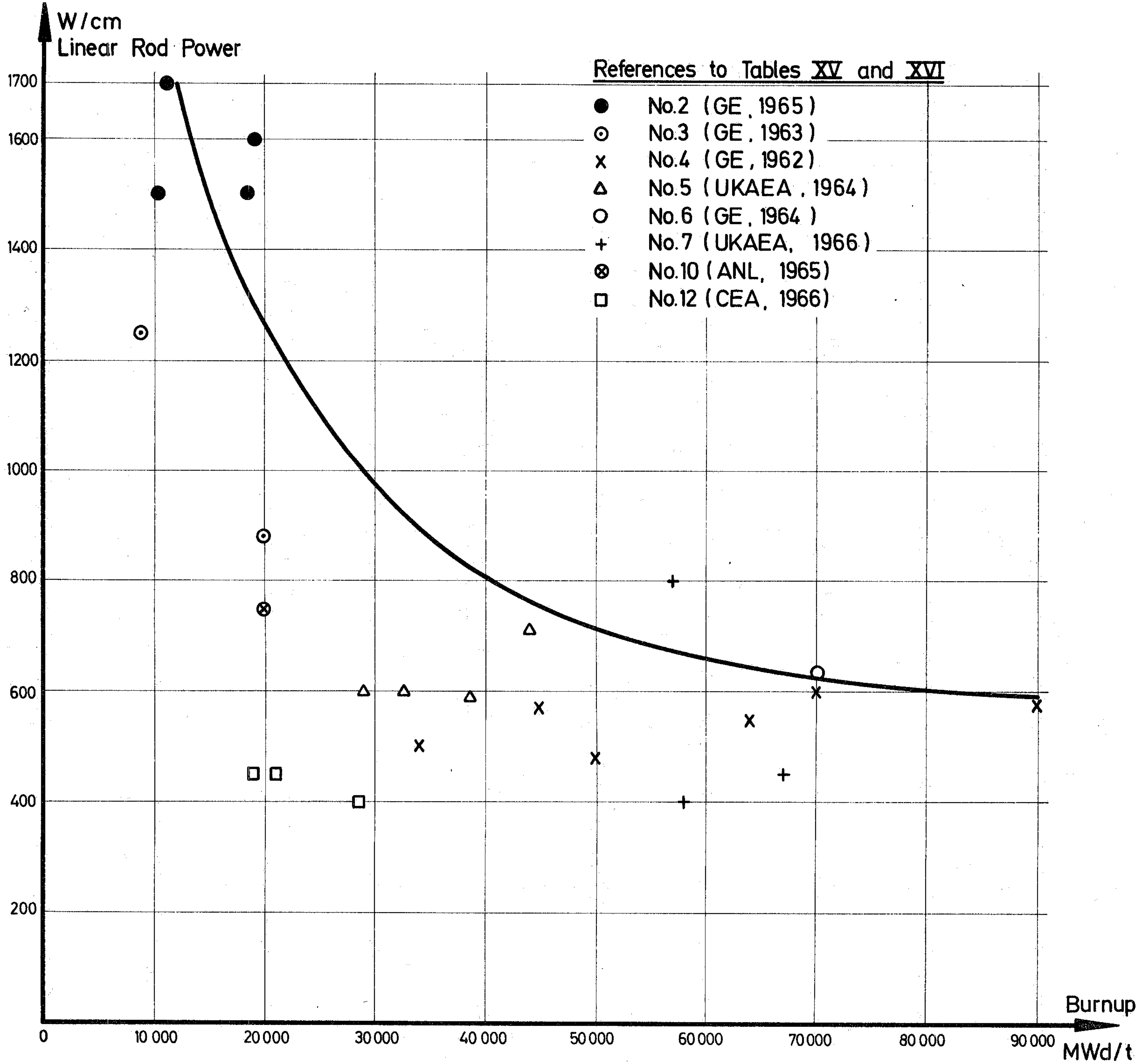


Fig. 14 IRRADIATION EXPERIENCE DATA

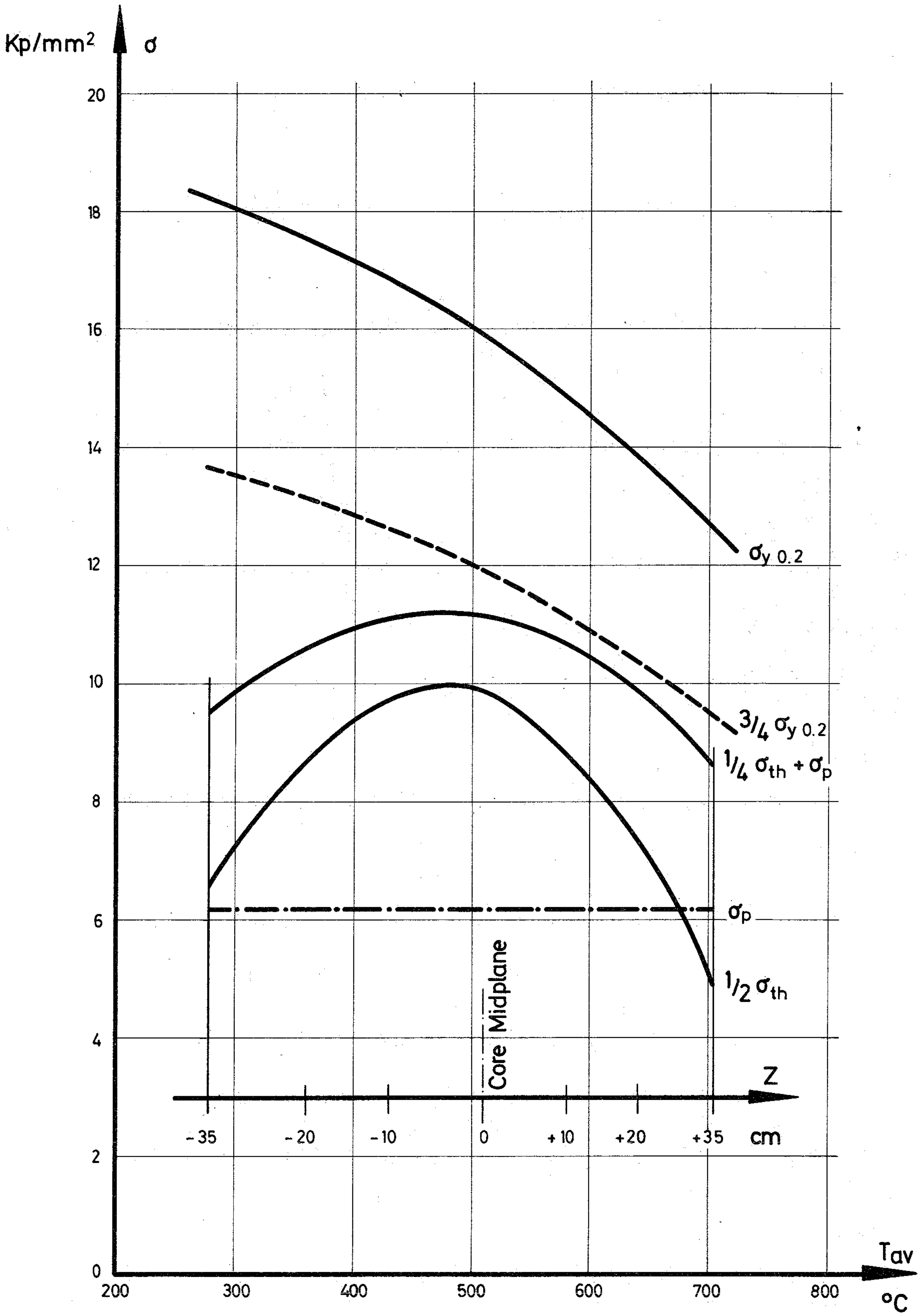


Fig. 15 STRESS CONDITIONS IN THE FUEL PIN

For SS-clad pins at 130 % of nominal power

Conditions: $\frac{1}{4}\ \sigma_{th} + \sigma_p < \sigma_{y\,0.2}$

$\frac{1}{2}\ \sigma_{th} < \sigma_{y\,0.2}$

$\sigma_p < \frac{3}{4}\ \sigma_{y\,0.2}$

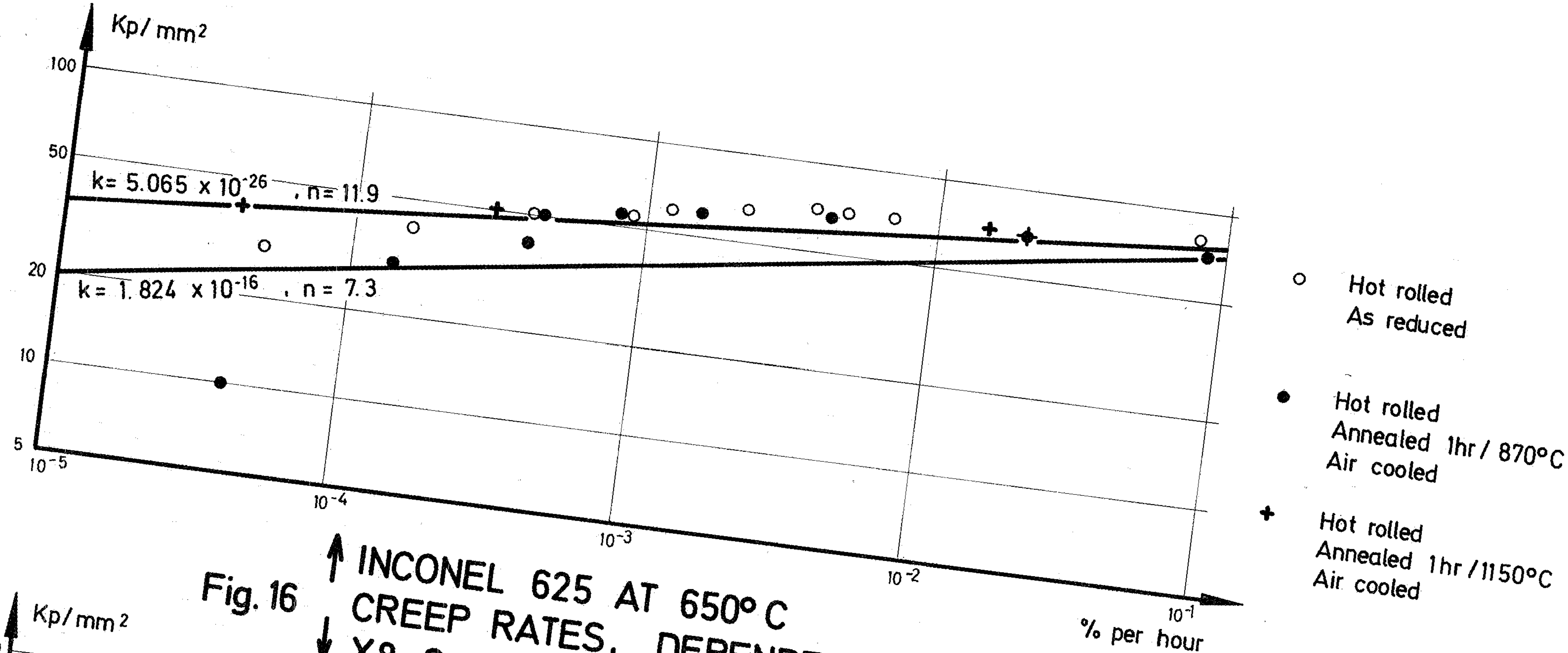
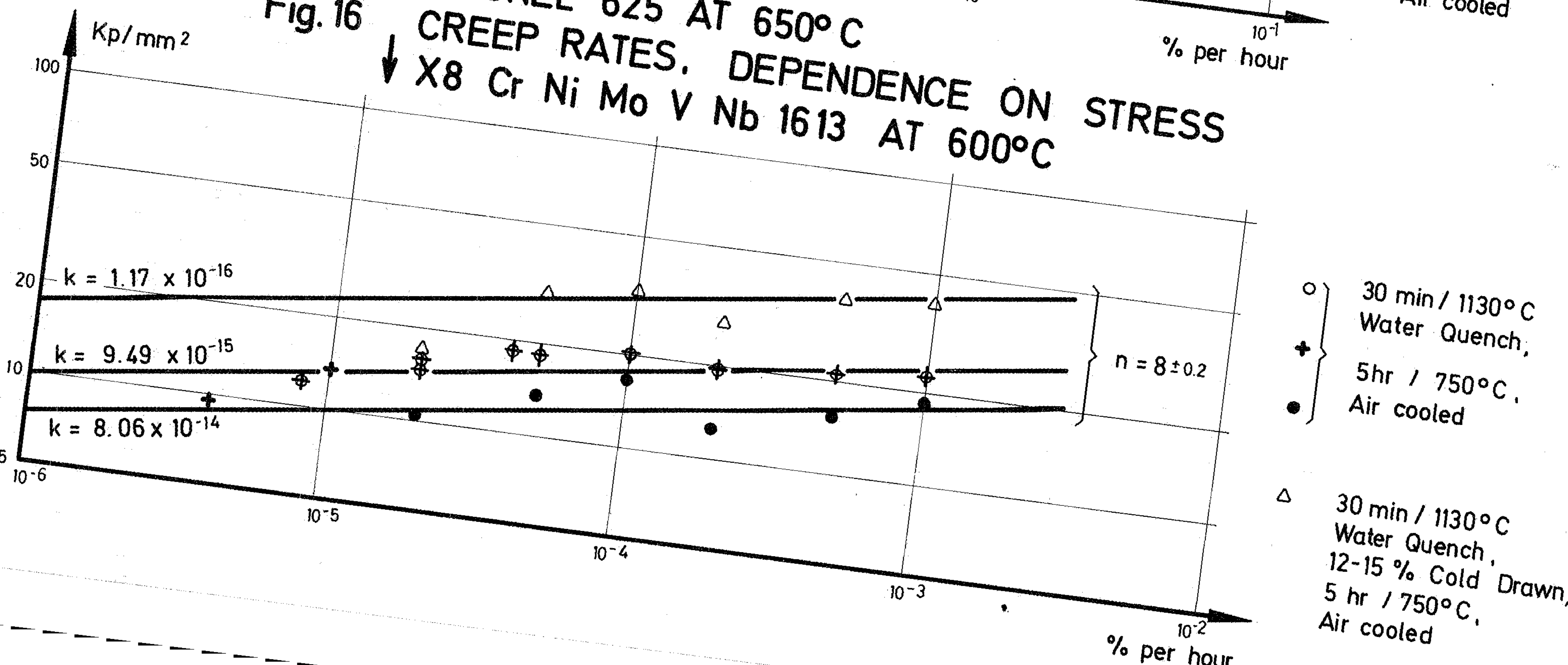


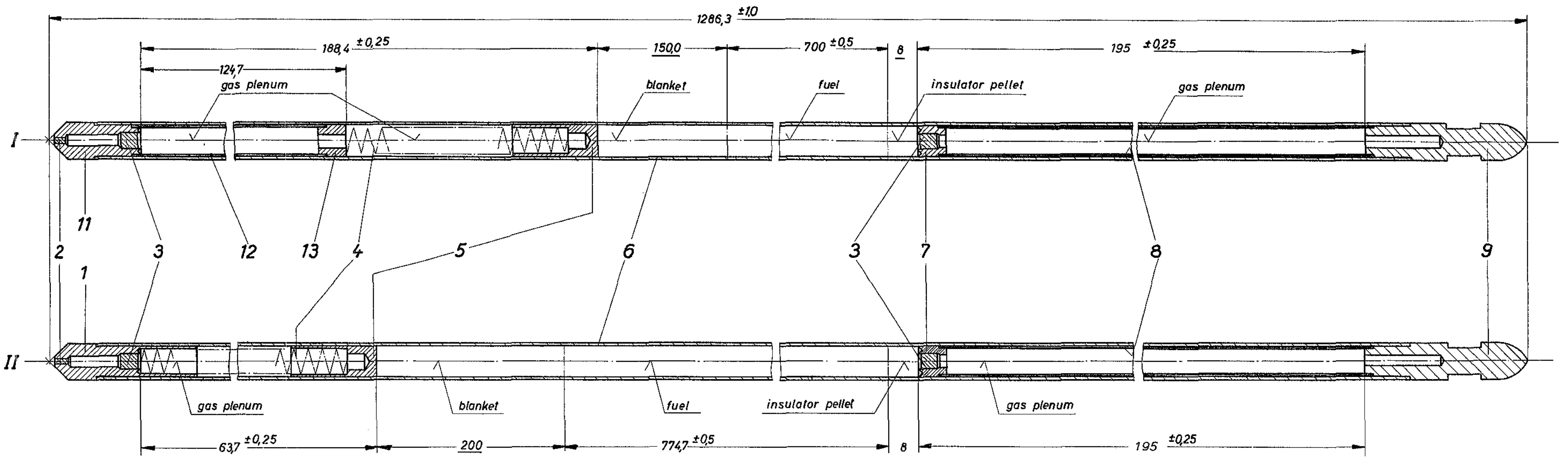
Fig. 16

↑ INCONEL 625 AT 650° C

CREEP RATES, DEPENDENCE ON STRESS

↓ X8 Cr Ni Mo V Nb 1613 AT 600°C

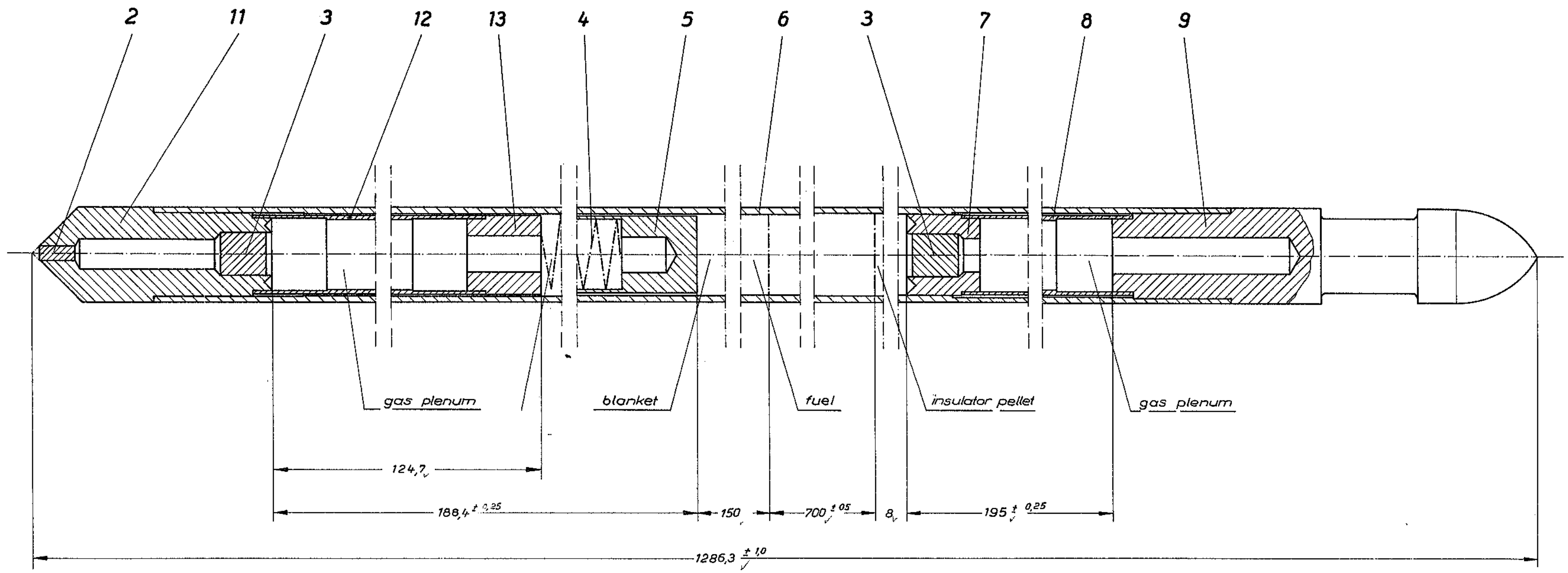




Teil	Stück	Benennung		Werkstoff Typ I	Werkstoff Typ II	Zchng. Nr.	Bemerkung
13		Stopfen	plug	4988		TA2K-16-07-4-1703	
12		Oberer Einsatz	upper inset	4988		TA2K-16-07-4-1703	
11		Oberer Endstopfen	upper end cap	4988		TA2K-16-07-4-1702	
10							
9		Unterer Endstopfen	lower end cap	4988	Inconel 625	TA2K-16-07-3-1688	
8		Unterer Einsatz	lower inset	4988	Inconel 625	TA2K-16-07-4-1687	
7		Stopfen	plug	4988	Inconel 625	TA2K-16-07-4-1687	
6		Hüllrohr	cladding	4988	Inconel 625	TA2K-16-07-4-1686	
5		Führungsstück	sliding inset	4988	Inconel 625	TA2K-16-07-4-1685	
4		Feder	spring	Cr Ni Stahl	Cr Ni Stahl	o.Z.	
3		Sinterstahlfilter	S.S. filter	Cr Ni Stahl	Cr Ni Stahl	TA2K-16-07-4-1683	
2		Stift	pin	4988	Inconel 625	TA2K-16-07-4-1684	
1		Oberer Endstopfen	upper end cap		Inconel 625	TA2K-16-07-4-1683	

Irradiation Test-Pins
for Enrico Fermi-Reactor

TA 2K-3-07-02-1755



Teil	Stück	Benennung	Werkstoff	Abmessung	Zeichngs. Nr. Norm	Bemerkung
13	1	Stopfen	4988	6ø x 5	TA2K-16-07-4-1703	
12	1	Oberer Einsatz	4988	5,4ø x 4,9ø x 122,7	TA2K-16-07-4-1703	
11	1	Oberer Endstopfen	4988	6,35ø x 16,1	TA2K-16-07-4-1702	
10						
9	1	Unterer Endstopfen	4988	5,6ø x 28,8	TA2K-16-07-3-1688	
8	1	Unterer Einsatz	4988	5,4ø x 5ø x 193	TA2K-16-07-4-1687	
7	1	Stopfen	4988	6ø x 5	TA2K-16-07-4-1687	
6	1	Hüllrohr	4988	6,35ø x 5,55ø x 1257	TA2K-16-07-4-1686	
5	1	Führungsstück	4988	5,4ø x 15	TA2K-16-07-4-1685	
4	1	Feder	Cr Ni Stahl		o.Z.	
3	2	Sinterstahlfilter	Cr Ni Stahl	2,8ø x 3	TA2K-16-07-4-1683	
2	1	Stift	4988	1ø x 3	TA2K-16-07-4-1684	
1						

Oberflächenzeichen	∼	▽	▽▽	▽▽▽	▽▽▽▽
Rauhtiefe max. in µ	1000	40	10	4	1,6

Freimaßtoleranz	bis 6	über 6 bis 30	über 30 bis 100	über 100 bis 300	über 300 bis 1000	über 1000 bis 2000
	± 0,1	± 0,2	± 0,3	± 0,5	± 0,8	± 1,2

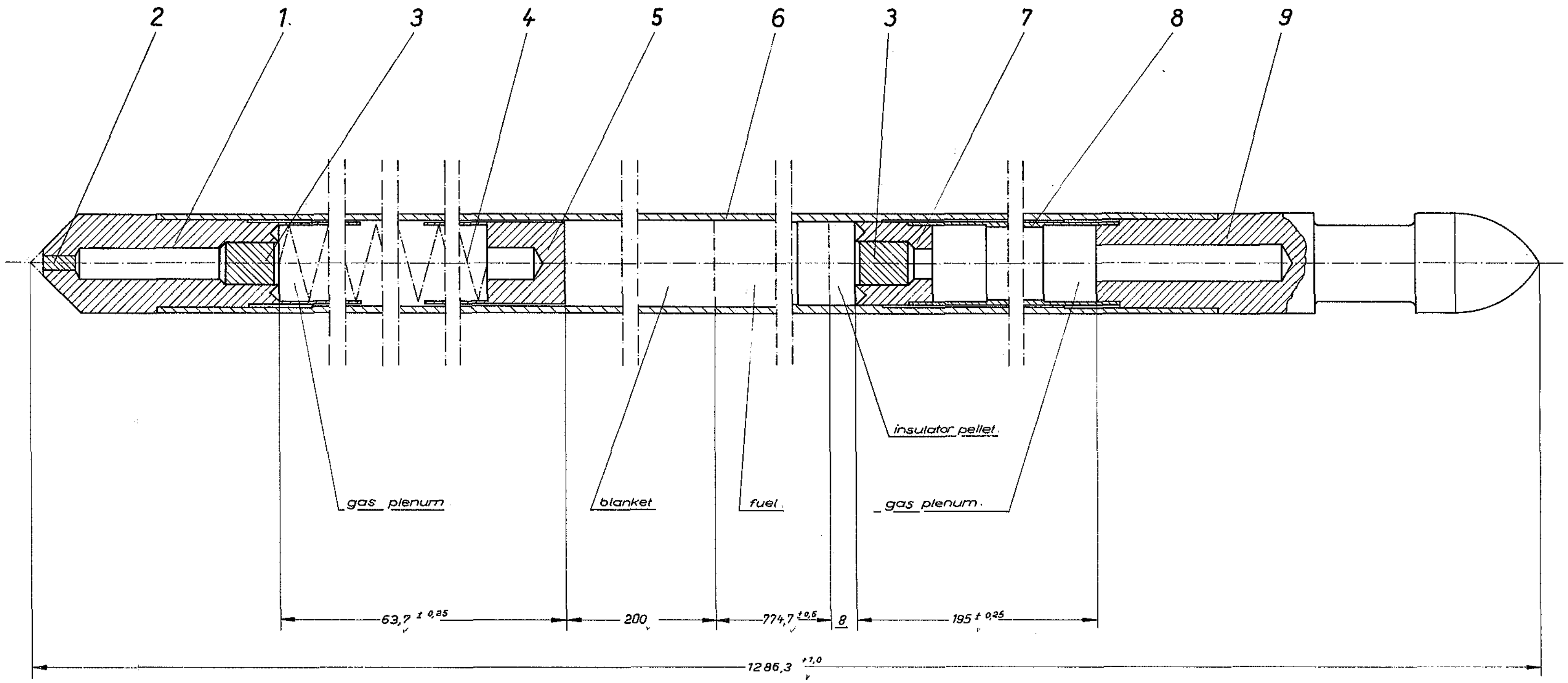
1966	Tag	Name	Werkstoff
gez.	27. 5.	Ra	
gepr.			
ges.			

Gesellschaft für Kernforschung m. b. H.
7500 Karlsruhe
Postfach 947

Zugeh. Zchng.
Ersatz für
Ersetzt durch

Benennung: Brennelement-Prüfst. Typ I
für Enrico Fermi - Reaktor

Zeichnungs Nr.
TA 2K-16-07-2-1701



Teil	Stück	Benennung	Werkstoff	Abmessung	Zeichngs. Nr. Norm	Bemerkung
9	1	Unterer Endstopfen	Inconel 625	5,6ϕ x 28,8	TA2K-16-07-3-1688	
8	1	Unterer Einsatz	Inconel 625	5,4 ϕ x 5 ϕ x 193	TA2K-16-07-4-1687	
7	1	Stopfen	Inconel 625	6 ϕ x 5	TA2K-16-07-4-1687	
6	1	Hüllrohr	Inconel 625	6,35ϕ x 5,55ϕ x 1257,0	TA2K-16-07-4-1686	
5	1	Führungsstück	Inconel 625	5,4 ϕ x 15	TA2K-16-07-4-1685	
4	1	Feder	Cr Ni Stahl		o. Z.	
3	2	Sinterstahlfilter	Cr Ni Stahl	2,8 ϕ x 3	TA2K-16-07-4-1683	
2	1	Stift	Inconel 625	1ϕ x 3	TA 2K-16-07-4-1684	
1	1	Oberer Endstopfen	Inconel 625	6,35 ϕ x 26,1	TA 2K-16-07-4-1683	

Oberflächenzeichen	~	▽	▽▽	▽▽▽	▽▽▽▽
Rauhtiefe max. in μ	1000	40	10	4	1,6

Freimaßtoleranz	bis 6	über 6 bis 30	über 30 bis 100	über 100 bis 300	über 300 bis 1000	über 1000 bis 2000
	± 0,1	± 0,2	± 0,3	± 0,5	± 0,8	± 1,2

1966	Tag	Name
gez.	20. 5.	Ko
gepr.		
ges.		

Werkstoff

Gesellschaft für Kernforschung m. b. H.
7500 Karlsruhe
Postfach 947

Zugeh. Zchng.

Ersatz für TA2K-16-07-2-1453

Ersetzt durch

Benennung **Brennelement-Prüfst. Typ II**
für Enrico Fermi-Reaktor

Zeichnungs Nr.
TA 2K-16-07-2-1682