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(Limited)

Proposal of an Irradiation of ATR UO_2-PuO_2 Fuel Test Assembly (IFA-423) in the HBWR

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Contents

1. Introduction

We have an irradiation program with one uranium-plutonium mixed oxide test fuel assembly (IFA-423) in HBWR. The test fuel assembly is composed of 7 fuel rods whose diameter is the same as the ATR fuels, ring-spring type spacers, and tie rods.

The fuel rods and other parts of the assembly will be sent to Halden in July of 1974. The irradiation testing is scheduled to start in September of 1974 and to get a peak linear heat rate of about 610 W/cm and a target burnup of 12,000 MWD/TMO in the Spring of 1976. The post irradiation test is to be finished by the Autumn of 1976.

2. Purpose of the Experiment

2.1 Object

The main objective of irradiating the UO_2-PuO_2 test fuel assembly is to examine:

1) The behaviour of plutonium-bearing ATR fuel rods which will be fabricated at the plutonium fuel fabrication facility of the PNC-Tokai-Works.
2) The effects of manufacturing tolerances of the pellet diameter, the straightness of cylindrical pellet surface, and plutonium homogeneity in the mixed oxide on heat transfer and fuel-cladding interaction.
3) The burnup characteristics of the plutonium-bearing fuel in boiling water reactor condition.

2.2 Desired Irradiation Condition

The desired irradiation condition for the IFA-423 fuel assembly in the HBWR are as follows:

1) Channel power	510 KW
2) Maximum fuel linear heat rate	610 W/cm
3) Maximum fuel surface heat flux	118 W/cm^2
4) Target burnup (peak rod)	12,000 MWD/T
5) Channel coolant condition	
a. Inlet flow velocity	more than 1.4 m/s
b. Total flow rate	more than 1.5 Kg/s

6) Core location of loading channel shall be so decide by the Halden Project as to obtain the desired channel power (510 KW).

3. Design Data Summary on the IFA-423 Fuel Assembly

3.1 Fuel assembly structure

The test fuel assembly is of a rod bundle type, and consists of seven single fuel rods. Each fuel rod contains a UO_2-PuO_2 pellet stacks of 1440 mm length. The fuel assembly will be as follows (reference to be attached drawing No. EH3-200 and EH3-300):

1) 7 fuel rods with intermediate locating spacers.
2) The fuel rods have an active column length of 1440 mm, containing dished pellets.
3) The 4 intermediately locating spacers tied up to 6 tie rods which are fixed to lower and upper tie plate.
4) 3 fuel rods out of 6 which locate at every 120° on the hexagon of the fuel bundle are fixed to the lower shroud tube extension by the end plug nut. The main IFA-423 fuel assembly design data are summarized in Table 3.1.

3.2 In-Core Instrumentation

The following instrumentation will be used to determine the operating thermal-hydraulic characteristics of the fuel assembly channel:

1) Turbine flow meters at channel inlet and outlet
2) Inlet and outlet coolant thermocouples
3) Six neutron thermometers to indicate channel power
4) Fuel channel steam sampling failure monitor.

Design and manufacture of the in-core instruments for the fuel assembly are to be undertaken by the Halden Project.

Table 3.1 Design Data Summary for IFA-423 Fuel Assembly

1. Assembly

1)	Number of fuel rods per assembly	7
2)	Configuration	closed hexagonal
3)	Pitch circle diameter	39.8 mm
4)	Number of spacers per assembly	4
5)	Number of tie rods per assembly	6
6)	Weight of UO_2-PuO_2 per assembly	17.1 Kg
7)	Weight of Pu fissile per assembly	137 g
8)	Weight of ^{235}U per assembly	1040 g

2. Fuel

1)	Material	cold pressed & sintered UO_2-PuO_2 pellets
2)	Enrichments	
	a. Uranium	7 w/o ^{235}U
	b. Plutonium	0.8 w/o (^{239}Pu+^{241}Pu)/MO
3)	Pellet density	95 % T.D
4)	Pellet diameter	14.40 mm
5)	Pellet height	16 mm
6)	Pellet end shape	dished
7)	Active fuel length	1440 mm

3. Cladding

1)	Material	Zircaloy-2
2)	Outer diameter	16.46 mm
3)	Inner diameter	14.70 mm
4)	Wall thickness	min. 0.8 mm
5)	Max. fuel-cladding diametral clearance	nominal 300 μm variable 200 ~ 400 μm
6)	Filling gas	He at 1 atm
7)	Plenum length	90 mm

4. Shroud

1)	Material	Zircaloy-2
2)	Inner diameter (minimum)	61.0 mm
3)	Length	1710 mm

4. Design Specifications

4.1 Fuels

The specifications of fuels are as follows:

Fuel type	cold pressed-sintered UO_2-PuO_2 pellets
Fuel composition	98.8 w/o UO_2-1.22 w/o PuO_2
Uranium enrichment	7.0 w/o ^{235}U
Plutonium enrichment	0.80 w/o Pu fissile/MO
Plutonium isotopic composition (w/o)	
Pu-238	0.85
Pu-239	64.76
Pu-240	21.87
Pu-241	9.49
Pu-242	3.03
Impurity content (ppm maximum)	
Typical element,	Al 100, B 1.0, Ca 100
	Cd 1.0, Si 200, Fe 200
	Mg 100, Ni 70, C 200
	N 200, F 25, Cl 25
Total,	4.0 ppm equivalent boron concentration
Moisture content	15 $\mu\ell$/g
Total gas content	60 $\mu\ell$/g
Pellet density	95.0 ± 1.5 % T.D
Pellet dimensions	
diameter	14.40 ± 0.05 mm
height	16.0 mm
dish diameter	8.0 mm
dish depth	0.2 mm

Effective fuel length	1440 ± 3 mm
Fuel stack weight	about 2.4 Kg

4.2 Fuel Rod Components

The specifications of fuel rod components are as follows:

1) Fuel cladding

Material	Zircaloy-2
Outer diameter	$16.46 \begin{smallmatrix} +0.00 \\ -0.08 \end{smallmatrix}$ mm
Inner diameter	14.70 ± 0.05 mm
Diametral gap fuel-cladding	nominal 300 μm variable 200 ～ 400 μm
Plenum length	90 mm
Drawing No.	EH3-221

2) End plugs(top and bottom)

Material	Zircaloy-2
Drawing No.	EH3-211 (top 1) EH3-222 (top 2) EH3-231 (bottom)

3) Plenum spring

Material	Inconel-X
Wire diameter	2.0 mm
Outer diameter	13.5 mm
Free length	110 mm
Effective number of turns	35
Spring constant	0.29 Kg/mm
Drawing No.	EH3-251

4) Thermal insulator

Material	ZrO_2
Outer diameter	14.40 mm

Thickness	5.0 mm
Density	more than 70 % T.D
Number	2 pieces per rod

4.3 Assembly Components

The specifications of assembly components are as follows:

1) Tie plates(top and bottom)

Material	AISI-304
Drawing No.	EH3-331 (top)
	EH3-341 (bottom)

2) Spacer

Material	Inconel-718
Drawing No.	EH3-310
Number	4 per assembly

3) Spacer tie rods(upper, intermediate, and lower)

Material	Zircaloy-2
Drawing No.	EH3-321 (upper)
	EH3-322 (intermediate)
	EH3-323 (lower)

4) Shroud tube

Material	Zircaloy-2
Inner diameter	$61.0^{+1.0}_{-0.0}$ mm
Wall thickness	about 1 mm
Length	1710 mm
Drawing No.	EH3-371

This component will be produced by the Halden Project.

5) Shroud tube supports(top and bottom)

Material	AISI-304 (casting)

Drawing No. EH3-351 (top)

EH3-361 (bottom)

This component will be produced by the Halden Project.

4.4 Assembling

The assembling for IFA-423 is performed at the HBWR site and the following parts are sent from PNC to the reactor site.

a. 7 fuel rods (DWG. No. EH-300)

b. Top and bottom tie plates

c. 4 spacers

d. One upper, three intermediate, and one lower spacer-tie-rods.

e. 12 nuts for tie rod (DWG. No. EH3-411).

f. 4 nuts for fuel rod (DWG. No. EH3-412).

g. 3 bottom guide rods (DWG. No. EH3-413).

h. 3 guide rod nuts (DWG. No. EH3-414).

i. 12 spring washers for tie rod (DWG. No. EH3-421).

j. 4 spring washers for fuel rod (DWG. No. EH3-422).

k. 6 spring washers for guide rod (DWG. No. EH3-423).

l. 12 screws for shroud (DWG. No. EH3-372).

5. Design Calculation and Fuel Performance Data

5.1 Fuel Test Condition and Fuel Performance Data Summary

A typical arrangement of fuel test channel in the HBWR during three years period 1973-75 is shown in Fig. 5.1.1 (refer to references (1)). Final decision of loading channel for the IFA-423 is to be made by the Halden Project as obtain a desired channel power

Same relevant operating data and the fuel test conditions are given in Table 5.1.1 from Halden Project's proposal[1]. These data are used as design basis for the design calculation described below. The design calculation data and fuel performance data are summarized in Table 5.1.2.

Table 5.1.1 Reactor Operating Data and Fuel Test Condition[1]

HBWR power level	~16 MW
Reactor and coolant pressure	34 atm
Heavy water saturation temperature	240°C
Channel inlet temperature	238°C
Average fuel power density in third charge UO_2 fuel	19.8 w/g
Average thermal neutron flux in third charge UO_2 fuel	4.8×10^{13} n/cm^2.sec
Core active length	170 cm
Maximum diameter of fuel test assembly	73 mm
Power form factors (no control rod inserted)	
Fuel with 1.7 m length	1.23
Fuel with 1.5 m length	1.16

Coolant flow

Natural circulation typical range	depends on channel design 0.5 to 2 Kg/s
Forced circulation range	2 to 4 Kg/s
Coolant inlet velocity	refer to Fig. 5.1.2

Table 5.1.2 Design Fuel Performance Data

1) Power	
Channel power	510 KW
Peak linear heat rate	610 W/cm
Average linear heat rate	502 W/cm^2
Peak surface heat flux	118 W/cm^2
Average surface heat flux	97.0 W/cm^2
Maximum rod burnup	12,000 MWD/T
2) Power distribution	
Radial form factor	1.048
Dip factor in fuel (Power ratio of fuel surface to center)	1.049
Axial form factor	1.16*
3) Thermal-hydrodynamics	
Coolant flow rate	1.56 Kg/s
Av. mass velocity	0.127 $Kg/s.cm^2$
Av. exit quality	20.0 %
Hot channel exit quality	26.2 %
Minimum burnout ratio	1.77

* given data

4) Temperature distribution

Max. fuel center	2430°C
Fuel surface	556°C
Cladding inner surface	318°C
Cladding outer surface	253°C

5) Maximum cladding stress

	Hot clean	12,000 MWD/T
Internal pressure	29.5	71.1 Kg/cm^2
Thermal stress	2.86	2.86 Kg/mm^2
Pressure stress	-1.06	2.76 Kg/mm^2
Pressure stress with loss of coolant pressure	2.71	6.53 Kg/mm^2

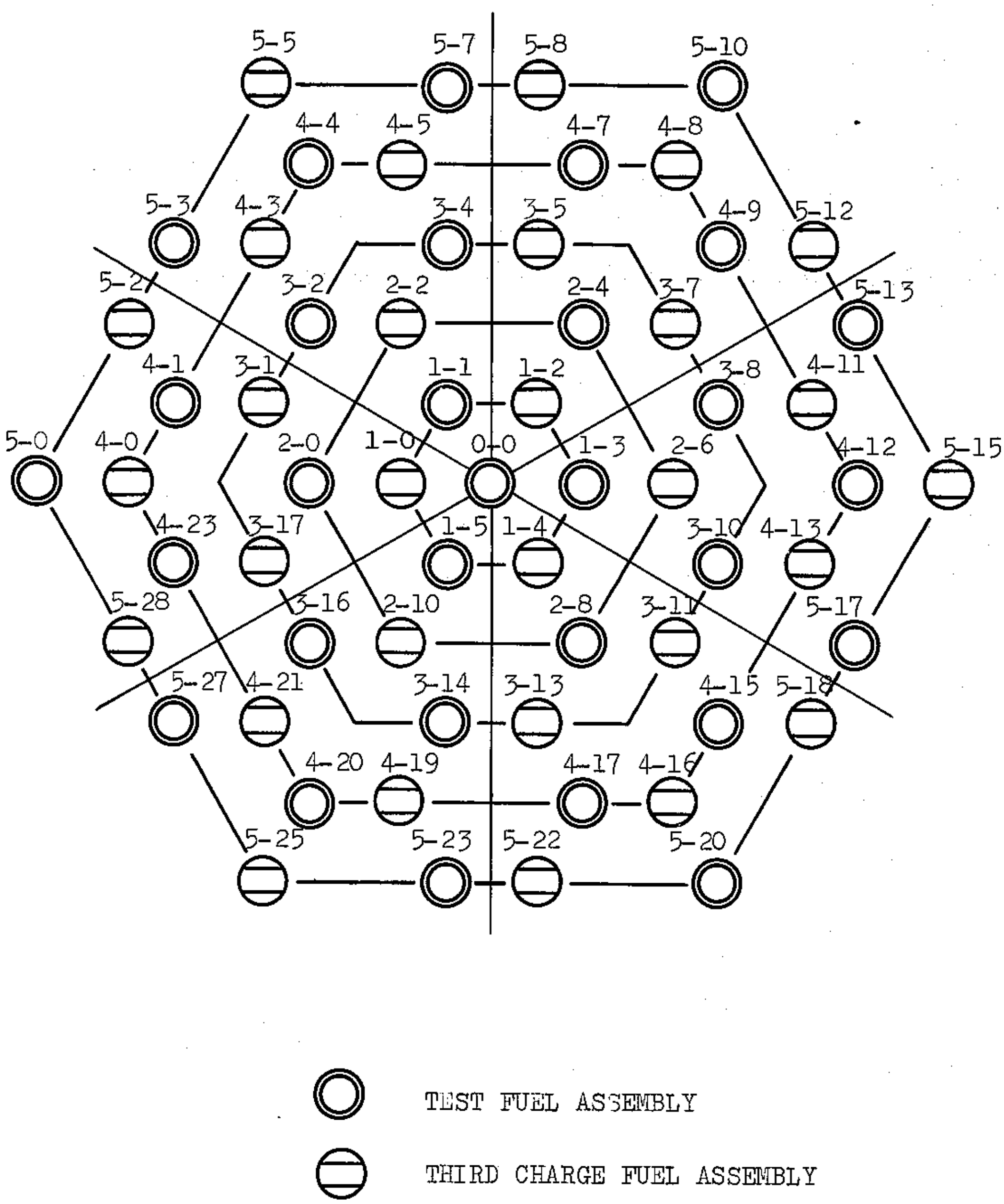


Fig. 5.1.1 Typical Arrangement of Fuel channel in the HBWR Core.

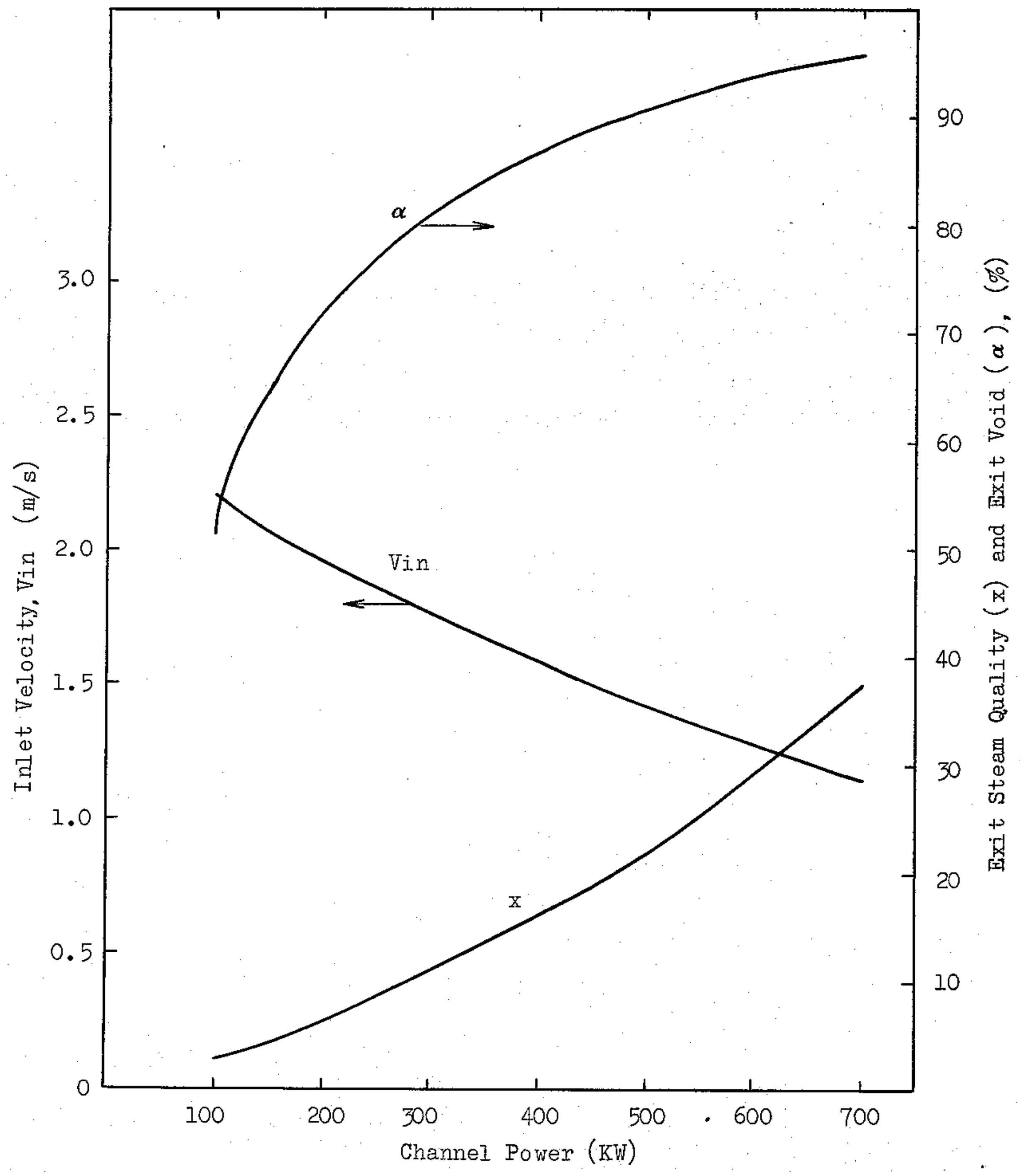


Fig. 5.1.2 Inlet Velocity, Exit Steam Quality and Exit void vs Channel Power in HBWR[2)]

5.2 Power distribution

5.2.1 Channel power and uranium enrichment

The fissile enrichment in the fuel that can achieve the above desired channel power, are determined assuming a constant fissile Pu content of 0.8 W/o, and varing U-235 enrichment.

From Halden project's proposal and METHUSELAH-II code, the average specific powers in the HBWR for IFA-423 were estimated as a function of the U-235 enrichment. As a basis for the calculation, it has been assumed that the IFA-423 are loaded in a channel with the average thermal neutron flux on the HBWR. The results are shown in Fig. 5.2.2 and the channel powers are shown in Fig. 5.2.3 with the estimated power. The estimated power are obtained by multipling the specific power by the weight of fuel in IFA-423. Fig. 5.2.1 shows the radial power form factor in the IFA-423 channel as a function of U-235 enrichment to obtained from the calculation of METHUSELAH-II[3].

On the other hand, the desired channel powers were calculated as follows:

a) Desired maximum linear heat rate, 610 W/cm

b) Average rod power of outer rods,

$$P_{outer} = 610\ \text{W/cm} \times \frac{1.0}{P_{AX}} \times 144\ \text{cm}$$

c) Average rod power of center rod,

$$P_{center} = 610\ \text{W/cm} \times \frac{P_R}{P_{AX}} \times 144\ \text{cm}$$

d) Channel power,

$$P_{assy} = P_{outer} \times 6^{rods} + P_{center} \times 1^{rod} \text{ (KW)}$$

Fig. 5.2.3 shows the results of the desired channel power, which is slightly decreased with the U-235 enrichment because of increasing the radial power form factor.

The uranium enrichment of 7 W/o for the IFA-423 are obtained from an intersection of the estimated and the desired powers in Fig. 5.2.3. The channel power of the IFA-423 is around 510 KW.

5.2.2 Radial power distribution in assembly with fuel burnup

As for IFA-423 fuel assembly with 1.22 W/o PuO_2 and 7 W/o U-235 enriched fuel, the radial power distribution in assembly with burnup were calculated by the METHUSELAH-II as assuming the constant channel power. The results of the radial power form factor shows Fig. 5.2.4 as a function of burnup to 12,000 MWD/T. The power form factor at final burnup is about 6 % higher than that of initial value.

5.2.3 Power distribution in fuel rod

The power distribution in fuel rod was calculated by METHUSELAH-II code and it was obtained as κ value of $Y = AI_o(\kappa r)$, by Bessel function fitting code. The κ value of 0.609 with 12,000 MWD/T was used calculation of temperature distribution in the fuel rod.

5.2.4 Axial power distribution

There is difficult to strictly evaluated the power distribution of axial direction because it depend on the fuel loading arrangement and the operating conditions.

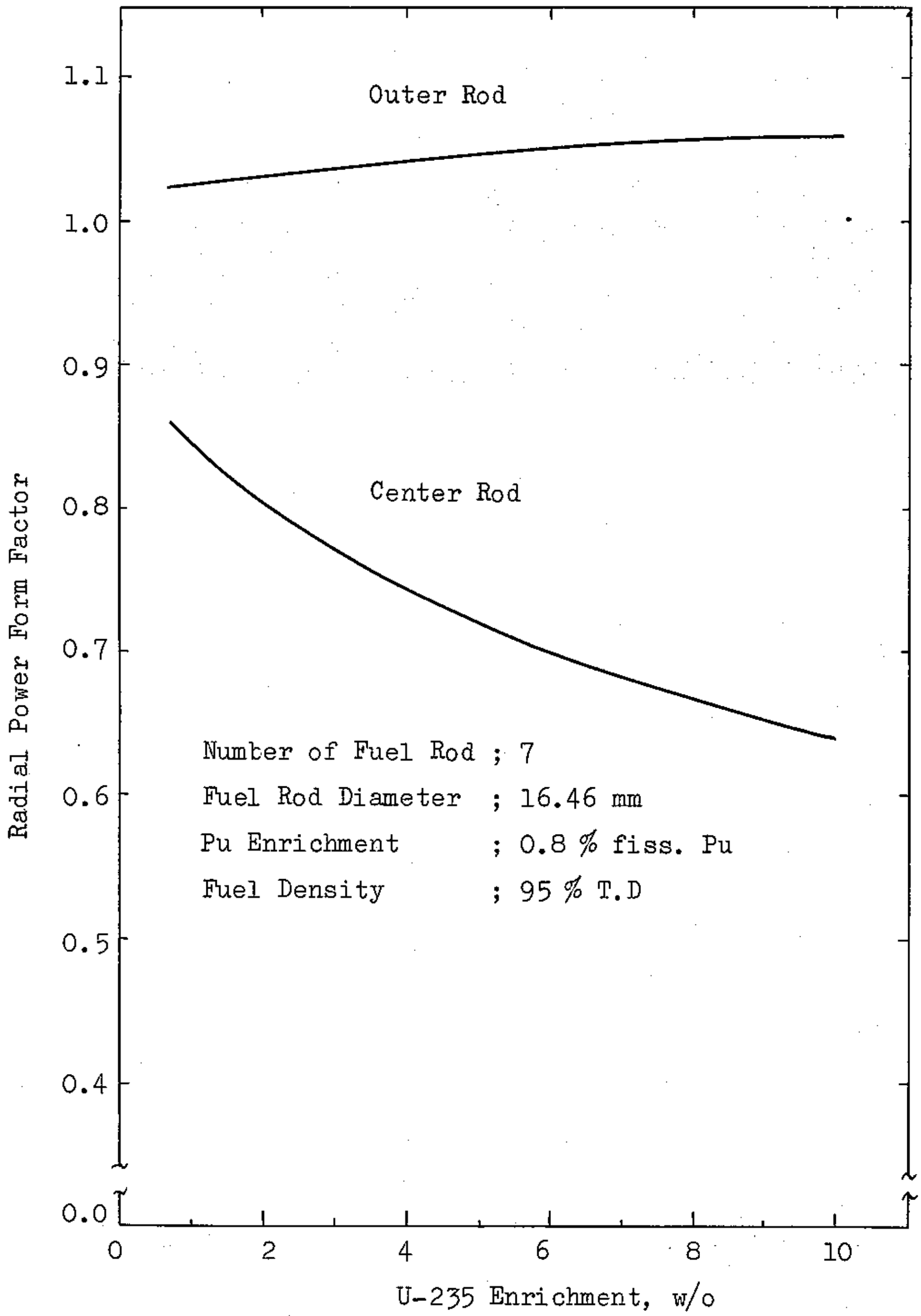


Fig. 5.2.1 Radial Form Factor in IFA-423 Assembly

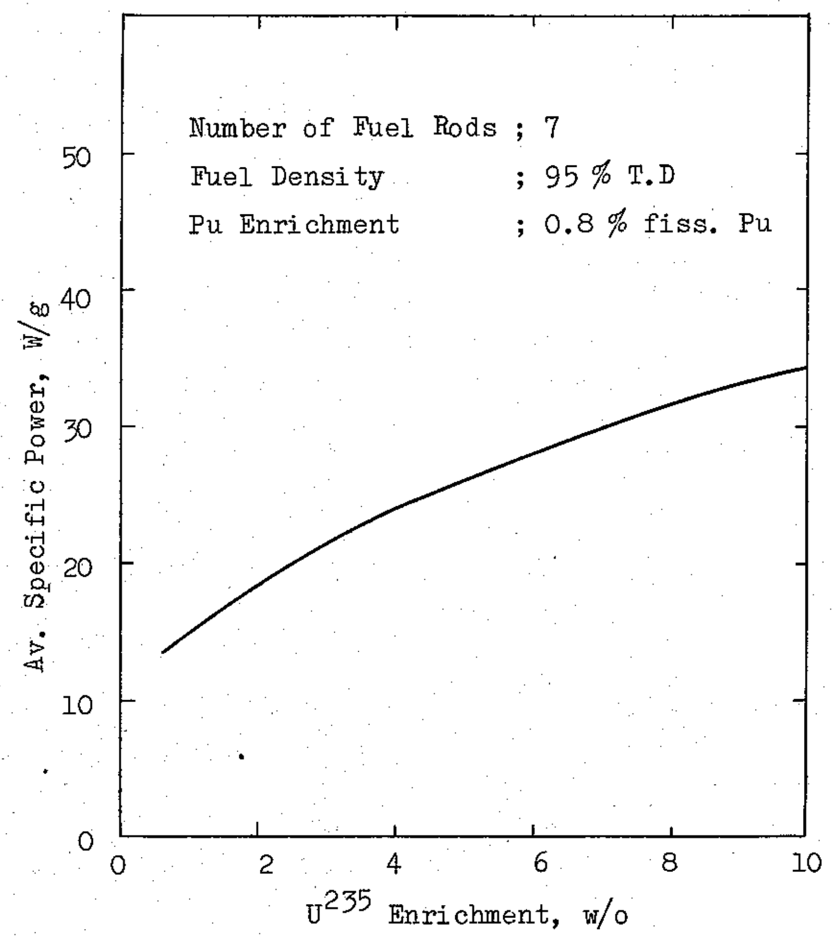


Fig. 5,2,2 Estimated Specific Power of IFA-423 at Average Flux in HBWR

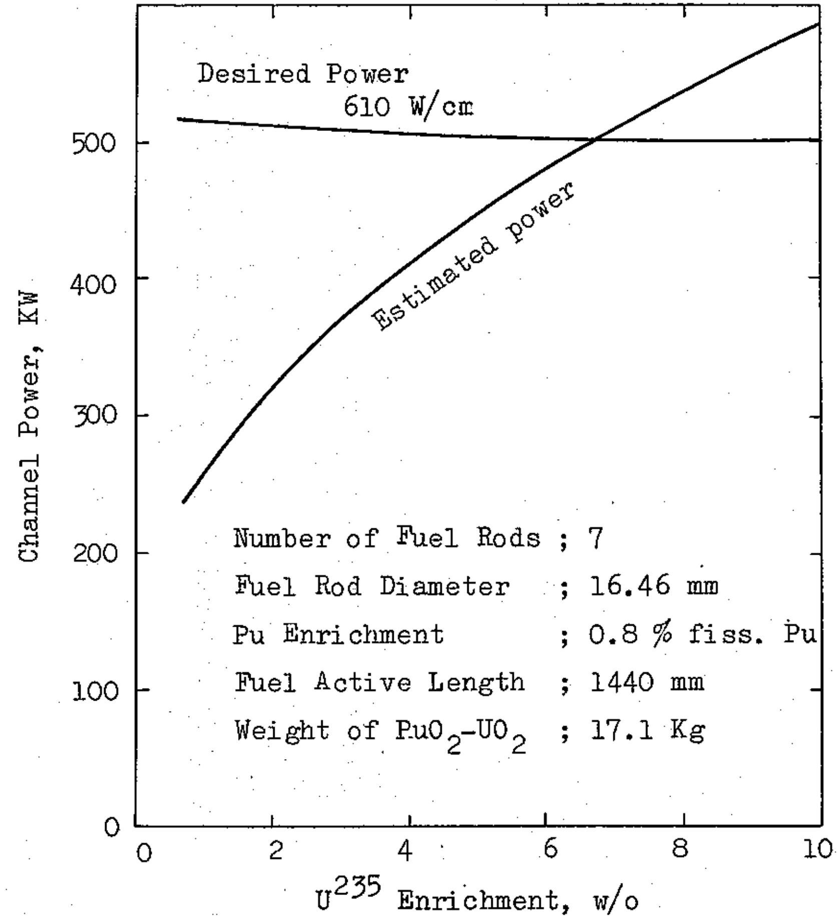


Fig. 5.2.3 Channel Power of IFA-423

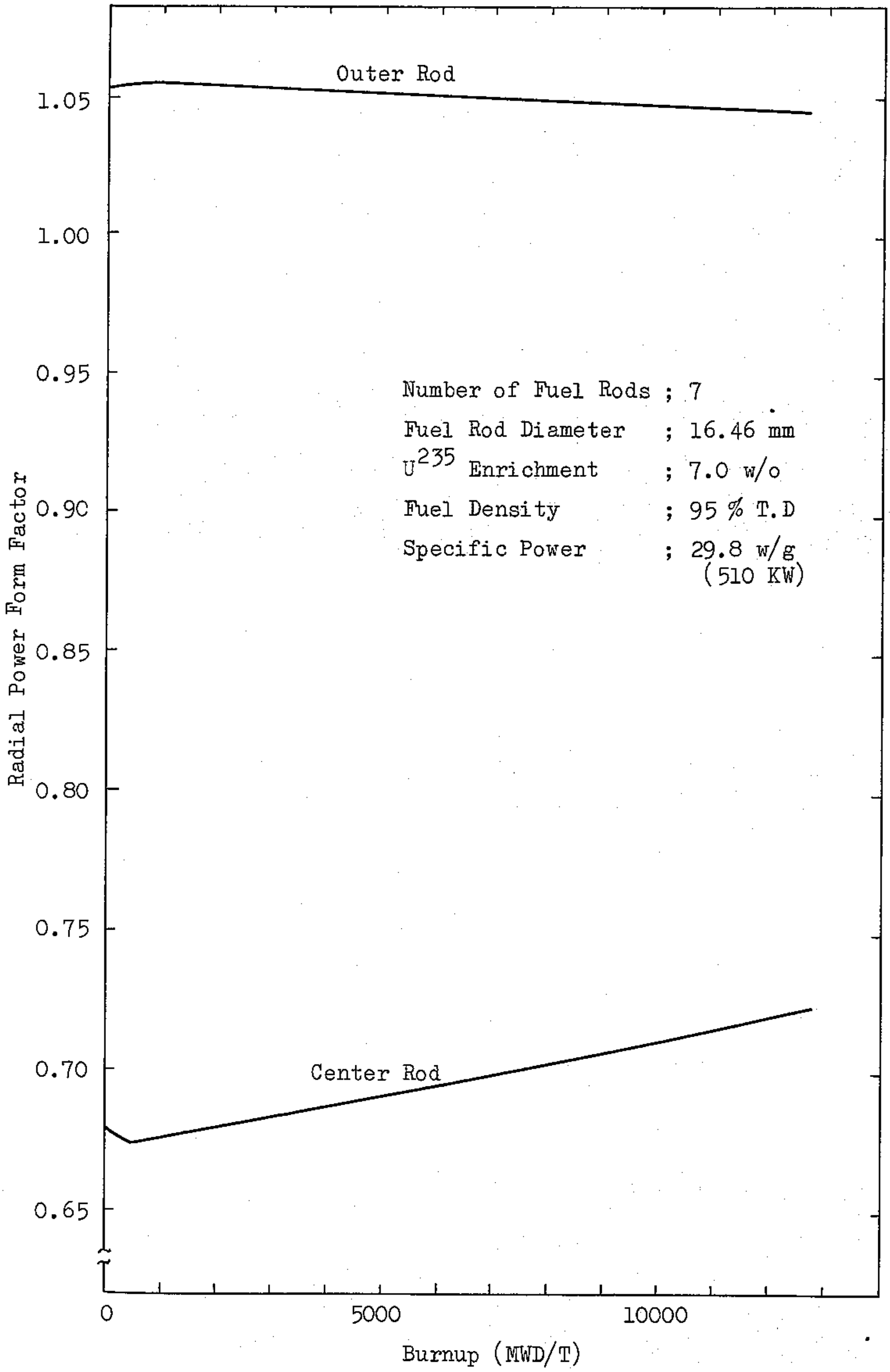


Fig. 5.2.4 Power Distribution vs Burnup for IFA-423 Assembly

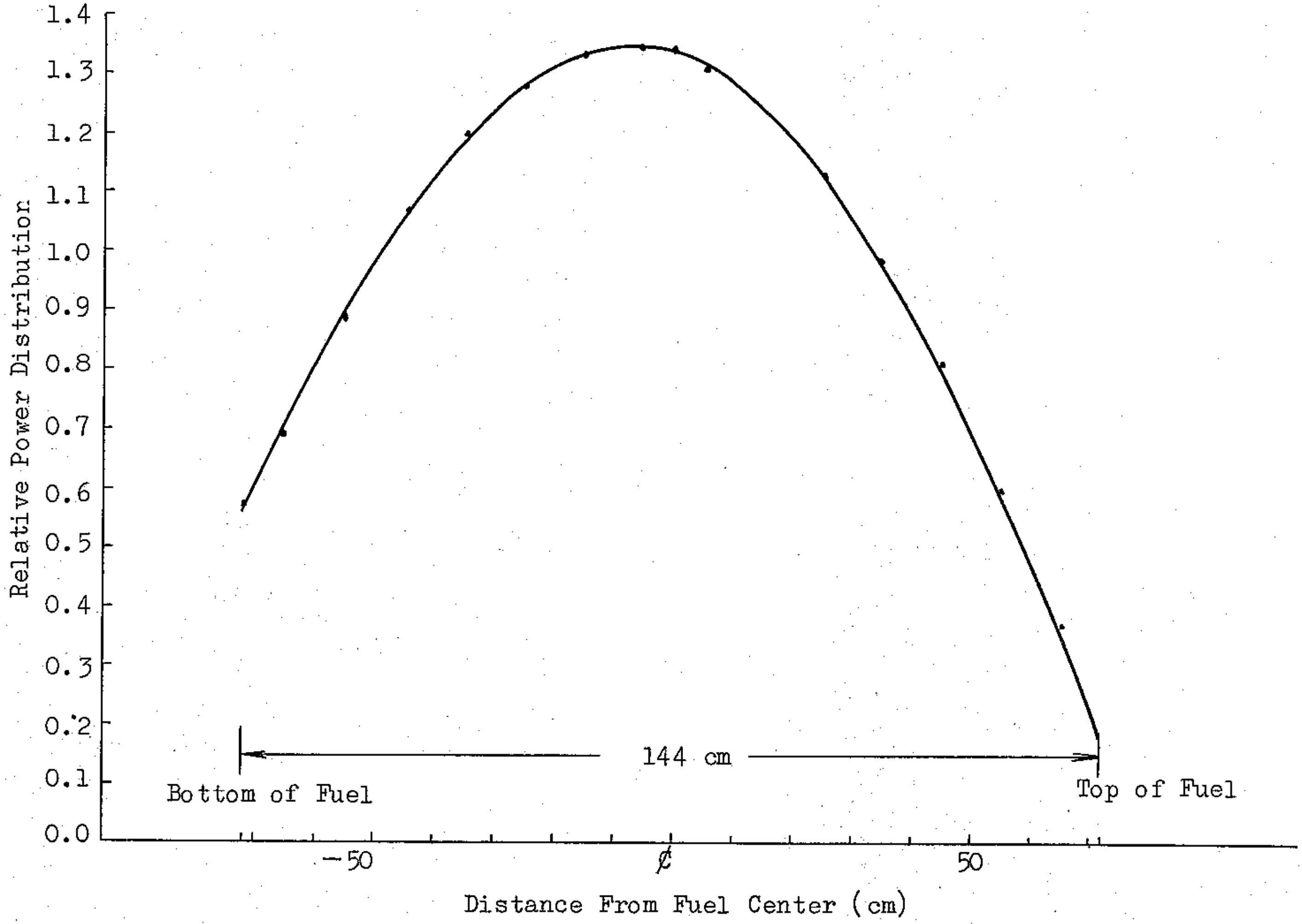


Fig. 5.2.5 Power Distribution of Axial Direction in IFA-423.

However, the axial power profile to be request in the design calculation was prepared as a basis of γ-scanning data of post-irradiation examination for the IFA-159 and IFA-160[4]. It is shown in Fig. 5.2.5.

The axial power form factors are assumed as follows:

1) In the calculation of channel power, we used a value of 1.16 with 1500 mm length in Table 5.1.1.

2) In case of the thermal and hydraulic calculation, we used a value of 1.35 that recommended on safety analysis at Halden Project.

5.3 Thermal-Hydraulic Characteristics

5.3.1 Calculation basis

The IFA-423 fuel assembly was designed so that the following conservative limits are not exceeded during normal HBWR operation.

a) Minimum burnout ratio (MBOR) greater than 1.50.

b) Hot channel factor defined the ratio of hot sub-channel quality to average channel quality less than 1.30.

c) Maximum local quality in channel below 30 %.

The thermal-hydraulic analysis for the IFA-423 assembly were carried out using the COBRA-II[5] code as a function of the channel power. The burnout correlation of Janssen-Levy[6] was used for the burnout ratio evaluations. The parameter used are summarized in Table 5.3.1. One-twelveth symmetry of channel geometry was used on calculation of COBRA-II, as shown in Fig. 5.3.1.

The channel was divided into three parts sub-channels, which were numbered from inner to outer.

5.3.2 Channel Averaged Characteristics

The results of the thermal-hydraulic characteristics on the channel averaged are shown in Fig. 5.3.2 and Fig. 5.3.3. Fig. 5.3.2 shows the MBOR and exit quality as a function of the channel power, and Fig. 5.3.3 shows the same values with the channel flow rate and channel inlet velocity.

The MBOR is about 1.77 for the IFA-423 design with the inlet velocity 1.4 m/s at channel power 510 KW, and the exit steam quality is around 20 % of the channel averaged.

So as not to be less than 1.50 of MBOR, it is necessary that the IFA-423 fuel assembly will be operated at a condition of channel flow rate above 1.3 kg/s or of inlet flow velocity above 1.2 m/s from Fig. 5.3.3.

5.3.3 Sub-channel Characteristics

The results of the steam quality and mass velocity distribution in axial direction on the hottest sub-channel and average channel are shown in Fig. 5.3.4. Fig. 5.3.5 shows the burnout ratios and heat flux distributions in axial direction on the center fuel rod and the hottest sub-channel. Fig. 5.3.6 shows the mass velocity and quality distribution in exit cross-section and the MBOR on each sub-channels. According to the Fig. 5.3.6, the hottest sub-channel is a sub-channel No.1, the exit quality becames around 26.2 %, the 1.77 of MBOR occure inside of the outer fuel rods (the fuel surface No.II in Fig. 5.3.1). The hot channel factor described in section 5.3.1 is around 1.30.

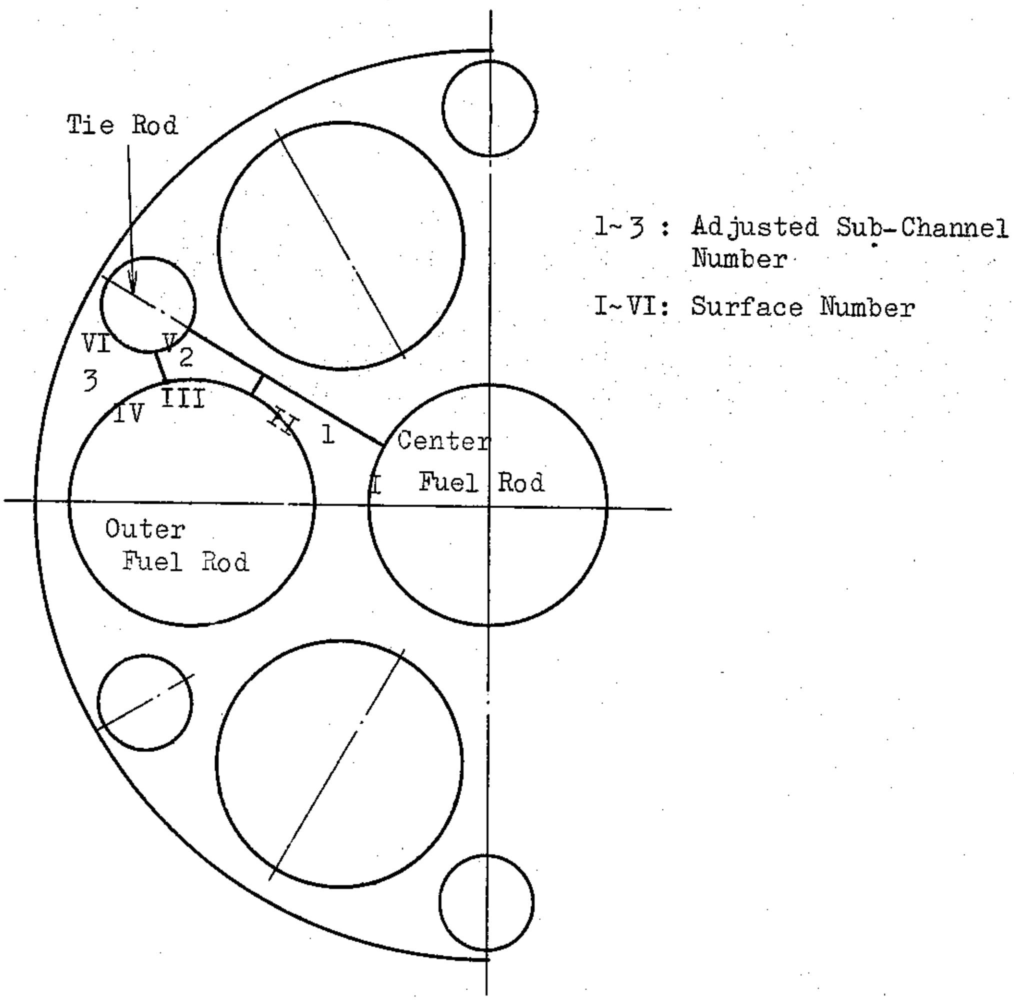


Fig. 5.3.1 Calculation Model of the Thermal-Hydraulic Characteristics (1/12 Cross-Section)

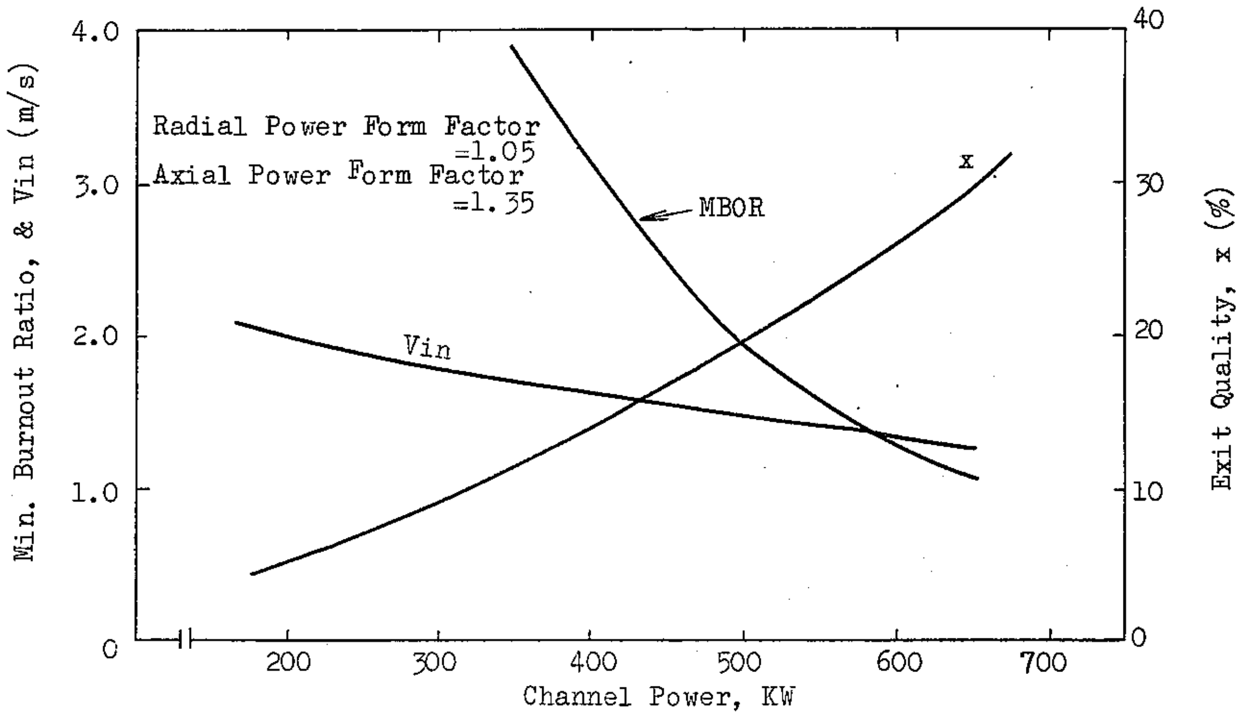


Fig. 5.3.2 Burnout Evaluation vs Channel Power for IFA-423

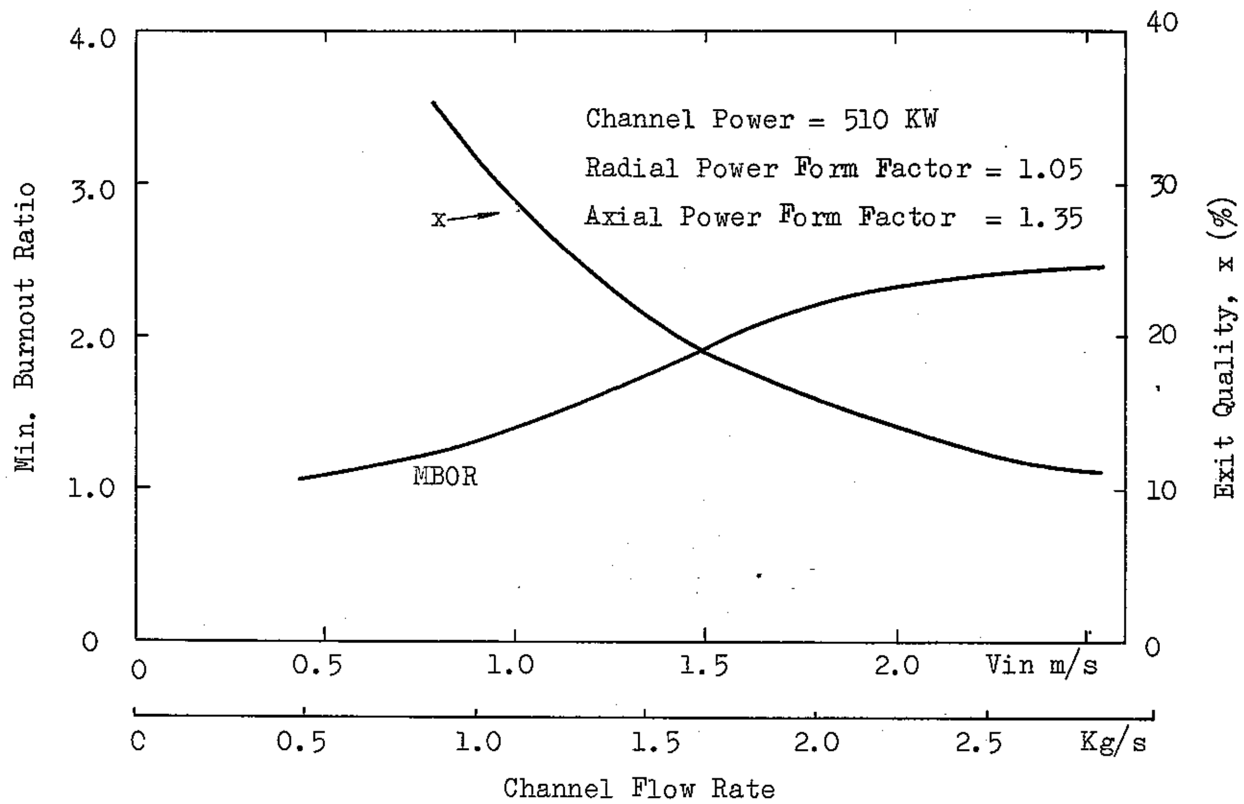


Fig. 5.3.3 Burnout Evaluation vs Channel Flow for IFA-423

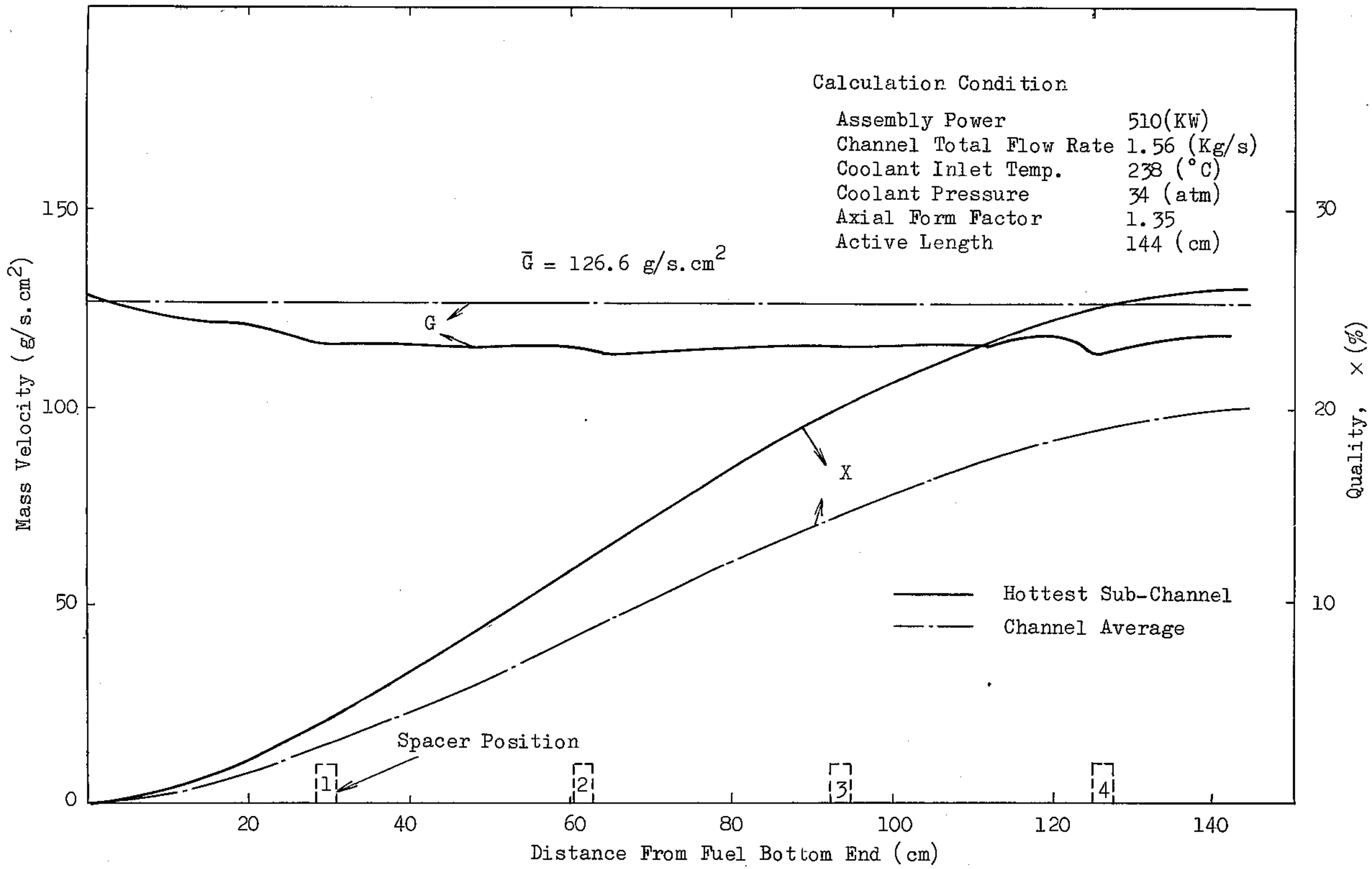


Fig. 5.3.4 Thermal-Hydraulic Character in Sub-Channel of IFA-423

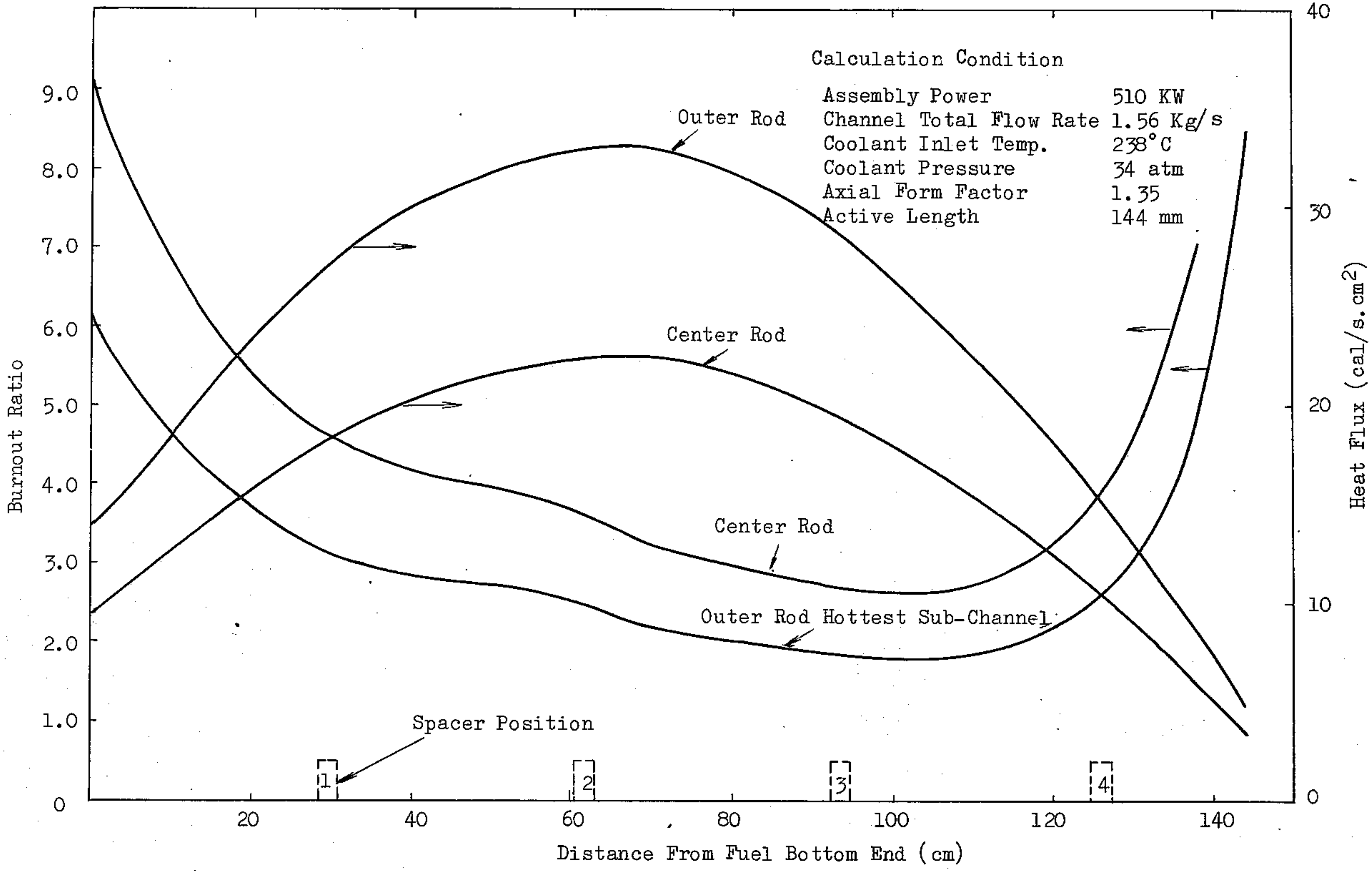
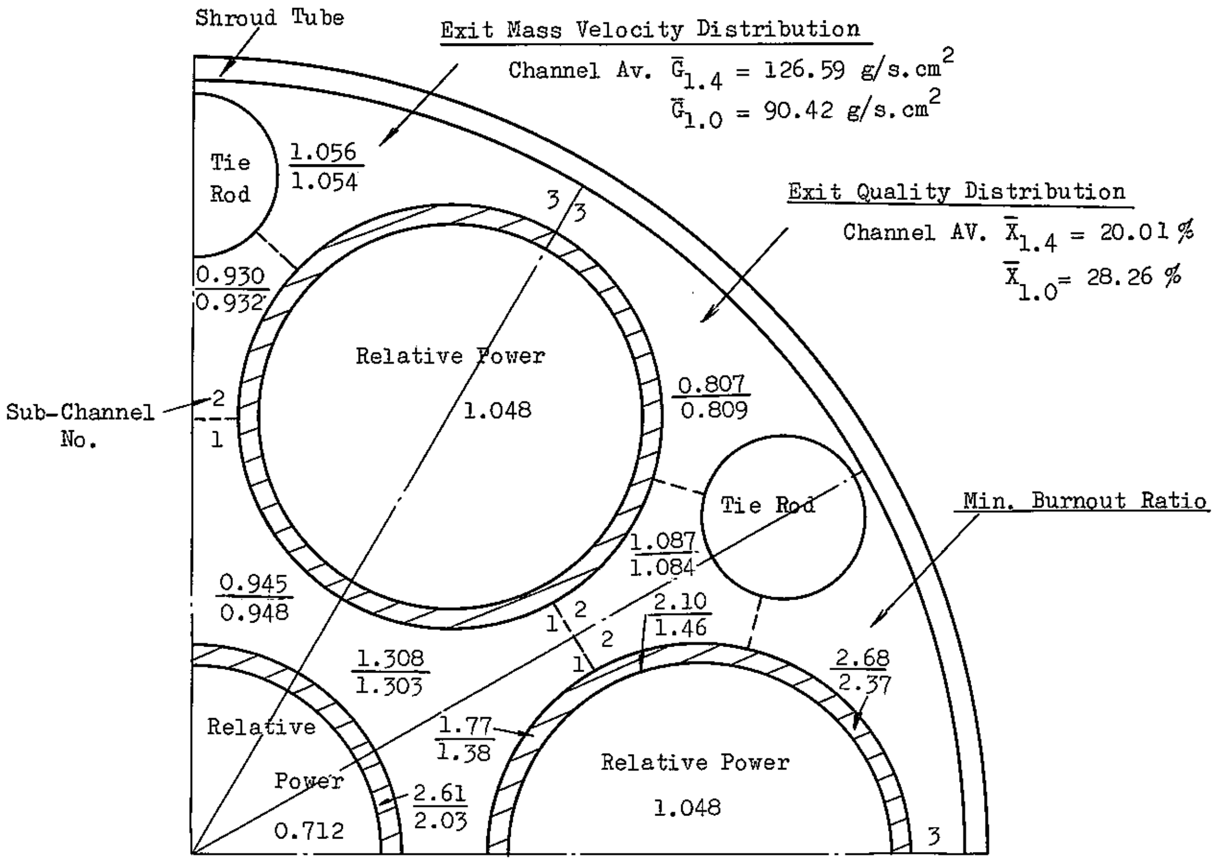


Fig. 5.3.5 Thermal-Hydraulic Character in Channel of IFA-423



Calculation Condition

Assembly Power	510 KW	
Channel Total Flow Rate	1.56 Kg/s (1.4 m/s) 1.12 Kg/s (1.0 m/s)	
Coolant Inlet temp.	238°C	
Coolant Pressure	34 atm	
Axial Power Form Factor	1.35	

Note

$$\frac{\text{Vin} = 1.4\ \text{m/s}}{\text{Vin} = 1.0\ \text{m/s}}$$

Fig. 5.3.6 Thermal-Hydraulic Character in Sub-Channel of IFA-423 Fuel Assembly

Table 5.3.1 Thermal-Hydraulic Parameter Used

1) Channel parameters

Channel power	510 KW
Channel flow	1.56 Kg/s (1.4 m/s)
Inlet enthalpy	237.7 cal/g
Saturated liquid enthalpy	240.0 cal/g
Coolant pressure	34 atm
Latent heat of vaporization	383.5 cal/g
Vapor to liquid density ratio	0.0209

2) Peaking factor

Axial form factor	1.35
Radial form factor	1.048 (outer rod)
	0.712 (Center rod)

3) Assembly geometry parameter

Coolant flow area	12.34 cm^2
Channel heated length	144 cm

4) Subchannel data

	No. 1	No. 2	No. 3
Subchannel flow area (cm^2)	0.325	0.172	0.531
Subchannel equivalent diameter (cm)	1.007	0.723	0.615
Wetted perimeter (cm)	1.293	0.890	3.451
Heated perimeter (cm)	1.293	0.628	1.096

5.4 Temperature distributions

The temperature distributions of the test fuel assembly were calculated as a parameter of linear heat rate. This calculation had been done for the following condition and data.

(i) Coolant temperature, 240°C

(ii) Coolant pressure, 34 atm

(iii) Film heat transfer coefficient

Film heat transfer coefficient between coolant and cladding outer surface is calculated by Jens-Lotte's equation.

(iv) Thermal conductivity of Zircaloy-2 cladding as follows;

$$k_C = 7.97 + 0.00316T \ (\mathrm{Btu/ft^2.hr.^oF})$$

where T; Temperature, ^oF

(v) Gap thermal conductance between cladding inner surface and fuel surface.

$$k_G = 1000 \ \mathrm{Btu/ft^2.hr.^oF}$$

(vi) Thermal conductivity of UO_2-PuO_2 pellet is used Lyon's equation[7].

(vii) The power distribution in the test fuel was calculated by computer code METHUSELAH-II[3] and it is given as a Bessel functions of I_0. (for temperature calculation)

Fig. 5.4.1 shows temperature distribution in radial direction. Fig. 5.4.2 shows temperature of fuel centerline, fuel surface, cladding inner surface and outer surface as a parameter of linear heat rate.

The temperature of fuel centerline is around 2430°C at the maximum linear heat rate (610 W/cm). This temperature is less than fuel melting point.

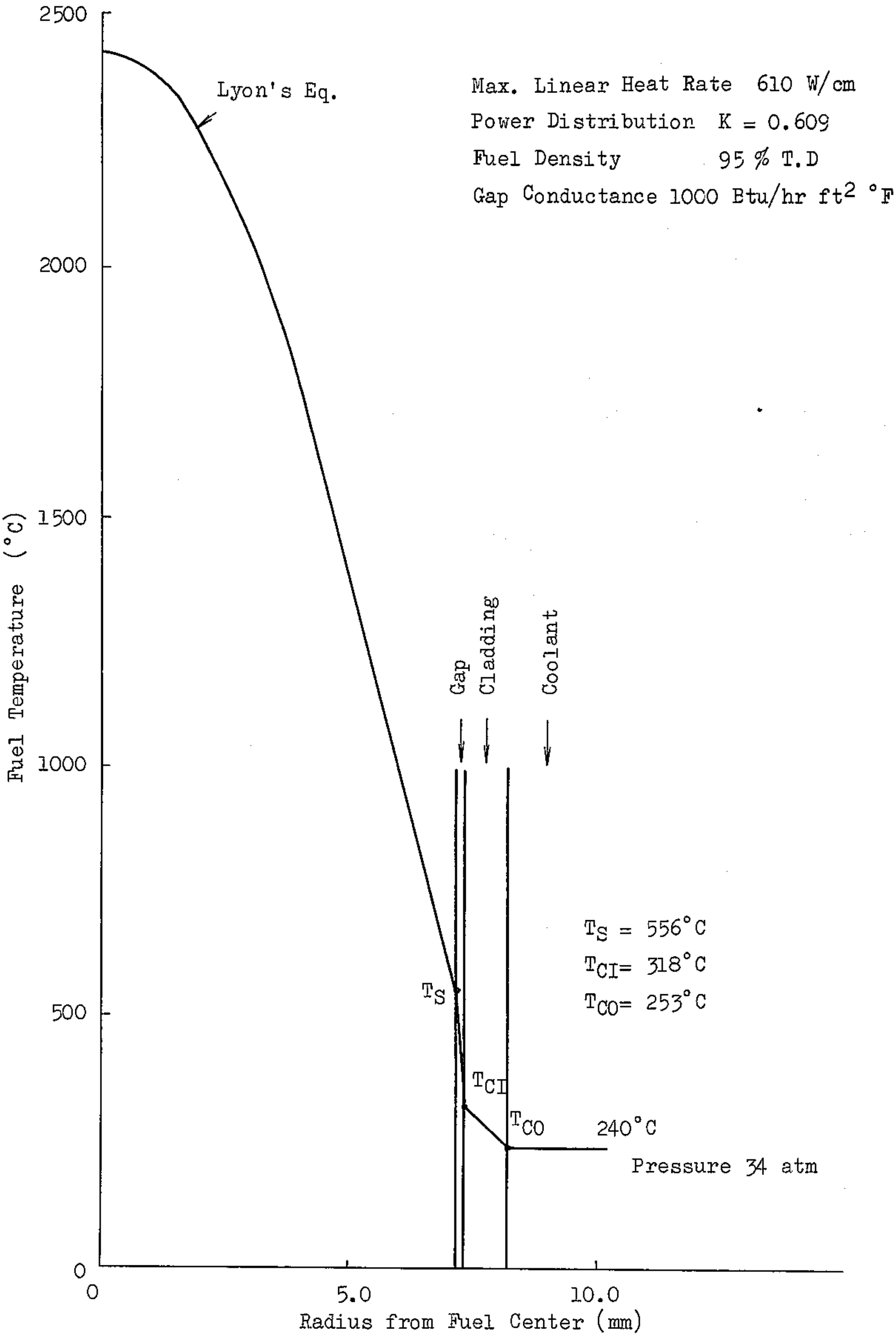


Fig.5.4.1 Fuel Temperature Distribution at Desired Power of IFA-423

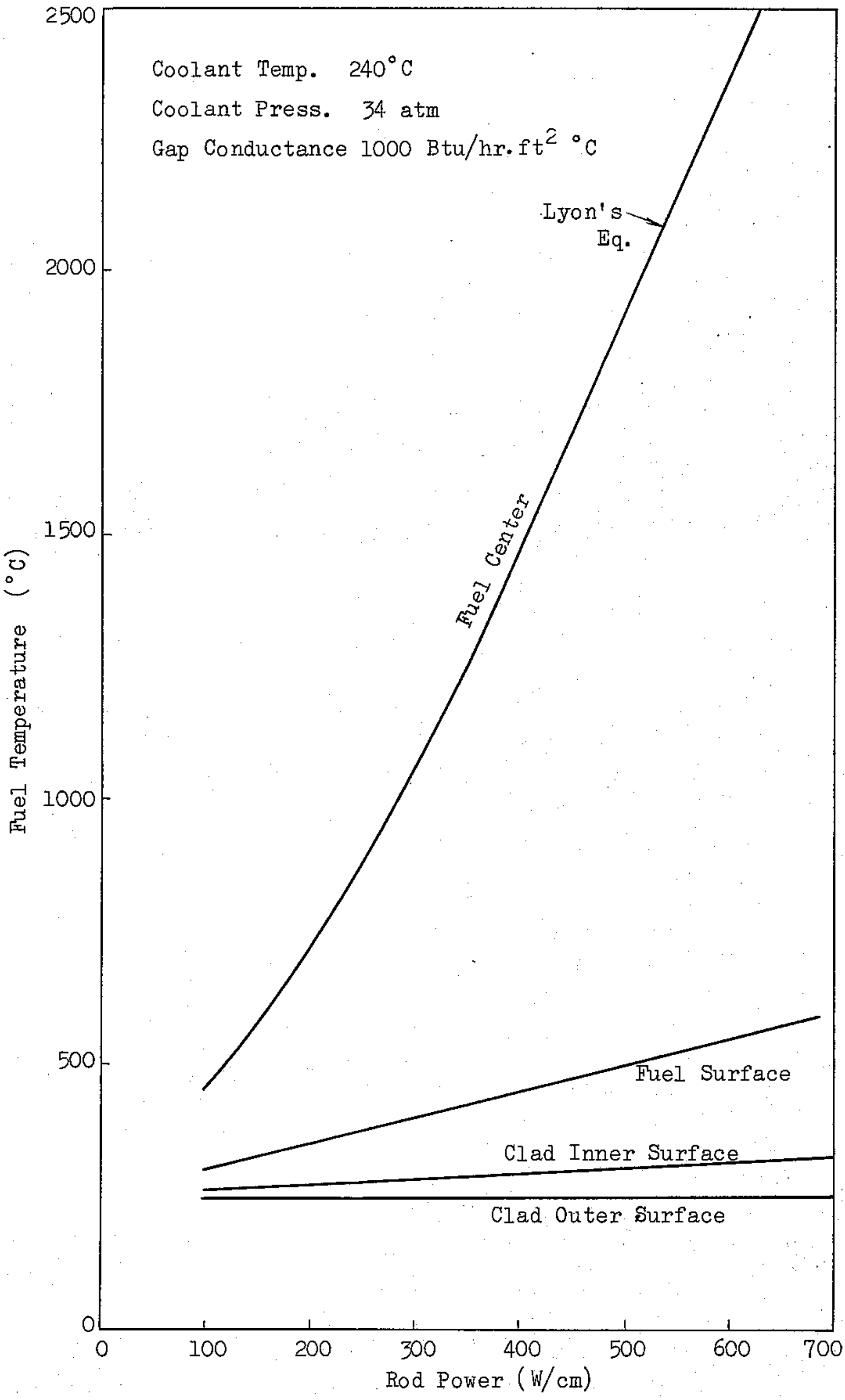


Fig. 5.4.2 Fuel Temperature Distribution of IFA-423

5.5 Cladding stress

The cladding stress of test fuel assembly was calculated as a parameter of burnup as follows:

a) Design basis

i) Maximum coolant pressure,	34 atm
ii) Maximum linear heat rate,	610 W/cm
iii) Fuel rod specification	
Cladding outer dia.	16.46 mm
Cladding inner dia.	14.70 mm
Cladding thickness	0.86 mm
(min. thickness 0.8 mm)	
Cladding ovality	0.02 mm
Weight of UO_2- PuO_2	2.44 Kg/rod
Plenum volume	11.3 cc/rod
Filling gas	He at 1 atm

b) Results of calculations

The results are shown in Table 5.5.1 and Fig. 5.5.1 as a parameter of burnup at normal operating and outer pressure loss accident.

The maximum cladding stress in case of outer pressure loss accident is far below than the yeild strength of high temperature for un-irradiated Zircaloy-2. The stress evaluation completely satisfied to a standard of ASME Section-III[8] etc.

Table 5.5.1 Cladding stress summary

a) Stress caused by coolant pressure

1)	Tangential stress	3.77 Kg/mm^2
2)	Axial stress	1.81 Kg/mm^2
3)	Thermal stress	2.86 Kg/mm^2

b) Stress caused by internal pressure

Burnup (MWD/T)	Internal Press. (Kg/mm^2)	Tangential Stress (Kg/mm^2)	Axial Stress (Kg/mm^2)
Initial	0.30	2.71	1.34
1000	0.33	3.03	1.52
3000	0.40	3.68	1.84
6000	0.50	4.62	2.31
9000	0.61	5.58	2.79
12000	0.71	6.53	3.27

c) Maximum cladding stress (Pressure stress)

Burnup (MWD/T)	Max. tangential stress (Kg/mm^2)	
	normal operating	outer pressure loss accident
initial	-1.06	2.71
1000	-0.68	3.03
3000	-0.09	3.68
6000	0.85	4.62
9000	1.81	5.58
12000	2.76	6.53

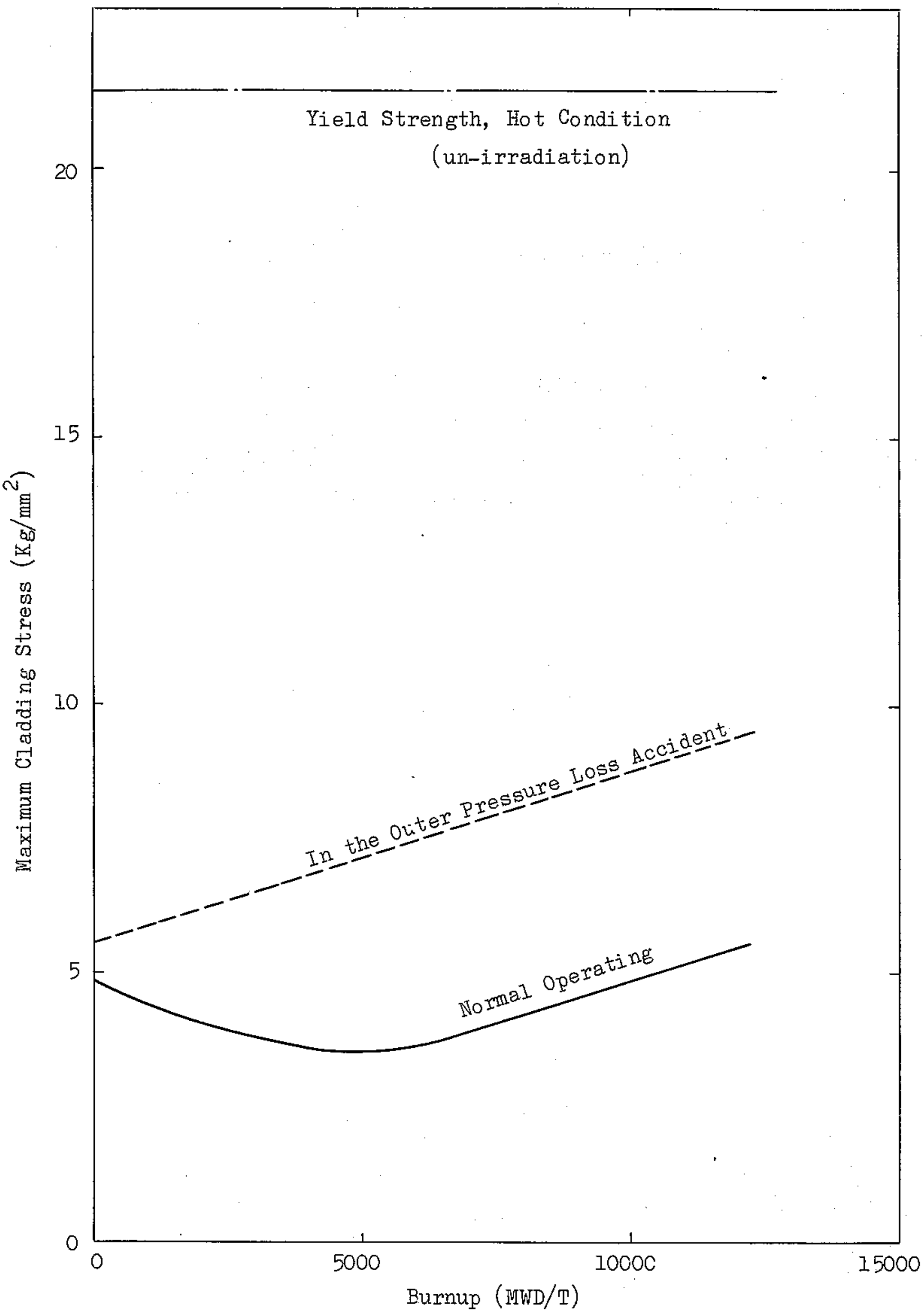


Fig. 5.5.1 Maximum Cladding Stress of IFA-423

5.6 Fuel Burnup

The fuel burnup for the IFA-423 are obtained as a function of operating times assuming a constant channel power and a load factor of 65 %[1], and are shown in Fig. 5.6.1. It is necessary about twenty monthes of irradiation period that the peak rod obtained 12,000 MWD/T of burnup.

5.7 Conclusion

It can be concluded that the irradiation examination for the IFA-423 fuel assembly will be carried out safety in the HBWR, as described in the above analyses.

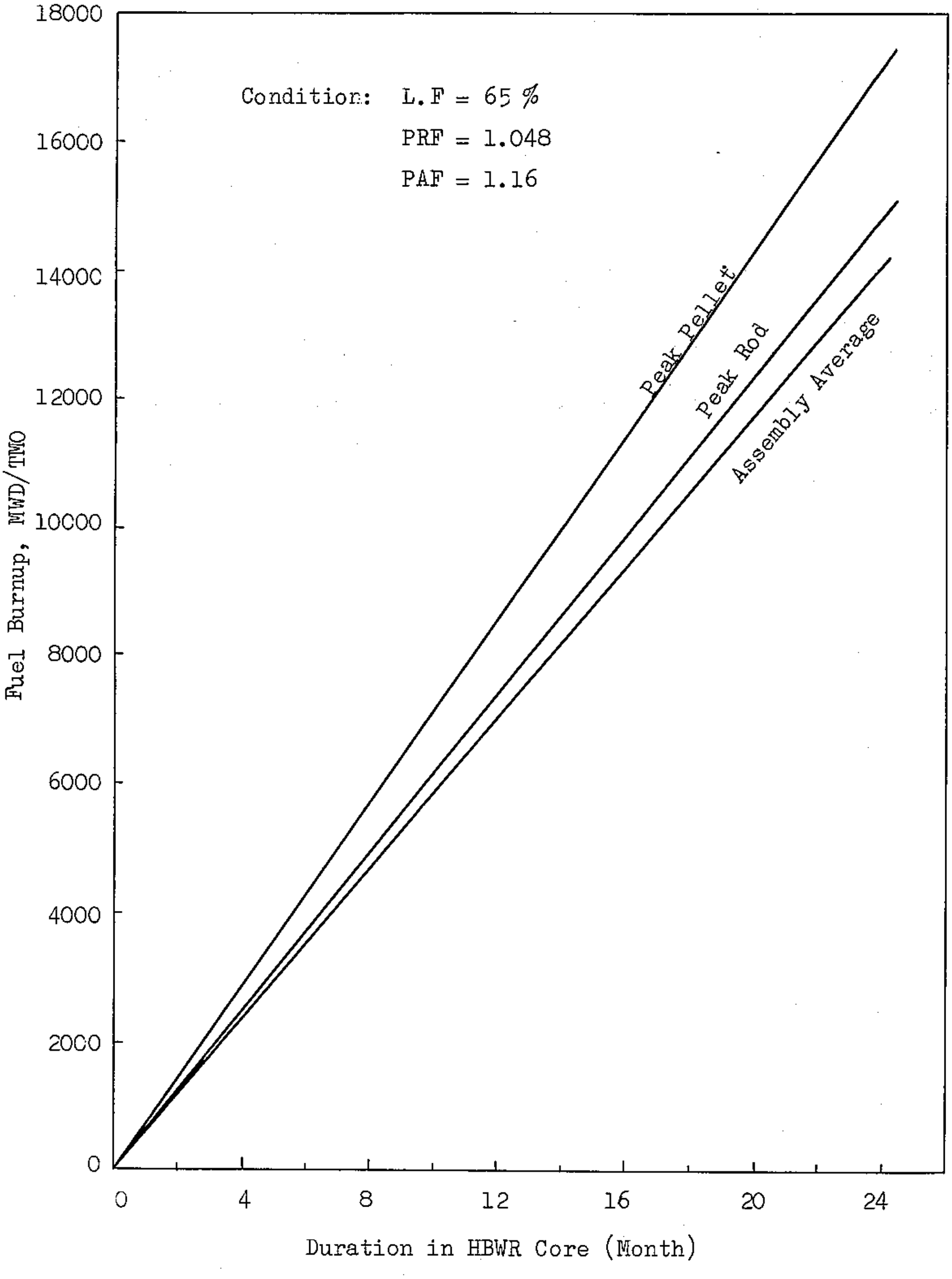


Fig. 5.6.1 Assumed Burnup of IFA-423

6. Post-Irradiation Examination Program

The post-irradiation examination program of the fuel assembly which will be carried out in the hot laboratory at Kjeller Research Center is as follows:

1) Nondestructive tests of fuel rods

a.	Visual examination and photography	~ 7 rods
b.	Gamma scanning	~ 7 rods
c.	Dimensional measurements (Length and rod dia. profile)	~ 7 rods
d.	Neutron radiography	~ 3 rods
e.	Eddy current test	~ 4 rods

2) Destructive tests of fuel rods

a.	Fission-gas sampling and analysis	2~3 rods
b.	Fuel & cladding metallography (Transverse and longitudinal)	6~8 sections
c.	β & γ autoradiography	4~6 sections
d.	α autoradiography	4~6 sections
e.	Fuel pellets density (pycnometry)	2~3 samples
f.	Burnup studies (Nd & hevey elements),	4~6 sections
g.	Radial burnup studies (~ 10 samples/section), The samples will be taken by micro-drilling method	1~3 sections
h.	Electron microprobe analysis	1~3 sections

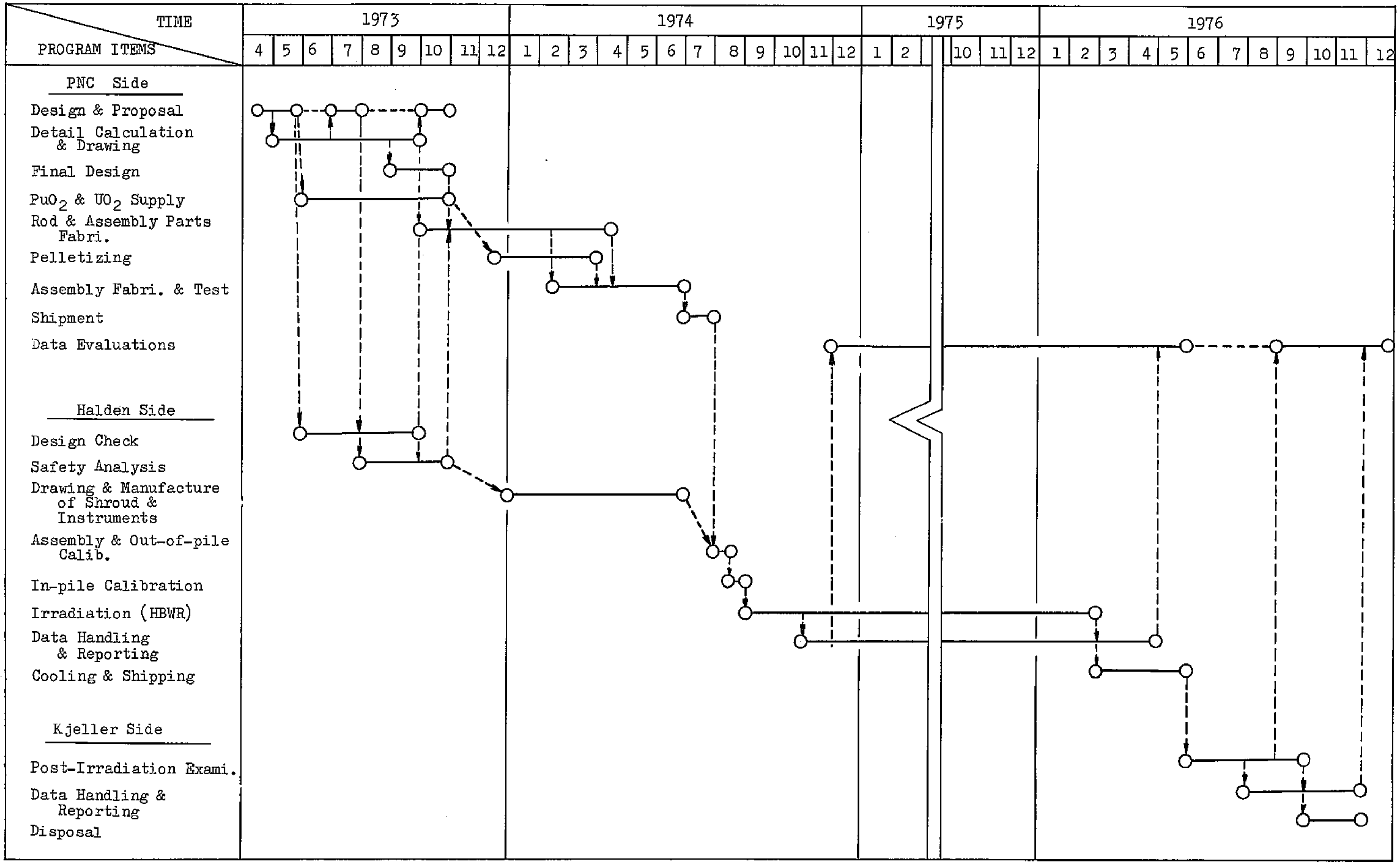
3) Examination on the cladding

a.	Hydrogen content in cladding	2~3 samples
b.	Neutron dosimetry (>1 MeV)	2~3 samples
c.	Burst test	2~3 samples
d.	Tensile test (Longitudinal and ring)	4~6 samples

7. Tentative Schedule

The proposed schedule of the irradiation program is shown in Fig. 7.1

Fig. 7.1 Proposed Schedule of Irradiation Program for IFA-423



8. References

1) "Programme Proposal for the Halden Reactor Project for the Three Year Period 1973-75" Institute for Atomenergi, Norway, February, 1971.

2) "The Irradiation of the Japanese Test Fuel Assemblies IFA-206 and IFA-207 in HBWR" V. Albergamo, EP-1207 (Ja), November, 1970.

3) "METHUSELAH-II A FORTRAN Program and Nuclear Data Library for the Physics Assessment of Liquid-Moderated Reactors." M. J. Brinkworth, J. A. Griffiths, AEEW-R480, 1966.

4) "The Post-Irradiation Examination of the Japanese Test Fuel Assemblies IFA-159 and IFA-160." K. D. Olshausen, P. Arnesen, P. Storvik, Report No. ME-46, March, 1971.

5) "COBRA-II; A Digital Computer Program for Thermal-Hydraulic Subchannel Analysis of Rod Bundle Nuclear Fuel Element" D. S. Rowe BNWL-1229 Feb., 1970.

6) "Burnout Limit Curves for Boiling Water Reactors" Janssen.E., and S. Levy, APED3892 (April 1962).

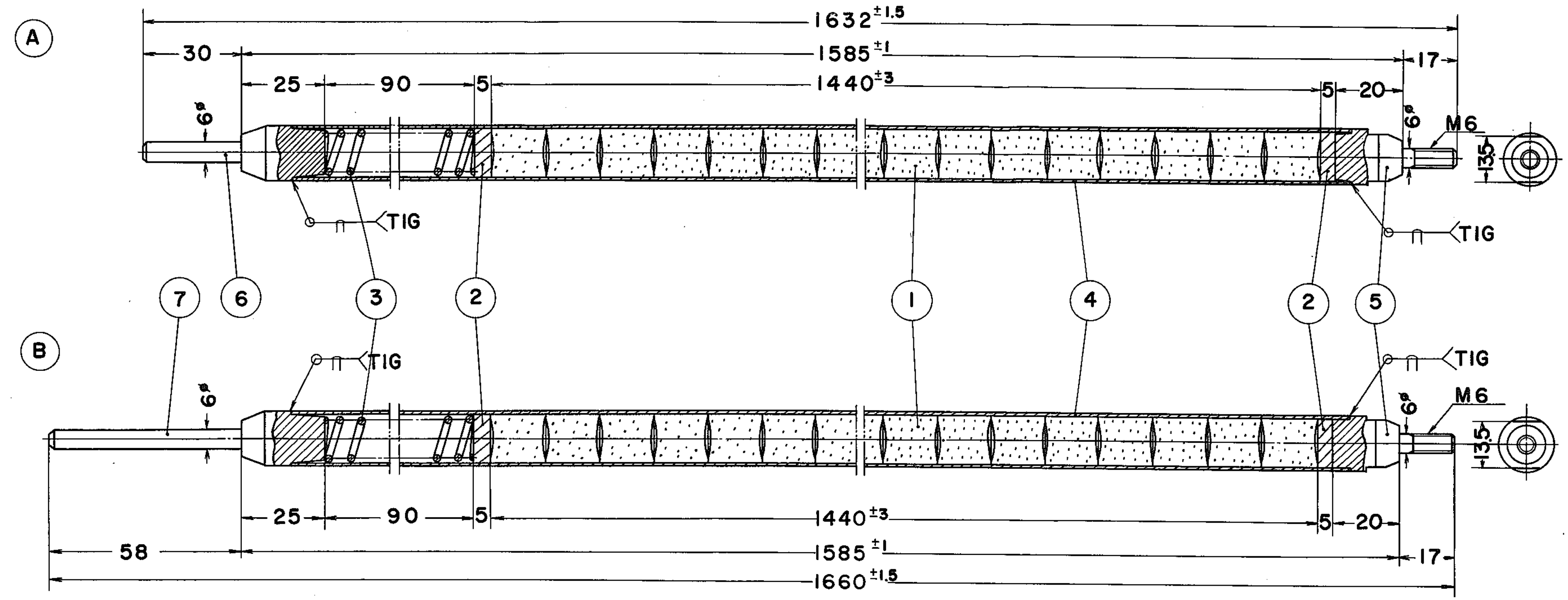
7) "UO_2 Pellet Thermal Conductivity From Irradiations with Central Melting" M. F. Lyons, GEAP-4624.

8) "ASME Boiler and Pressure Vessel Code section-III" July, 1971.

9. Drawing

Drawing List

Drawing No.	Title
EH3-200	FUEL ROD
EH3-300	FUEL ASSEMBLY
EH3-211	FUEL CLADDING
EH3-221	TOP END PLUG (1)
EH3-222	TOP END PLUG (2)
EH3-231	BOTTOM END PLUG
EH3-251	SPRING
EH3-310	SPACER
EH3-321	SPACER TIE ROD (UPPER)
EH3-322	SPACER TIE ROD (INTER-MEDIATE)
EH3-323	SPACER TIE ROD (LOWER)
EH3-331	TOP TIE PLATE
EH3-341	BOTTOM TIE PLATE
EH3-351	TOP SUPPORT
EH3-361	BOTTOM SUPPORT
EH3-371	SHROUD
EH3-411	NUT(1) FOR TIE ROD
EH3-412	NUT(2) FOR FUEL ROD
EH3-413	BOTTOM GUIDE ROD
EH3-414	GUIDE ROD NUT
EH3-421	SPRING WASHER(1) FOR TIE ROD
EH3-422	SPRING WASHER(2) FOR FUEL ROD
EH3-423	SPRING WASHER(3) FOR GUIDE ROD
EH3-372	SCREW FOR SHROUD



B	A	ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
3	/	7	Top End Plug (2)	Zry-2		EH3-222
/	4	6	Top End Plug (1)	〃		〃 -221
3	4	5	Bottom End Plug	〃		〃 -231
3	4	4	Cladding	Zry-2		〃 -211
3	4	3	Spring	Inconel-X		〃 -251
6	8	2	Thermal Insulator	ZrO_2		〃 -241
270	360	1	Fuel Pellet	PuO_2-UO_2		EH3-111

TITLE
IFA-423
FUEL ROD-(A)(B)

SCALE 1/1

ANGLE.

	DATE.
DESIGN T.K.	'73-8-
DWG. J. Hirota	'73-9-
CHECK R. Yumoto	'73-9-
APPR H. Akutsu	-9-27

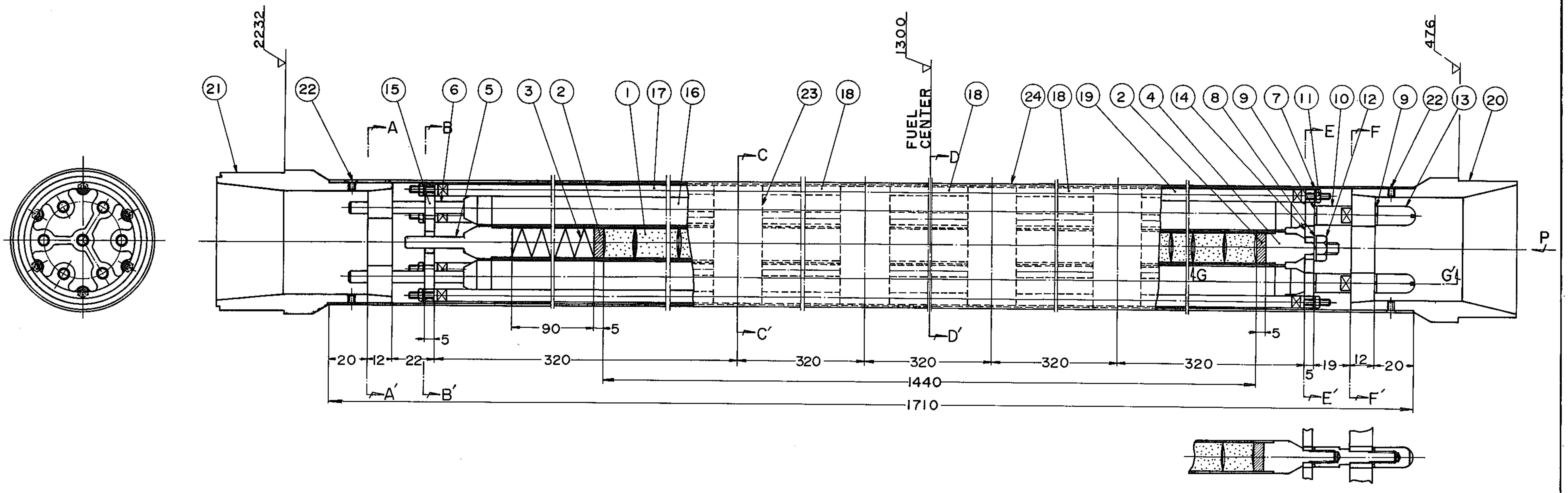
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT COPPORATION
TOKAI - JAPAN

DWG. NO
EH3-200

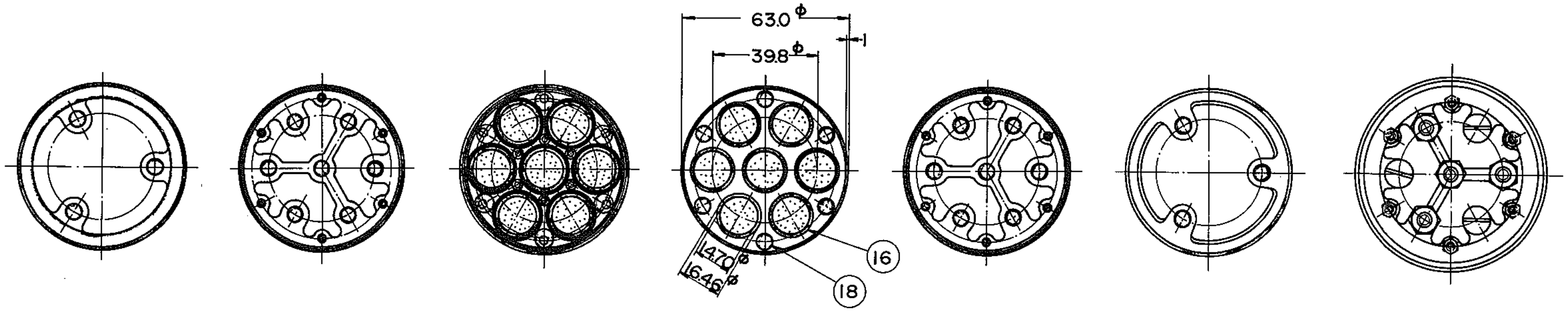
730518

REF. DWG

MARK	REVISION	DATE	NAME
△			
△			
△			



SECTION G-G′

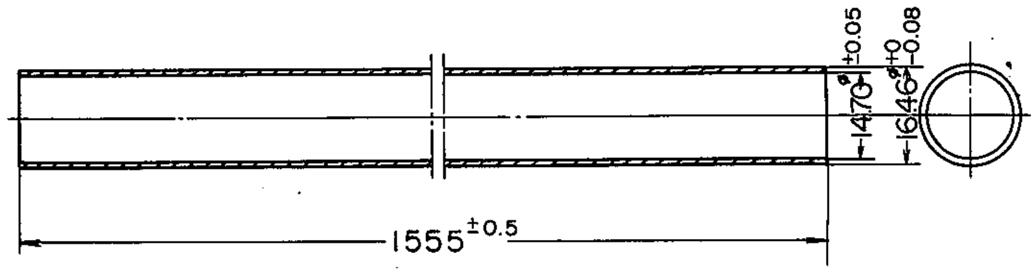


SECTION A-A′ SECTION B-B′ SECTION C-C′ SECTION D-D′ SECTION E-E′ SECTION F-F′ VIEW-P

ITEM	NAME	MATERIAL	SUPL.NO	REF. DWG
24	Shroud	Zry-2	1	EH3 - 371
23	Spacer	Inconel-718	4	〃 - 310
22	Screw	AISI 304	12	〃 - 372
21	Top Support	AISI 304	1	〃 - 351
20	Bottom Support	AISI 304	1	〃 - 361
19	Tie Rod (Lower)	Zry-2	6	〃 - 323
18	Tie Rod (Intermediate)	Zry-2	18	〃 - 322
17	Tie Rod (Upper)	Zry-2	6	〃 - 321
16	Fuel Cladding	Zry-2	7	〃 - 211
15	Top Tie Plate	AISI 304	1	〃 - 331
14	Bottom Tie Plate	AISI 304	1	〃 - 341
13	Guide Rod Nut	AISI 304	3	〃 - 414
12	Nut (2)	AISI 304	4	〃 - 412
11	Nut (1)	AISI 304	12	〃 - 411
10	Bottom Guide Rod	AISI 304	3	〃 - 413
9	Spring Washer (3)	AISI 304	6	〃 - 423
8	Spring Washer (2)	AISI 304	4	〃 - 422
7	Spring Washer (1)	AISI 304	12	〃 - 421
6	Top End Plug (2)	Zry-2	3	〃 - 222
5	Top End Plug (1)	Zry-2	4	〃 - 221
4	Bottom End Plug	Zry-2	7	〃 - 231
3	Spring	Inconel-X	7	〃 - 251
2	Thermal Insulator	ZrO_2	14	〃 - 241
1	Fuel Pellet	PuO_2-UO_2	630	EH3 - 111

ANGLE	SCALE		DESIGN	DWG.	CHECK	APPR.
Third	1/1 ( )	NAME	T. Kajiyama	J. Hirata	R. Yumoto	H. Akutsu
		DATE	'73-8-	'73-9-	'73-9-	'73-9-

CLASSIFICATION	TITLE
FUGEN	IFA-423 FUEL ASSEMBLY
CODE NO.	DRAWING NO. EH3-300
	730519



ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Fuel Cladding	Zry - 2	7	

TITLE

IFA-423
FUEL CLADDING

	NAME	DATE
DESIGN	T. Kajiyama	'73 - 8 -
DWG.	J. Hirata	'73 - 9 -
CHECK	R. Yumoto	'73 - 9 -
APPR.	N. Akutsu	- 9-27

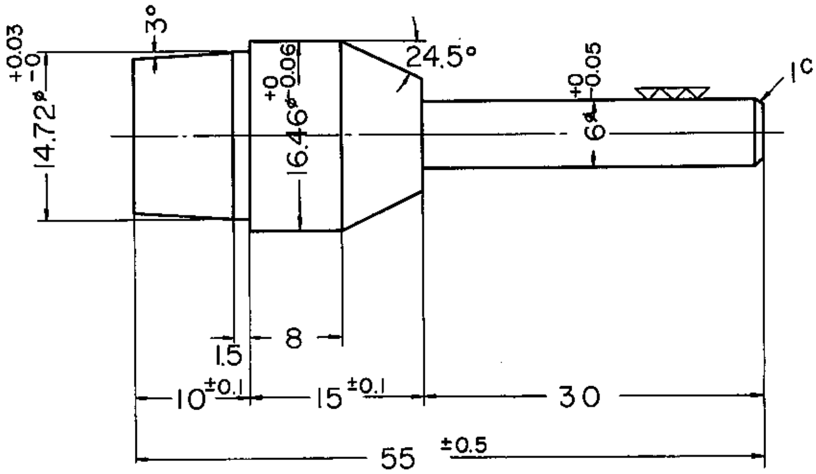
POWER REACTOR AND NUCLEAR FUEL
DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 1/1

ANGLE

DRAWING NO.
EH3-211

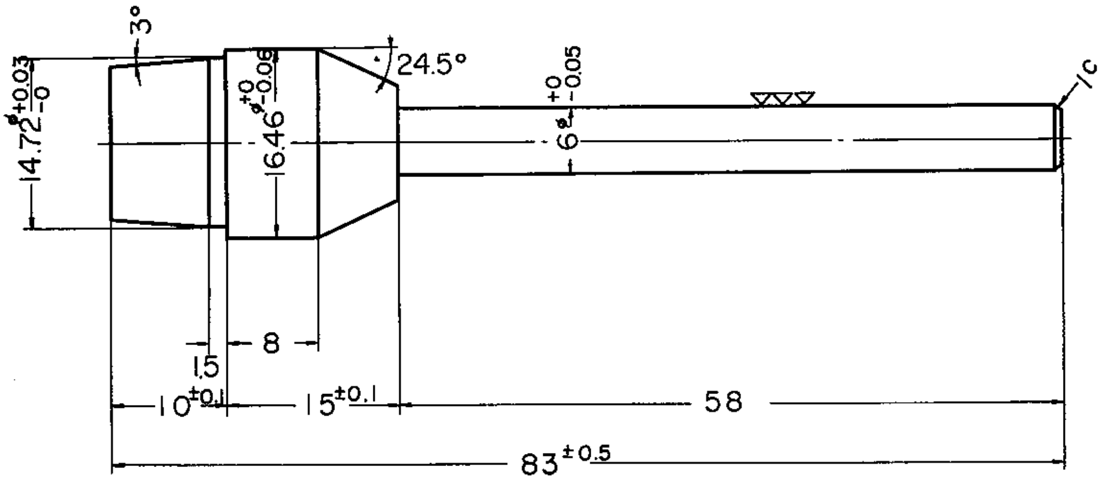
730521



ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Top End Plug (1)	Zry - 2	4	

TITLE		NAME	DATE
IFA-423 TOP END PLUG (I)	DESIGN	T. Kajiyama	'73 - 8 -
	DWG.	T. Hirata	'73 - 9 -
	CHECK	R. Yumoto	'73 - 9 -
	APPR.	[illegible]	- 9 - 27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION TOKAI - JAPAN	SCALE 2/1 ANGLE	DRAWING NO. EH3-221 730522



1	Top End Plug (2)	Zry -2	3	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE

IFA-423 TOP END PLUG (2)

	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	J. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	Id. Akutsu	-9-27

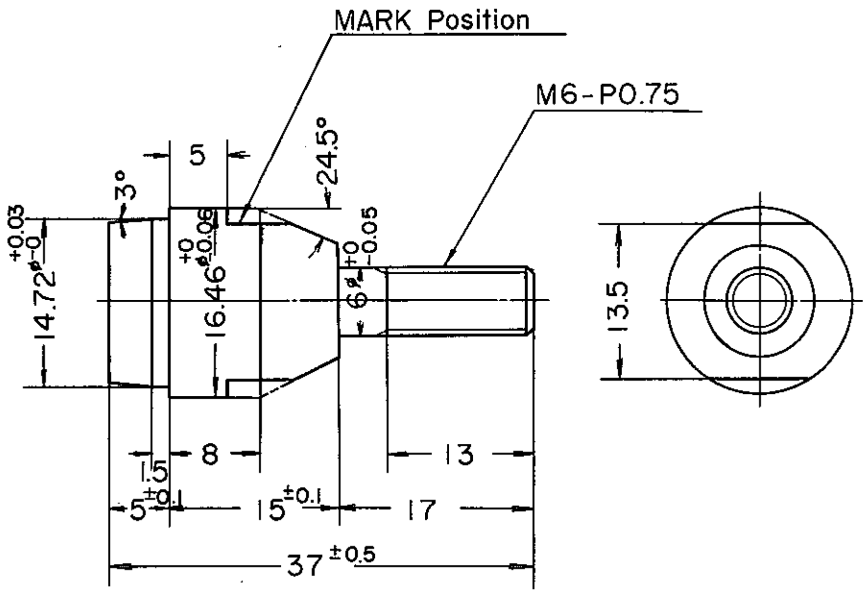
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 2/1

ANGLE

DRAWING NO. EH3-222

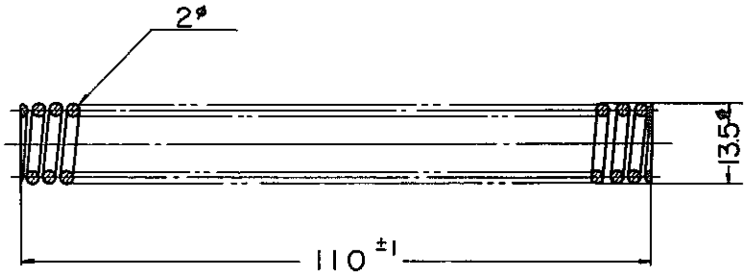
730523



1	Bottom End Plug	Zry-2	7	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE		NAME	DATE
IFA-423 BOTTOM END PLUG	DESIGN	T.Kajiyama	'73-8-
	DWG.	J. Hirata	'73-9-
	CHECK	R. Yumoto	'73-9-
	APPR.	N. Akutsu	-9-27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION TOKAI - JAPAN	SCALE 2/1	DRAWING NO. EH3-231
	ANGLE	730524



SPECIFICATION

MATERIAL	Inconel - X
WIRE DIAMETER	2.0 mm $^{\pm 0.05}$
OUTER DIAMETER OF COIL	13.5mm $^{\pm 0.2}$
TOTAL NUMBER OF TURNS	37
EFFECTIVE NUMBER OF TURNS	35
FREE LENGTH	110mm
TEST LOAD	4.88 kg
LENGTH OF SPRING AT TEST LOAD	93mm
STRESS OF SPRING AT TEST LOAD	22.5kg/mm^2
SPRING CONSTANT	0.29kg/mm ±1%

ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Spring	Inconel - X	7	

TITLE

IFA-423 SPRING

	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	J. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	H. Akutsu	-9-27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

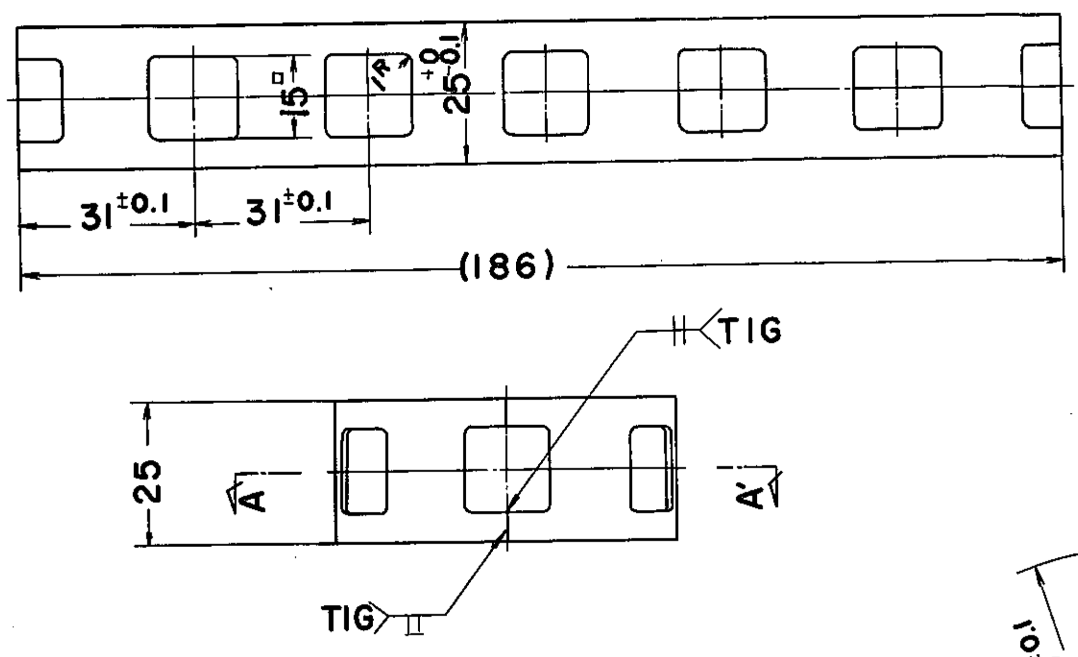
SCALE 1/1

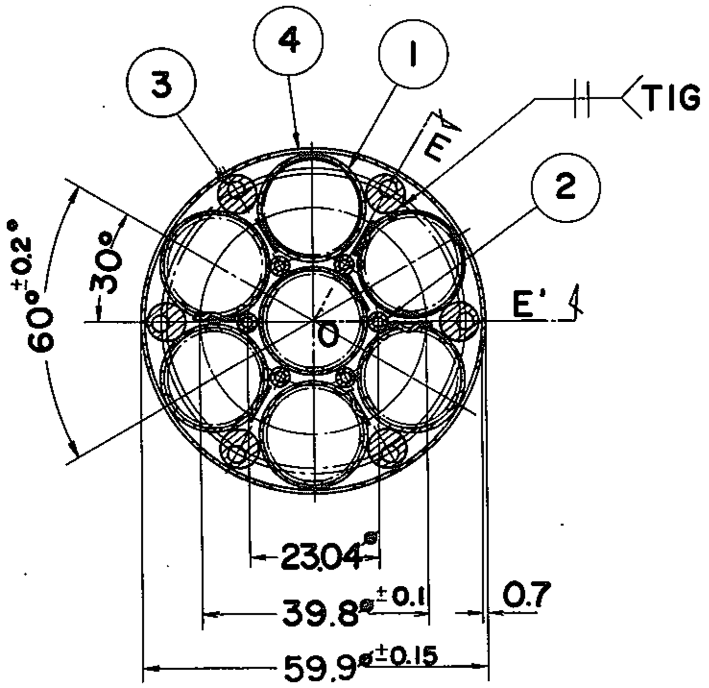
ANGLE

DRAWING NO. EH3-251

730526

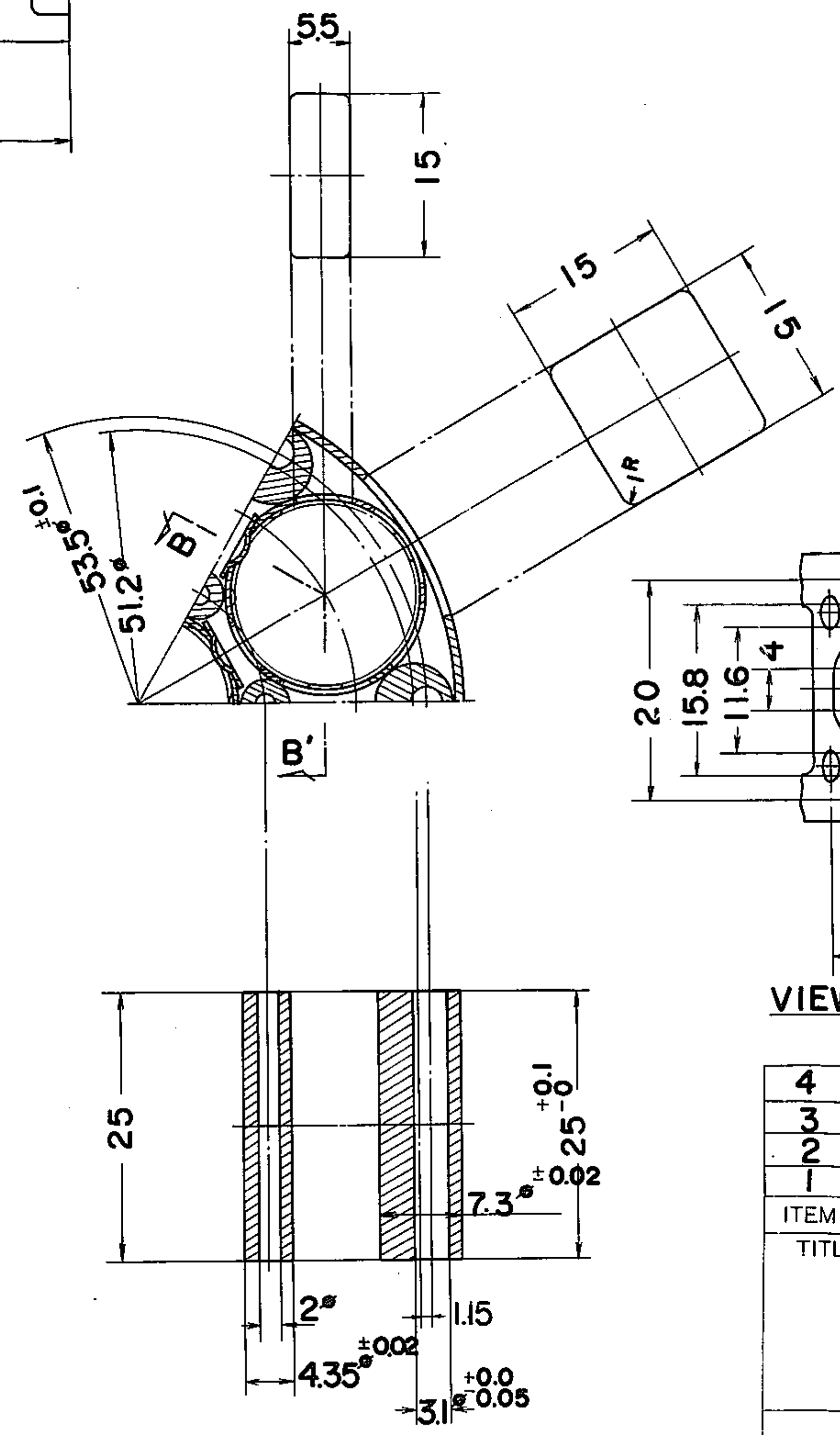
SPACER BAND DEVELOPMENT



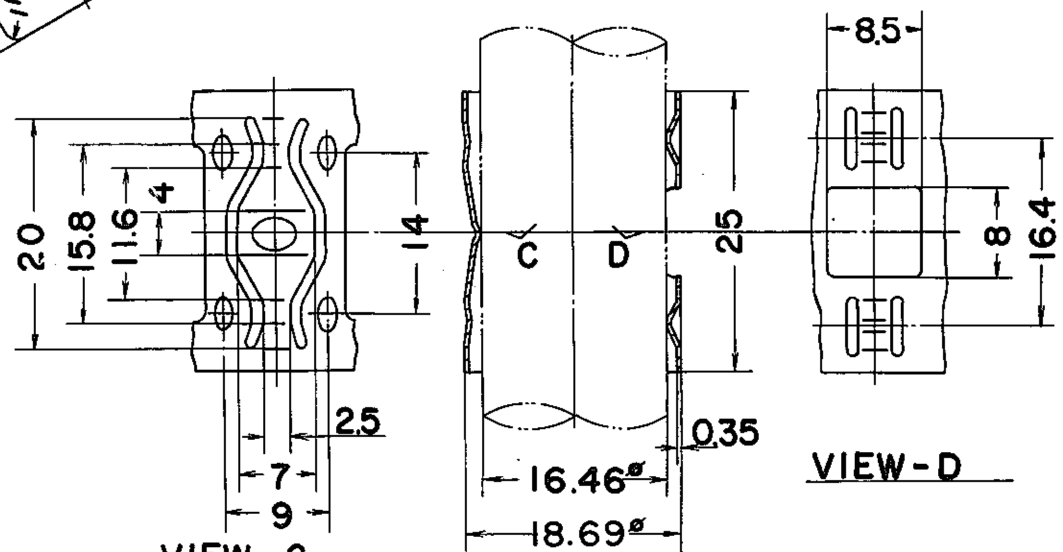


A-A' SECTION

DETAIL E-O-E' (2/1)



B-B' SECTION (2/1)



VIEW-C

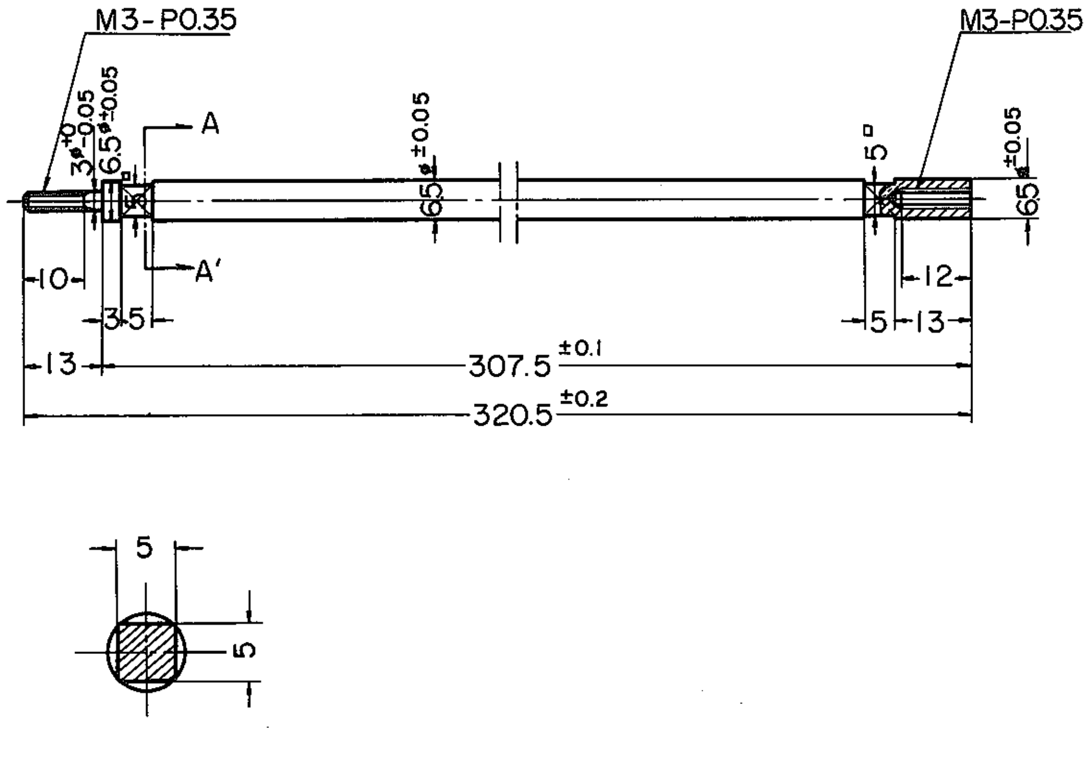
VIEW-D

ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
4	Band	Inconel-718	4×1	
3	Tie Rod Support	〃	4×6	
2	Support Tube	〃	4×6	
1	Rod Support Ring	Inconel-718	4×7	EH3-311

TITLE	SCALE		DATE.
IFA-423 SPACER	1/1 (2/1)	DESIGN T.K.	'73-8-
	ANGLE.	DWG. J. Hirata	'73-8-3
		CHECK R. Yumoto	'73-9-
		APPR. H. Akutsu	'73-9-27
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT COPPORATION TOKAI - JAPAN	DWG. NO EH3-310		

730527

Note: 1 Remove all burrs and sharp edges



SECTION-A-A'(2/1)

ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Tie Rod	Zry-2	6	

TITLE

IFA-423 SPACER TIE ROD (UPPER)

	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	J. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	N. Akutsu	-9-27

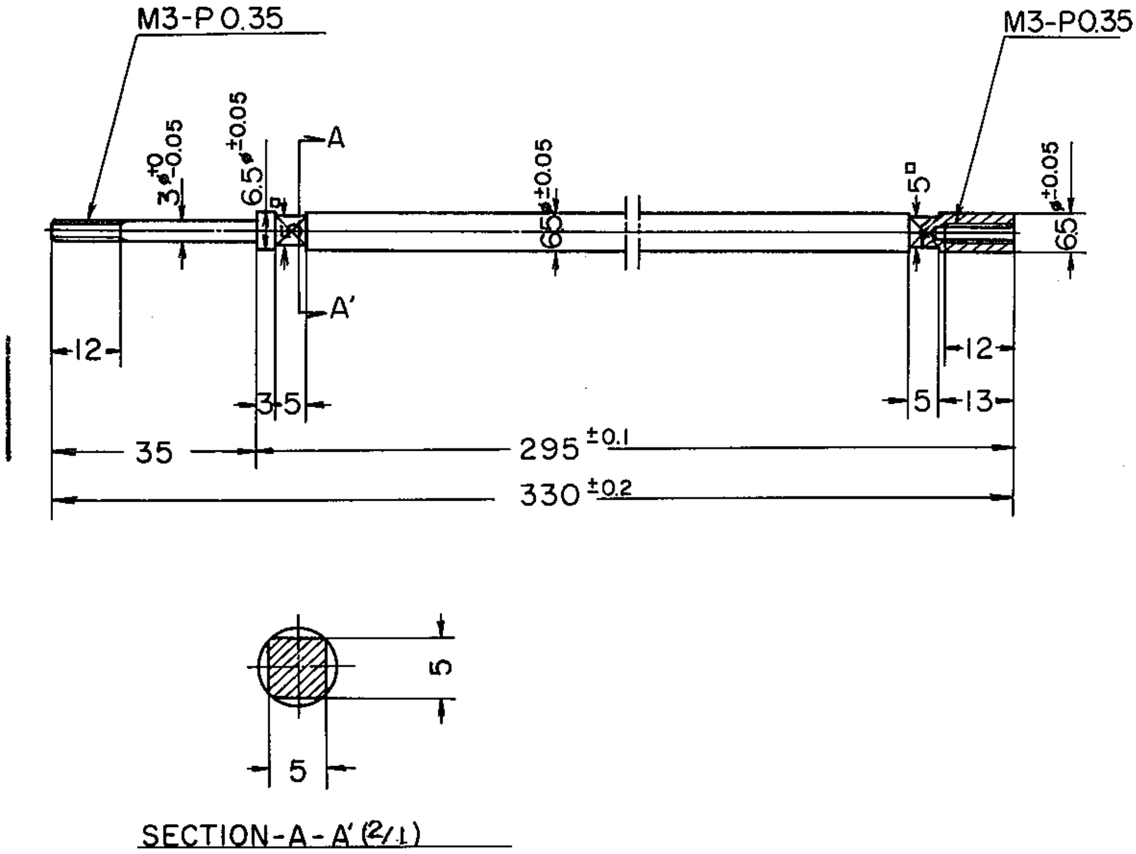
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 1/1 (2/1)

ANGLE

DRAWING NO. EH3-321

730528



1	Tie Rod	Zry-2	18	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

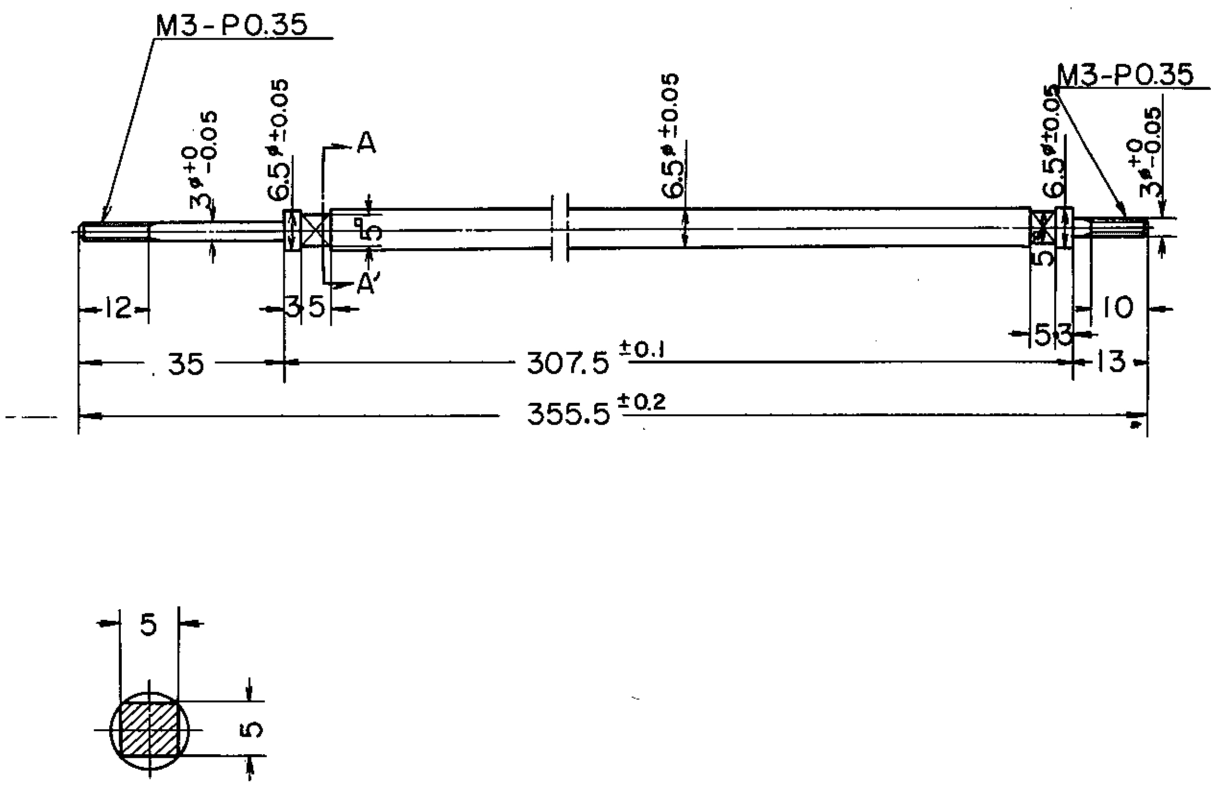
TITLE

IFA-423
SPACER TIE ROD
(INTERMEDIATE)

	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	J. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	N. Akutsu	-9-27

POWER REACTOR AND NUCLEAR FUEL
DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE	DRAWING NO.
1/1 (2/1)	EH3-322
ANGLE	730529



SECTION - A-A' (2/1)

1	Tie Rod	Zry - 2	6	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE
IFA-423
SPACER TIE ROD
(LOWER)

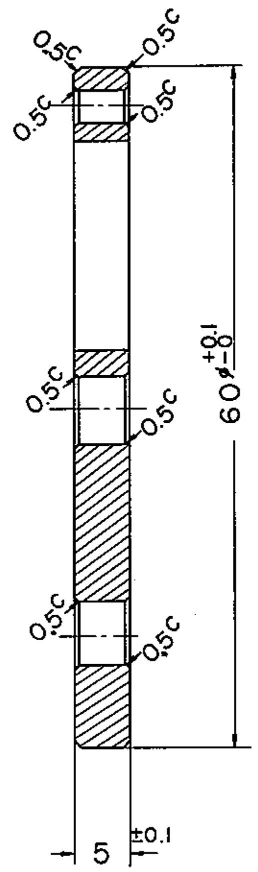
	NAME	DATE
DESIGN	T. Kajiyama	'73- 8 -
DWG.	J. Hirata	'73- 9 -
CHECK	R. Yumoto	'73- 9 -
APPR.	N. Akutsu	- 9-27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

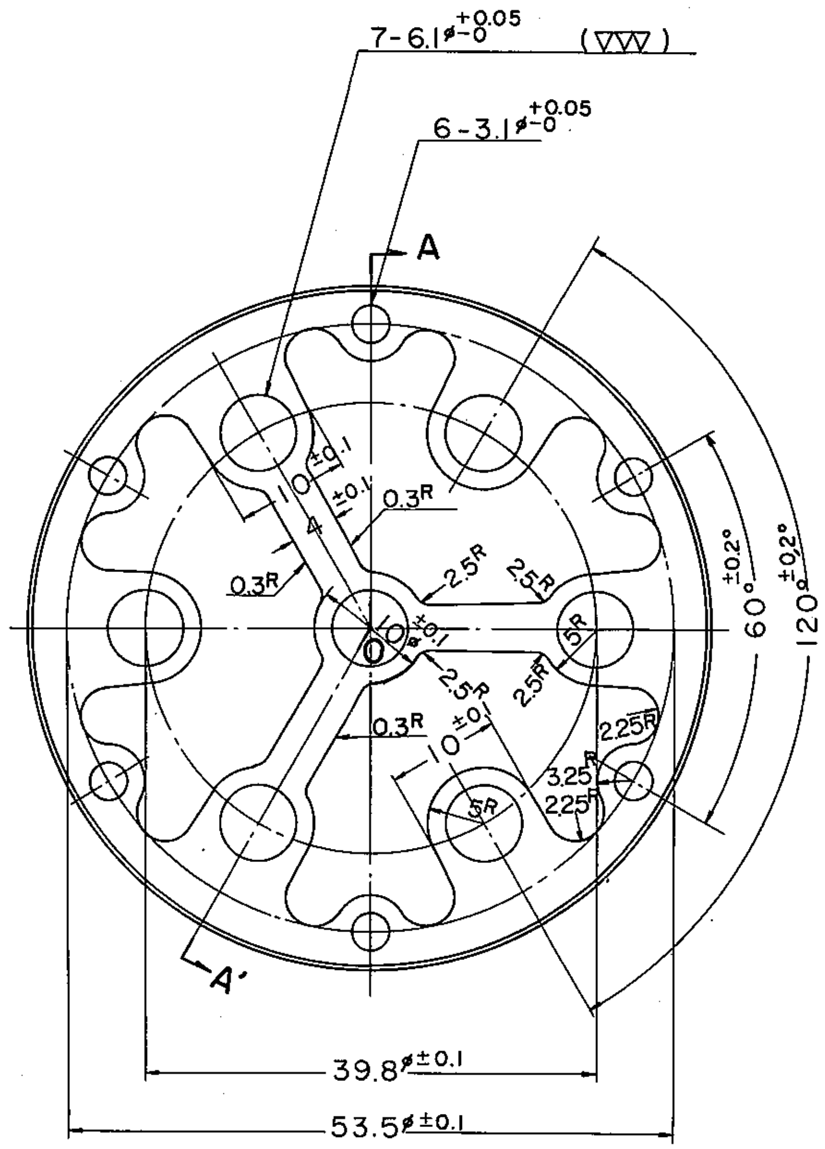
SCALE 1/1 (2/1)
ANGLE

DRAWING NO. EH3-323

730530



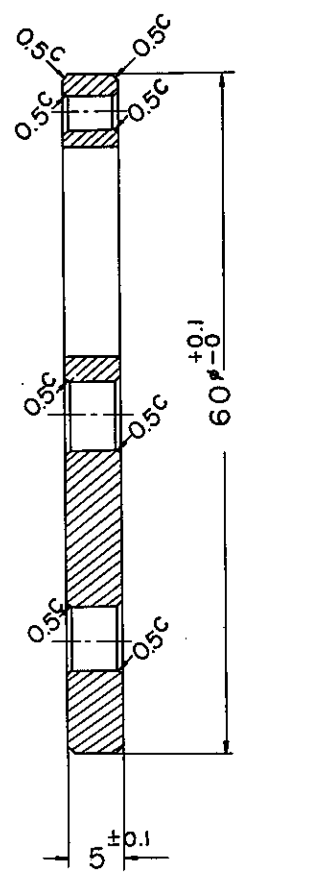
SECTION A-O-A'



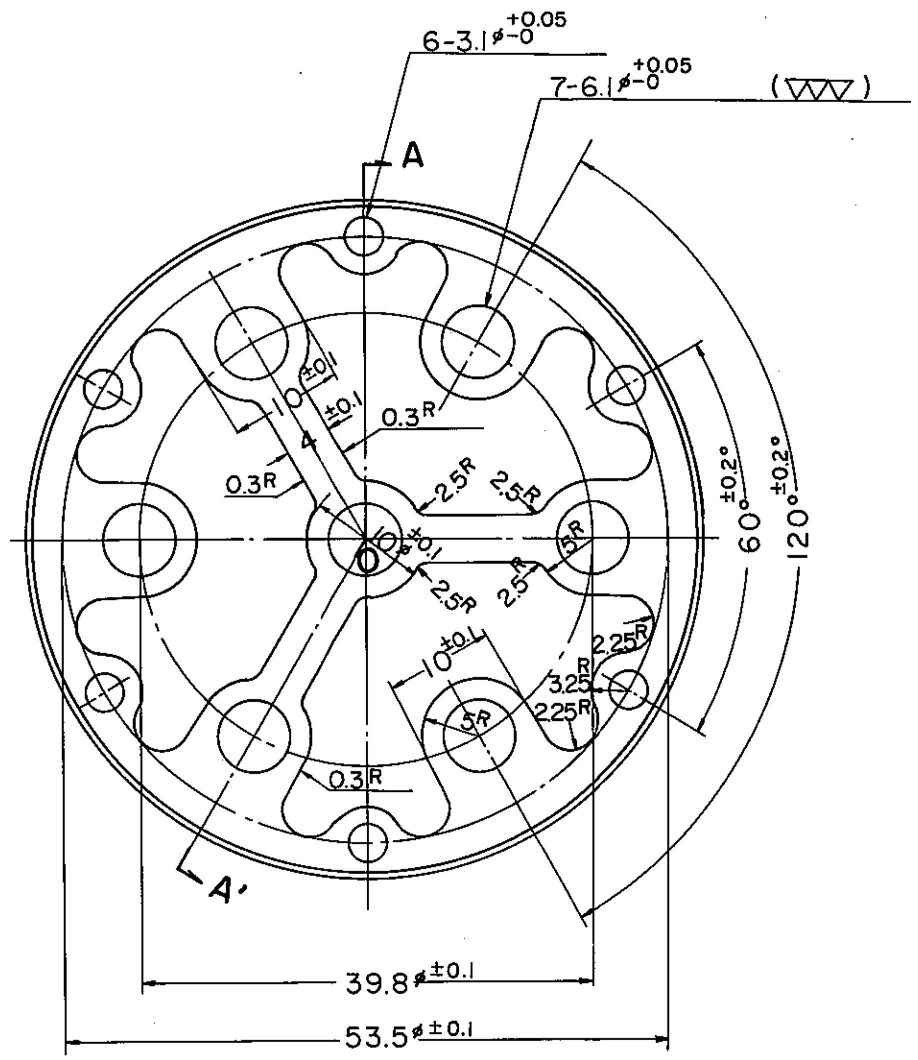
Note: 1 The sharp edge of both ends round off 0.3R.

I	Top Tie Plate	AISI304	I	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE IFA-423 TOP TIE PLATE	SCALE 2/1 ANGLE.		DATE.
		DESIGN T. K.	'73-8-
		DWG. T. Hirata	'73-9-
		CHECK R. Y	- -
		APPR. N. [signature]	9-27
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT COPPORATION TOKAI - JAPAN	DWG. NO EH3-331 730531		



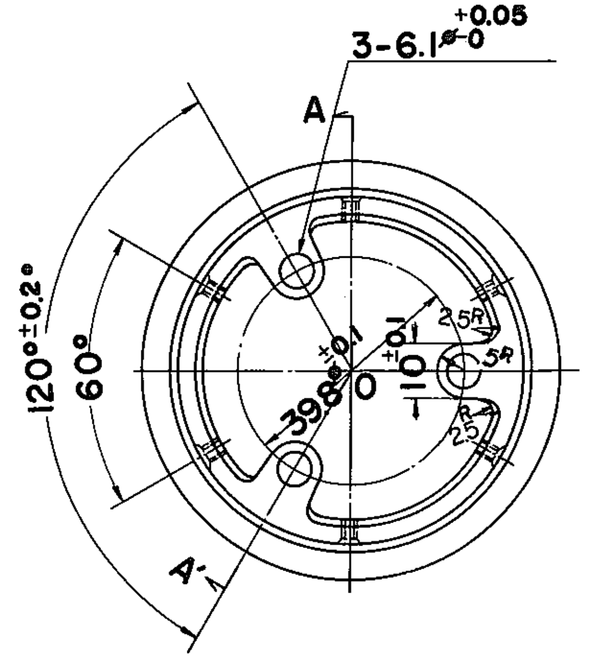
SECTION A-O-A′

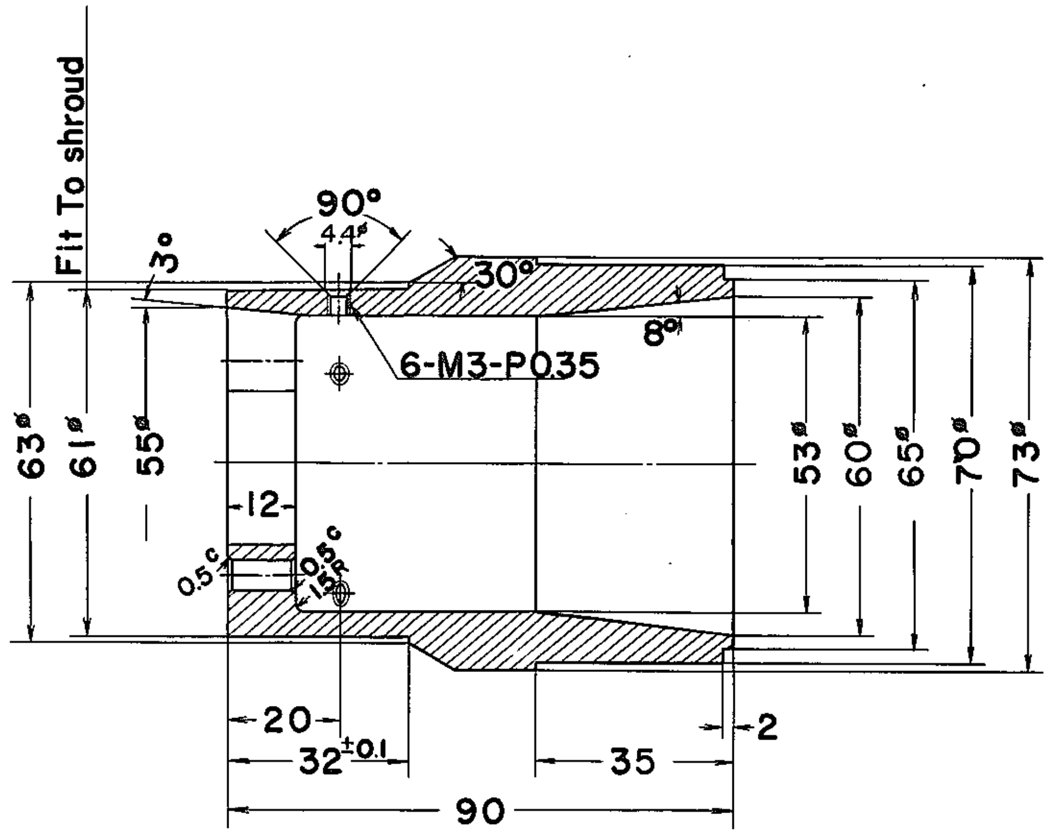


Note: 1 The Sharp edge of both ends round off 0.3R.

ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Bottom Tie Plate	AISI304	1	

TITLE IFA-423 BOTTOM TIE PLATE	SCALE 2/1		DATE.
		DESIGN T.K.	'73-8-
	ANGLE.	DWG. T. Hirota	'73-9-
		CHECK R.Y	– –
		APPR'D [signature]	-9-27
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT COPPORATION TOKAI - JAPAN	DWG. NO EH3-341 730532		



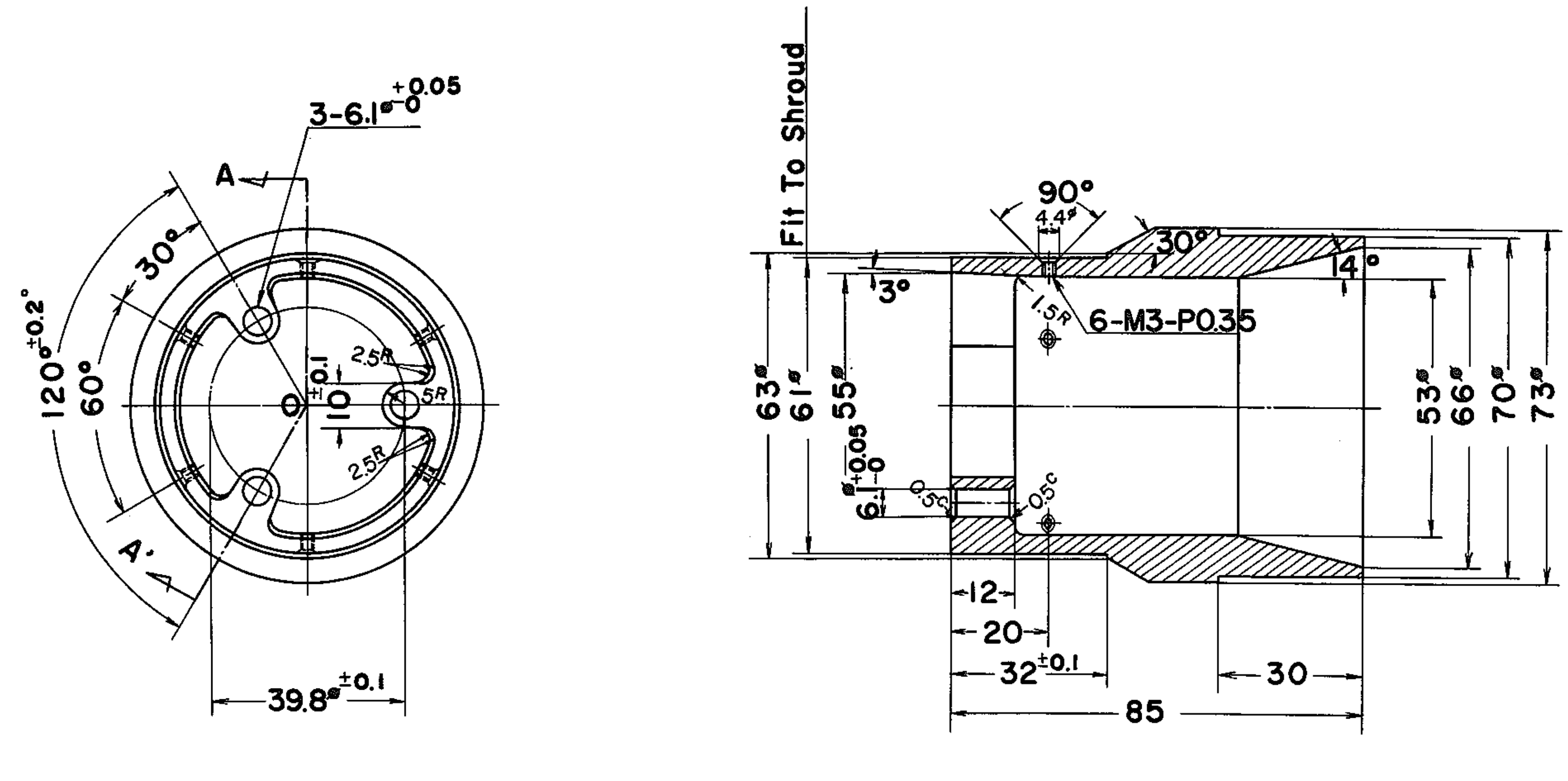


SECTION A-O-A'

Note : 1 Tapered holes of shroud to be fitted to those of Top support.

1	Top Support	AISI 304	1	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE IFA-423 TOP SUPPORT	SCALE 1/1 ANGLE.		DATE.
		DESIGN T.K.	'73-8-
		DWG. J.Hirota	'73-9-
		CHECK R.Y	'73-9-
		APPR. H.Akutsu	9-27
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT COPPORATION TOKAI - JAPAN	DWG. NO EH3-351 730533		



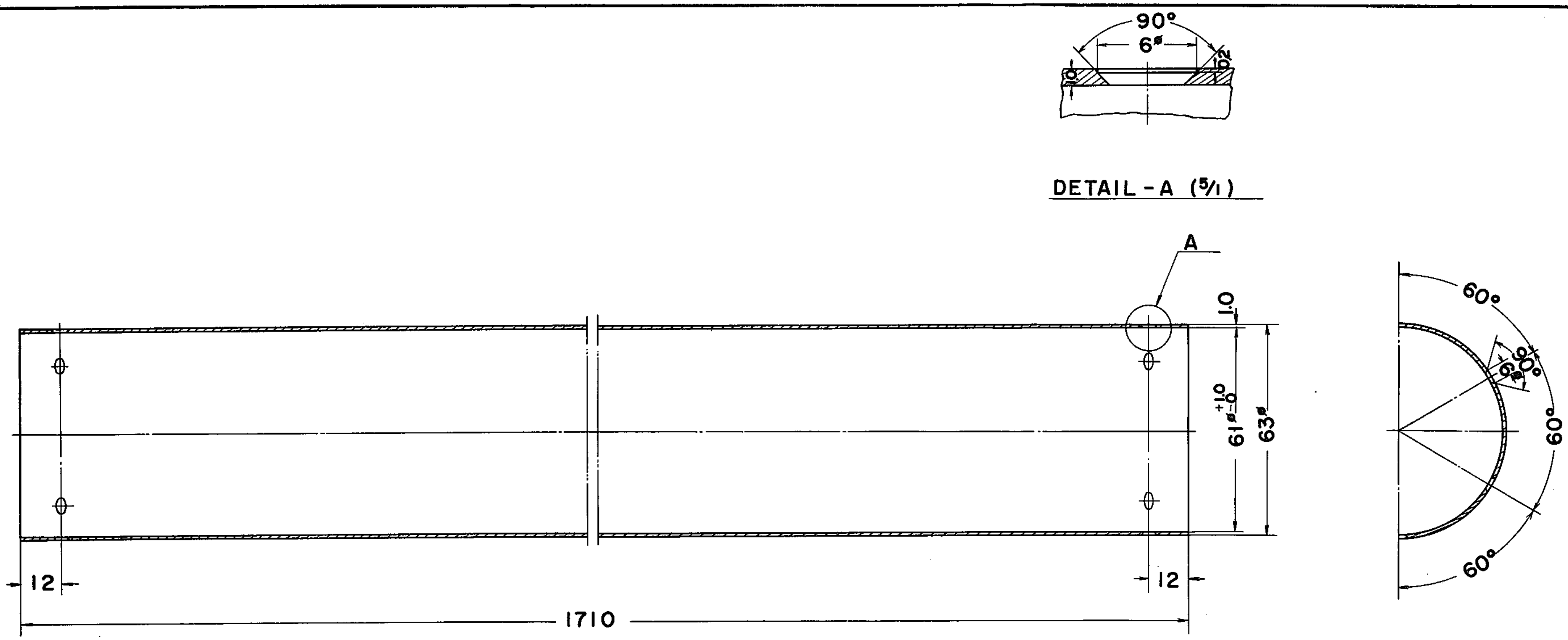
SECTION A-O-A'

Note :1 Tapered holes of shroud to be fitted to those of Bottom support.

ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Bottom Support	AISI 304	1	

TITLE	SCALE		DATE.
IFA-423 BOTTOM SUPPORT	1/1	DESIGN T.K.	'73-8-
		DWG. J. Hirata	'73-9-
	ANGLE.	CHECK R. Yamada	'73-9-
		APPR. [illegible]	-9-27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT COPPORATION TOKAI - JAPAN	DWG. NO EH3-361 730534



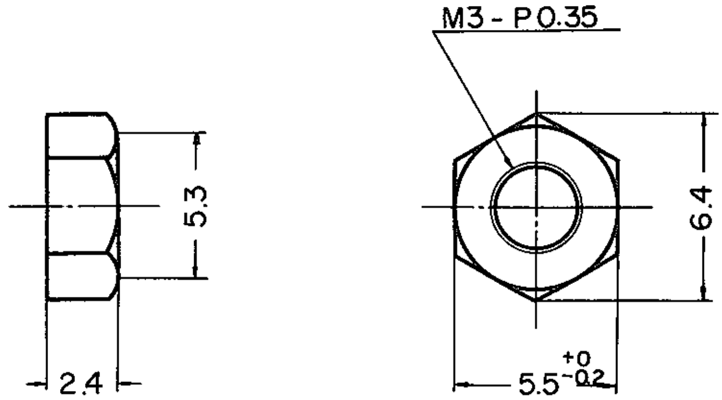
1	Shroud	Zry-2	1	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE
IFA-423
SHROUD

SCALE		DATE.
1/1 (5/1)	DESIGN T.K.	'73-8-
	DWG. J. Hirota	'73-9-
ANGLE.	CHECK R. Yumi	'73-9-
	APPR.	-9-27

POWER REACTOR AND NUCLEAR FUEL
DEVELOPMENT COPPORATION
TOKAI - JAPAN

DWG. NO
EH3-371
730535



ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Nut (1)	AISI 304	12	

TITLE
IFA-423
NUT (1) FOR TIE ROD

	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	J. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	H. [illegible]	-9-27

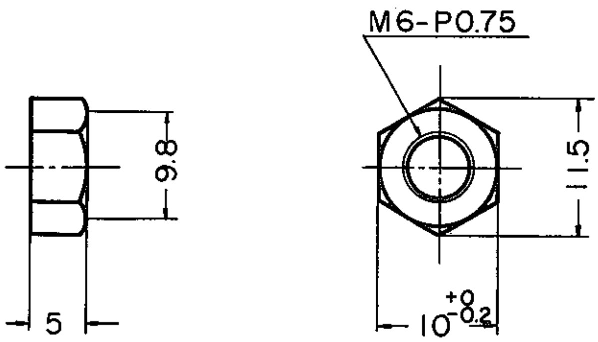
POWER REACTOR AND NUCLEAR FUEL
DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 5/1

ANGLE

DRAWING NO. EH3-411

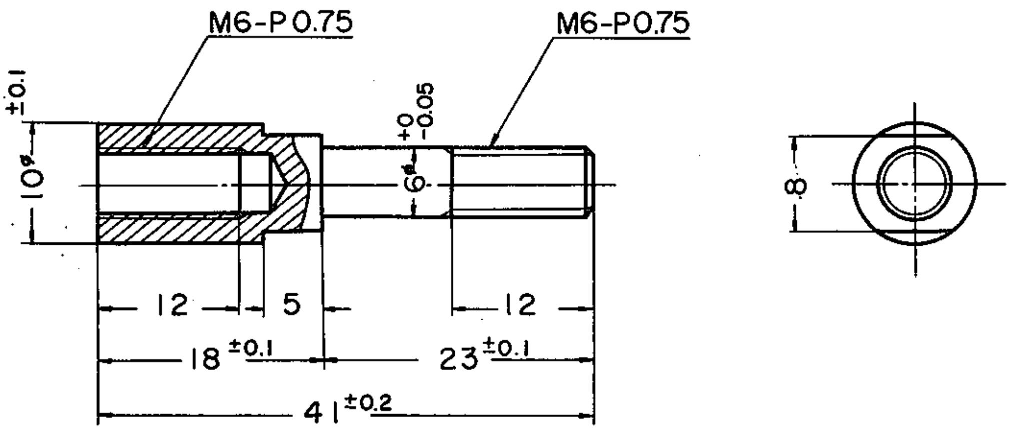
730536



ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Nut (2)	AISI 304	4	

TITLE		NAME	DATE
IFA-423	DESIGN	T. Kajiyama	'73-8 -
NUT(2) FOR FUEL ROD	DWG.	J. Hirata	'73-9 -
	CHECK	R. Yumoto	'73-9 -
	APPR.	H. Akutsu	- 9 -27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION TOKAI - JAPAN	SCALE 2/1 ANGLE	DRAWING NO. EH3-412 730537



ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Bottom Guide Rod	AISI-304	3	

TITLE **IFA-423**

BOTTOM GUIDE ROD

	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	J. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	H. Akutsu	-9-27

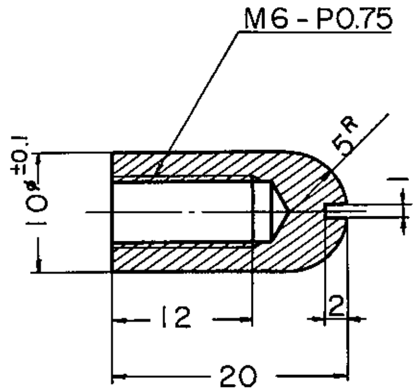
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 2/1

ANGLE

DRAWING NO. **EH3-413**

730538



ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Guide Rod Nut	AISI 304	3	

TITLE
IFA-423
GUIDE ROD NUT

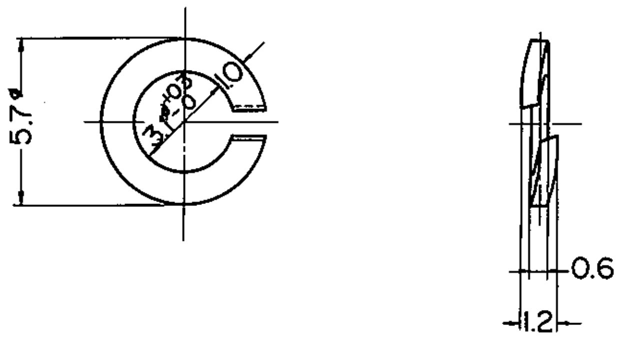
	NAME	DATE
DESIGN	T. Kajiyama	'73- 8 -
DWG.	J. Hirata	'73- 9 -
CHECK	R. Yumoto	'73- 9 -
APPR.	H. [illegible]	- 9 -27

POWER REACTOR AND NUCLEAR FUEL
DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 2/1

ANGLE

DRAWING NO.
EH3-414 730539



1	Spring Washer (1)	AISI 304	12	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE

IFA-423
SPRING WASHER (1)
FOR TIE ROD

	NAME	DATE
DESIGN	T. Kajiyama	'73-8 -
DWG.	J. Hirata	'73-9 -
CHECK	R. Yumoto	'73-9 -
APPR.	H. Akutsu	- 9 -27

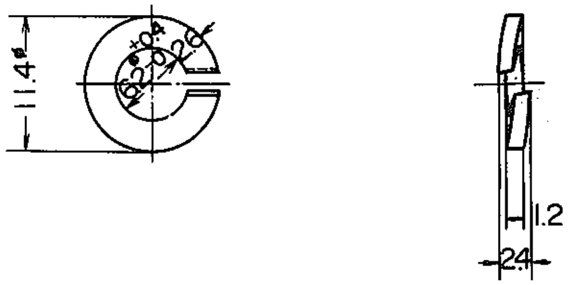
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 5/1

ANGLE

DRAWING NO. EH 3-421

730540



1	Spring Washer (2)	AISI 304	4	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE

IFA-423
SPRING WASHER (2)
FOR FUEL ROD

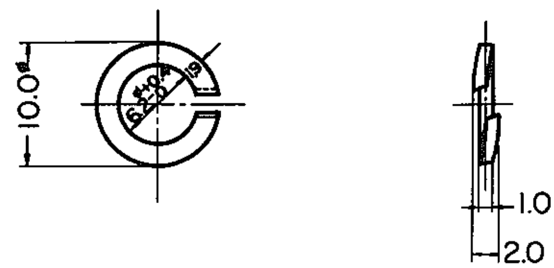
	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	I. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.		-9-27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 2/1

ANGLE

DRAWING NO.
EH3-422
730541



	Spring Washer (3)	AISI 304	6	
ITEM	NAME	MATERIAL	SUPL. NO	REMARKS

TITLE

IFA-423
SPRING WASHER (3) FOR GUIDE ROD

	NAME	DATE
DESIGN	T.Kajiyama	'73-8-
DWG.	T. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	H Akutsu	-9-27

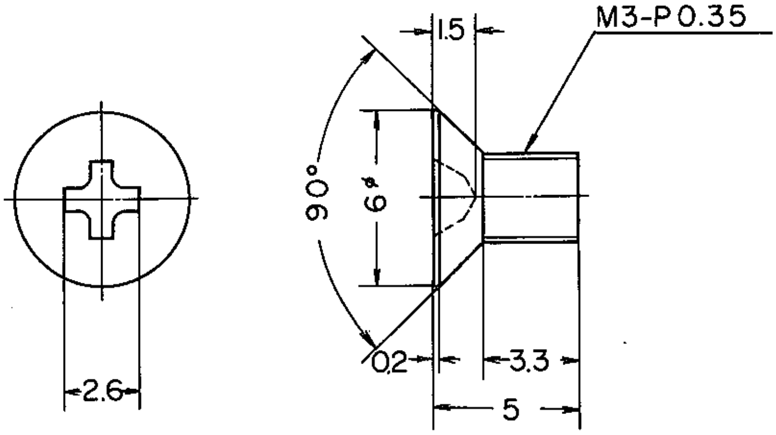
POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 2/1

ANGLE

DRAWING NO.
EH3-423

730542



ITEM	NAME	MATERIAL	SUPL. NO	REMARKS
1	Screw	AISI 304	12	

TITLE

IFA-423
SCREW

	NAME	DATE
DESIGN	T. Kajiyama	'73-8-
DWG.	T. Hirata	'73-9-
CHECK	R. Yumoto	'73-9-
APPR.	H. [illegible]	-9-27

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION
TOKAI - JAPAN

SCALE 5/1

ANGLE

DRAWING NO. EH3-372

730543