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THERMAL-HYDRAULIC ANALYSES OF THE JMTR AND THE JRR-2 WITH LEU FUELS

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The thermal-hydraulic analyses of the JMTR and the JRR-2 with LEU (low-enriched uranium) fuels were performed as part of analyses for converting the JMTR and the JRR-2 to utilize LEU fuels. The thermal-hydraulic calculation code used in these analyses is COBRA-3C/RERTR which was developed for the RERTR (reduced enrichment research and test reactor) program by ANL. It was found that the JMTR and the JRR-2 with LEU fuels would be able to retain nearly the same margin to ONB (onset of nucleate boiling) and DNB (departure from nucleate boiling) without changing the present pump systems as that of the present HEU fuel cores.

These analyses were performed by the author at ANL as part of the JAERI-ANL joint study on the use of reduced enrichment fuels in the JAERI research reactors.

Keywords : RERTR Program, LEU Fuel, Reduced Enrichment, Onset of Nucleate Boiling, Margin to ONB, Margin to DNB, Thermal-hydraulic Analysis, JMTR, JRR-2.

ＪＭＴＲ及びＪＲＲ－２の低濃縮燃料炉心の熱水力的検討

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ＪＭＴＲ及びＪＲＲ－２の炉心を低濃縮燃料に転換するための検討の一環として，これら低濃縮炉心に対する熱水力的検討を行なった。使用した熱水力計算コードはＡＮＬがＲＥＲＴＲプログラム（研究炉用燃料の濃縮度低減化プログラム）のために開発したＣＯＢＲＡ－３Ｃ／ＲＥＲＴＲである。ＪＭＴＲ及びＪＲＲ－２の低濃縮燃料炉心は，現在のポンプシステムを交換しなくても，現行炉心と同程度のＯＮＢ（核沸騰開始点における熱流束）及びＤＮＢ（限界熱流束）に対する余裕度を持ち得るとの検討結果を得た。

なお本検討は，日本原子力研究所の研究炉用燃料の濃縮度低減化に関するＪＡＥＲＩ－ＡＮＬ共同研究の一環として，著者がＡＮＬにおいて行ったものである。

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1. Introduction

In view of the proliferation resistance of fuels and fuel cycles, the use of HEU (high-enriched uranium) fuels in research and test reactors was discussed at INFCE and IAEA, and the enrichment reduction in research and test reactors was proposed. Following this trend of times, JAERI is implementing the core conversion of JAERI reactors to utilize MEU (medium-enriched uranium)fuels. A joint study has been in progress between JAERI and ANL since January, 1980, to assess the feasibility of converting the JAERI reactors to use of reduced enrichment fuels. Table 1 shows the outline of the JAERI-ANL joint study.

The enrichment reduction in research and test reactors is based on the practical criterion that enrichment reduction should not cause reduction of significant flux performance and burnup performance relative to the HEU core. This requires that the reduced enrichment fuel element contains more ^{235}U than the high enrichment fuel element. In the case of the LEU (<20%) core, the LEU fuel element should have thicker fuel meat than the HEU fuel element based on the present fuel technology, so that the LEU core might keep the LEU core performances. For this reason, the LEU core have a trend toward smaller thermal safety margin compared with the HEU core, and thermal-hydraulic analyses are important as well as neutronic analyses in the feasibility studies on the use of LEU fuels.

As can be seen in Table 1, the Phase A, which was the first stage of the joint study, ended in July, 1980. The thermal-hydraulic analyses were performed preliminary in the Phase A, but the additional details of thermal-hydraulic data about the JMTR and the JRR-2 with LEU fuels are necessary in order to assess the feasibility of the use LEU fuels in the JMTR and JRR-2. This report describes the results of thermal-hydraulic analyses of the JMTR and the JRR-2 performed by using a thermal-hydraulic calculation code, COBRA-3C/REFTR, as part of the joint study. This code was written for the RERTR program by ANL.

2. Thermal-Hydraulic Code COBRA-3C/RERTR

COBRA-3C/RERTR is a thermal-hydraulic subchannel code prepared for the RERTR program centered at ANL. This code was developed by modifying COBRA-3C/MIT, and by extending its applicability to research and test reactors which operate at low pressure and temperature, and which may use plate-type fuel elements. COBRA-3C/MIT is an advanced version of COBRA-IIIC,

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As can be seen in Table 1, the Phase A, which was the first stage of the joint study, ended in July, 1980. The thermal-hydraulic analyses were performed preliminary in the Phase A, but the additional details of thermal-hydraulic data about the JMTR and the JRR-2 with LEU fuels are necessary in order to assess the feasibility of the use LEU fuels in the JMTR and JRR-2. This report describes the results of thermal-hydraulic analyses of the JMTR and the JRR-2 performed by using a thermal-hydraulic calculation code, COBRA-3C/REFTR, as part of the joint study. This code was written for the RERTR program by ANL.

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which has been widely used and accepted by the power industry.

A number of other similar codes exist for subchannel analysis, such as COBRA-IIIC, MACABRE, THINC, and TORC. But most were developed for high-power commercial reactors. Therefore, the heat transfer correlations, critical heat flux correlations, and boiling flow regimes considered in these codes are not adequate for analysing research and test reactors.

3. Check Calculation of COBRA-3C/RERTR

In order to examine whether COBRA-3C/RERTR is suitable for thermal-hydraulic calculations of JAERI reactors, the fuel surface temperature distribution and the coolant bulk temperature distribution in the present JMTR were calculated with COBRA-3C/RERTR and FUELTEMP[2], and compared. FUELTEMP is a thermal-hydraulic code developed by JAERI for analysing the thermal safety margin of the JMTR. Table 2 shows the data used in these check calculations. Figure 1 shows the heat flux distribution. Figure 2 shows the comparison of the results calculated with COBRA-3C/RERTR and with FUELTEMP using these data. The coolant bulk temperatures are in very good agreement, and the fuel surface temperatures are fairly in agreement.

4. Thermal-Hydraulic Analyses for the JMTR

JMTR is a 50 MW tank type research and test reactor, cooled and moderated by light water and utilizing 93%EU fuel in the modified ETR type element. The present core consists of 22 standard and 5 follower fuel elements which are shown in Figure 3 and Figure 4, respectively. Figure 5 shows the present JMTR core configuration.

The JMTR has little thermal-hydraulic margin left. Thus, in the feasibility studies on the use of the LEU fuel in the JMTR, the thermal-hydraulic analyses were first made, and the neutronic analyses will be performed taking account of results of the thermal-hydraulic analyses.

4.1 Correlations Used in Thermal-Hydraulic Calculations for the JMTR.

Correlations used in thermal-hydraulic calculations for the JMTR are summarized in Table 3.

4.1.1 Friction Factor

In the case of smooth surface, the friction factor[3] can be expressed as:

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In the case of smooth surface, the friction factor[3] can be expressed as:

$$f = 0.316\ Re^{-0.25} \tag{1}$$

in the range $5{,}000 < Re < 51{,}094$

$$f = 0.184\ Re^{0.2} \tag{2}$$

in the range $51{,}094 < Re$.

According to the rough estimation, the Reynolds number for the JMTR fuel is:

$$Re \fallingdotseq 90{,}000 \ .$$

Therefore, Eq. (2) was used.

4.1.2 Heat Transfer Coefficient

The modified Colburn correlation[4] was used for the single phase heat transfer coefficient.

$$Nu = 0.023\ (Re_f)^{0.8}\ (Pr_f)^{0.3} \tag{3}$$

where

f = physical properties based on the film mean temperature.

This calculation was used for the design calculations of the JMTR and the HFBR.

The Bergles-Rohsenow correlation[5] for the subcool boiling heat transfer was used in order to predict the onset of nucleat boiling. Using this correlation, the heat flux at the onset of nucleate boiling is expressed as:

$$q'' = 15.6\ P^{1.156}\ (T_w - T_{sat})^{P^{\frac{2.30}{0.0234}}} \tag{4}$$

where

q'' = heat flux at ONB (Btu/hr·ft^2)

P = pressure (psia)

T_w = fuel surface temperature (°F)

T_{sat} = saturation temperature at the pressure P (°F).

The applicable range for the parameter is:

$$P = 15 - 2000 \text{ psia} \ .$$

The heat transfer regims are identified by using the correlation

$$T_w \geqq T_{w.s} \ . \tag{5}$$

T_w is the cladding wall (fuel surface) temperature based on single-phase heat transfer, and the cladding wall temperature, $T_{w.s}$, can be calculated with various correlation for subcool heat transfer. Thus, the fuel surface temperature at the onset of nucleate boiling is defined by:

$$(T_w)_{ONB} = T_{w.s} \quad . \tag{6}$$

In the case of the present JMTR, $(T_w)_{ONB}$ calculated by using the modified Colburn correlation and the Bergles-Rohsnow correlation is 201°C.

4.1.3 Critical Heat Flux

In order to estimate the margin to DNB, the following critical heat flux correlations were used.

1) Bernath Correlation[6]

$$q'' = h_c \ (T_{w,c} - T_b) \tag{7}$$

where

$$h_c = 10890 \quad \left(\frac{De}{De+Di}\right) + \frac{48\ V}{De^{0.6}} \quad , \text{ and}$$

$$T_{w,c} = 1.8 \ \left(57 \ln P - 54 \frac{P}{P+15} - \frac{V}{4}\right) + 32$$

T_b = coolant bulk temperature (°F)

V = coolant velocity (ft/s)

P = system pressure (psia)

De = hydraulic diameter (inch)

D_i = heated perimeter/π (inch)

The applicable ranges for the parameters are:

P = 23 - 3000 psia

V = 4.0 - 54 ft/sec

D = 0.143 - 0.66 inches

2) Labuntsov Correlation[7]

$$q'' = 4.61*10^5 \ F\ (P) \ \left(1 + \frac{0.232\ V^2}{F(P)} \right)^{0.25} \left(1 + \frac{15 C_p \ \Delta t}{P^{0.5} H_{fg}} \right) \tag{8}$$

where

q'' = surface heat flux (Btu/hr·ft^2)

$F(P)$ is a function of P, = $P^{1/3}\ (1-Pr)^{4/3}$

P = the system pressure in abs atm

Pr = reduced pressure (P/P_c)

P_c = critical pressure (abs atm)

V = velocity (ft/s)

C_p = specific heat (Btu ℓb-1 °F^{-1})

Δt = saturation temperature minus bulk water temperature (°F)

H_{fg} = heat of vaporization (Btu/lbm)

The applicable ranges for the parameters are:

1 atm < P < 200 atm

0.7 m/s < V < 45 m/s

This correlation was used to determine the burnout heat flux for each element configuration in the Oak Ridge Research Reactor.[8]

4.1.4 Critical Velocity

The Miller correlation[9] was used to compute the critical velocity.

$$V_c = \left(\frac{15\ g\ E\ (a^3 - tm^3)\ h}{\rho\ b^4\ (1 - \nu^2)} \right)^{1/2} \tag{9}$$

where

ρ = density (0.988 g/cm^3)

b = channel width (6.66 cm)

ν = Poisson's ratio (0.34)

g = gravity acceleration (980 cm/sec^2)

E = Young's modulus (6.3×10^8 g/cm)

tm = meat thickness (cm)

h = channel width (cm)

a = plate thickness (cm) .

(COBRA-3C/RERTR does not compute critical velocity.)

4.2 Parametric Study

Different cases considered in the parametric study are summarized in Table 4. In calculations for the 18-plate fuel element, fuel plates were assumed to be uniformly spaced, and the average heat flux was estimated assuming that the fuel follower had 15 fuel plates. Case 1 corresponds to the present HEU fuel element, and Case 7 corresponds to the present HEU fuel element where the plates are assumed to be uniformly spaced.

The velocities used in Cases 2 and 3 were estimated based on the assumption that the flow rate per fuel element was the same as that in Case 1.

According to the JMTR Final Design Calculation Report[10], the velocity of 11 m/sec is the maximum permissible velocity at normal operation in order to avoid the flow induced vibration of the fuel plate. Thus, the velocity of 11 m/sec was used in Cases 4, 5, 6, 8, 9 and 10.

The peaking factors F_b and F_f were estimated, assuming that the standard fuel element contained 340 g ^{235}U[11] and that the maximum burnup was 60%.

4.3 Results

The results are summarized in Table 5, and the fuel surface temperature distributions and the bulk water temperature distributions are shown in Figures 6 through 15.

4.3.1 Coolant Velocity

The coolant velocity shown for each case of 19-plate fuel element designs is that for the narrowest channel in the element. Coolant velocities in the differnt channels of the element were calculated to give the same inlet pressure gradient. Figure 16 shows the comparison of the measured[10] and computed velocity distributions in the present HEU fuel element. These are in good agreement.

4.3.2 Total Pressure Drop ΔP_T

The total pressure drop for each case was estimated based on the value of 3.2 kg/cm^2 for the present HEU core. The pressure drop excluding that due to friction across the active fuel region, ΔP_T-ΔP_f, can be expressed as:

$$\Delta P_T - \Delta P_f = cv^2$$

where

c = constant

v = coolant velocity in the element excluding the active fuel region.

Therefore,

$$\frac{\Delta P_T - \Delta P_t}{(\Delta P_T)_o - (\Delta P_t)_o} = \left(\frac{v}{v_o}\right)^2 \qquad (11)$$

The subscript o denotes the present HEU core.

The relationship between the coolant velocities in the active fuel region and in the element excluding the active fuel region is:

$$A \cdot V = a \cdot v \tag{12}$$

where

A = flow area in the active fuel region

a = flow area in the element excluding the active fuel region (This is kept constant.)

V = coolant velocity in the active fuel region

v = coolant velocity in the element excluding the active fuel region.

Thus,

$$\frac{A \cdot V}{A_o \cdot V_o} = \frac{v}{v_o} \tag{13}$$

From Eqs. (11) and (13), the total pressure drop can be expressed as:

$$\Delta P_T = \Delta P_f + [(\Delta P_T)_o - (\Delta P_f)_o] \cdot \left(\frac{A \cdot V}{A_o \cdot V_o}\right)^2 \tag{14}$$

4.3.3 Total Flow Rate

The total flow rate was estimated based on the fact that in the present HEU core the flow rates through the fuel region and the reflector region were about 3000 m^3/hr. The flow rate through the fuel region can be expressed as:

$$FR_F = (FR_F)_o \cdot \frac{A_o \cdot V_o}{A \cdot V} = 3000 \times \frac{A_o \cdot V_o}{A \cdot V} \tag{15}$$

The flow rate through the reflector region can be expressed as:

$$FR_R = C \cdot (\Delta P_T)^{0.5} \tag{16}$$

where

$$C = \frac{(FR_R)_o}{(\Delta P_T)_o^{\,0.5}} = 1677$$

Therefore, the total flow rate can be estimated by the following equation:

$$FR_T = 3000 \times \frac{A_o \cdot V_o}{A \cdot V} + 1677 \times (\Delta P_T)^{0.5} \quad . \tag{17}$$

As can be seen in Table 5, the total flow rates are less than the maximum flow rate of 6300 m^3/hr attained by the present pump system without losing the pump efficiency.

4.3.4 Margin to Critical Velocity

The margin for the LEU fuel case is larger than that for the present HEU case.

4.3.5 Margin to ONB

The margin to ONB is the minimum ratio of the actual heat flux to the heat flux at the onset of nucleat boiling. The margin for the LEU fuel case is larger than that of the present HEU fuel case.

4.3.6 Margin to DNB

The margin to DNB, MDNBR, is the minimum ratio of the actual heat flux to the critical heat flux calculated using the Labunstov correlation or the Bernath correlation. The margin for the LEU fuel case is larger than that for the present HEU fuel case. The JMTR parameters are within the applicable ranges for the parameters used in the Bernath correlation. But the margin calculated using this correlation seems to be in error because it is unreasonable to expect that the value for DNB increases as the meat thickness increases at the same average heat flux and the same coolant velocity.

4.4 Conclusions

Based on the results from the thermal-hydraulic analyses as shown in Table 5, the core conversion into the LEU core without changing the present pump system however with increased flow rate appears to be possible. In order to estimate the specification of the LEU fuel, further neutronic calculations must be performed. The ranges of possible design changes to be considered are as follows.

a. Meat thicknesses of 0.7 mm and 0.8 mm for 19-plate design.

b. Meat thicknesses of 0.8 mm and 0.9 mm for 19-plate design.

These ranges are concluded from the results of the thermal-hydraulic analyses, the relations between the fuel meat thickness and the uranium density (see Table 6), and the present fuel technology (see Table 7). In these LEU fuel designs, each LEU fuel element has 340 gr ^{235}U [12], and the uranium density is less than 3.0 gr/cm^3.

5. Thermal-Hydraulic analyses of the JRR-2

The JRR-2 is a 10 MW research reactor moderated and cooled by heavy water with 93% EU fuel elements. The present core consists of 20 MTR-type fuel and 4 cylindrical type fuel elements shown in Figure 17. The core configuration is shown in Figure 18.

In the case of the JRR-2, a specification of the LEU fuel was estimated roughly by neutronic and thermal-hydraulic analyses[11]. In these analyses, it was assumed that the LEU fuel core consisted of 24 cylindrical type fuel elements. The uranium density of this LEU fuel is 2.4 g/cm^3, and the meat thickness is 1.0 mm. (Based on the present fuel technology, it seems to be difficult to fabricate curved fuel plates with the meat thickness of 1.0 mm). Thus, the thermal-hydraulic analyses with COBRA-3C/RERTR were performed on the core with these LEU cylindrical type fuel elements.

5.1 Thermal-Hydraulic Calculations

Correlations and data used in thermal-hydraulic calculations are summerized in Tables 8 and 9, respectively. The coolant velocity for the LEU core is increased to 4.03 m/sec from 3.80 m/sec as in the HEU core. Figure 19 shows the heat flux distribution in the JRR-2.

5.2 Results

Table 10, Figures 20 and 21 summarize the results. As can be seen in Figures 20 and 21, the maximum fuel surface temperature is at the core inlet, and the maximum fuel surface temperature for the LEU case is lower than that for the HEU fuel case. Thus, the LEU fuel core has larger margin to ONB and DNB than the HEU core.

5.3 Conclusions

The margin to ONB and DNB for the LEU fuel case are larger than that for the HEU fuel cases due to increased coolant velocity. Therefore, based on the results from the thermal-hydraulic analyses the conversion into LEU fuel with the meat thickness of 1.0 mm appears to be feasible.

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Table 1 Outline of ANL-JAERI Joint Study on Reduced Enrichment Fuel for JAERI Reactors

Phase A (October 1979 - July 1980)

- Normalization calculations of IAEA benchmark problem.
- Transmittal by JAERI of detailed information on JAERI reactors.
- Feasibility studies on the use of LEU (<20%) Fuels in JAERI reactors.

** Planning and Preparations for Burnup Tests.

- Planning and preparations for critical experiments and full-core demonstration.

Phase B (July 1980 - December 1981)

- Burnup tests in the ORR (<20%, 45%).
- Burnup tests in the HFR Petten (<20%).
- Burnup tests in JAERI reactors (<20%, 45%).
- Critical experiments and full-core demonstration in the FNR (<20%).
- Critical experiments in the JMTRC (45%).
- Hydraulics tests at JAERI
- Further feasibility and analytical studies.

Phase C (December 1981 -)

- Full core demonstration tests in JAERI reactors.
- Final studies and evaluations.

Table 2 Data Used in the Check Calculation of COBRA-3C/RERTR

Number of Plates	19
Meat Thickness (mm)	0.5
Plate Thickness (mm)	1.27
Active Fuel Length (mm)	750
Active Fuel Width (mm)	61.6
Coolant Channel Width (mm)	66.6
Coolant Channel Thickness (mm)	3.02 × 4 2.92 × 2 2.67 × 12
Average Heat Flux (W/cm^2)	115.0
Hot Spot Factors	
Nuclear Unvertainty Factor $F_N = F_R \cdot F_Z$	
Axial Peak to Average F_Z	Fig. 1
Radial Peak to Average F_R	2.86
Uncertainty Factor for Bulk Water F_b	1.28
Uncertainty Factor for Film Temp. F_f	1.36
Coolant Velocity (m/sec)	10.0
Coolant Temperature at Core Inlet °C	49
Coolant Pressure at Core Inlet (kg/cm^2 A)	15.0

Table 3 Correlations Used in Thermal-Hydraulic Calculations for the JMTR

	Correlation
Friction Factor	$f = 0.184\ Re^{-0.2}$ $(Re)_{JMTR} = 90{,}000$
Single Phase Heat Transfer Coefficient	Modified Colburn $Nu = 0.023\ (Re_f)^{0.8}\ (Pr_f)^{0.3}$ This correlation takes into account the temperature rise in the film.
Subcool Boiling Heat Transfer Coefficient	Bergles-Rohsenow
Critical Heat Flux	Bernath, Labuntsov
Critical Velocity	Miller

Table 4 JMTR Parametric Study

Datum Item \ Case No.	1	2	3	4	5	6	7	8	9	10
Number of Plates	19	19	19	19	19	19	19	18	18	18
Heat Thickness (mm)	0.5	0.6	0.7	0.8	0.9	1.0	0.5	0.8	0.9	1.0
Plate Thickness (mm)	1.27	1.37	1.47	1.57	1.67	1.77	1.27	1.57	1.67	1.77
Active Fuel Length (mm)	750.0	750.0	750.0	750.0	750.0	750.0	750.0	750.0	750.0	750.0
Active Fuel Width (mm)	59.5	59.5	59.5	59.5	59.5	59.5	59.5	59.5	59.5	59.5
Coolant Channel Width (mm)	66.6	66.6	66.6	66.6	66.6	66.6	66.6	66.6	66.6	66.6
Coolant Channel Thickness (mm)	3.02 x 4 2.92 x 2 2.67 x 12	2.92 x 4 2.82 x 2 2.57 x 12	2.82 x 4 2.72 x 2 2.47 x 12	2.72 x 4 2.62 x 2 2.37 x 12	2.62 x 4 2.52 x 2 2.27 x 12	2.52 x 4 2.42 x 2 2.17 x 12	2.79 x 18	2.72 x 17	2.62 x 17	2.52 x 17
Average Heat Flux (W/cm^2)	119.2	119.2	119.2	119.2	119.2	119.2	119.2	125.9	125.9	125.9
Hot Spot Factor Nuclear Uncertainty Factor $F_N = F_R \cdot F_Z$ Axial Peak to Average F_Z	Fig. 1	Fig. 1	Fig. 1	Fig. 1	Fig. 1	Fig. 1	Fig. 1	Fig. 1	Fig. 1	Fig. 1
Radial Peak to Average F_R	2.86	2.86	2.86	2.86	2.86	2.86	2.86	2.86	2.86	2.86
Uncertainty Factor for Bulk Water Temperature F_b	1.25	1.27	1.27	1.27	1.28	1.28	1.25	1.27	1.27	1.27
Uncertainty Factor for Film Temperature F_f	1.35	1.36	1.36	1.36	1.36	1.36	1.35	1.36	1.36	1.36
Coolant Velocity (m/sec)	10.0	10.4	10.8	11.0	11.0	11.0	10.0	11.0	11.0	11.0
Coolant Temperature at Core Inlet (°C)	49.0	49.0	49.0	49.0	49.0	49.0	49.0	49.0	49.0	49.0
Coolant Pressure at Core Inlet (kg/cm^2 A)	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0

Table 5 Thermal-Hydraulic Analysis of the JMTR with the LEU Fuel

Case No.	Meat Thickness	Number of Plates	Average Heat Flux (w/cm^2)	Coolant Channel Thickness (mm)	Coolant Velocity (m/sec)	Maximum Fuel Surface Temperature (Tw) max (°C)	Coolant bulk Temperature at Outlet (°C)	Pressure Drop by Friction ΔP_f (kg/cm^2)	Total Pressure Drop ΔP (kg/cm^2)	Total Flow Rate (m^3/hr)	Margin to Critical Velocity	Margin to ONB	Margin to DNB MDNBR Lobuntsov	Margin to DNB MDNBR Bernath
1	0.5[1]	19	119.2	2.67	9.7	187	102	1.26	3.20	6000	1.71	1.12	3.14	2.66
2	0.6	19	119.2	2.57	10.0	183	103	1.41	3.35	6069	1.80	1.16	3.13	2.74
3	0.7	19	119.2	2.47	10.4	181	103	1.58	3.52	6147	1.87	1.19	3.18	2.88
4	0.8	19	119.2	2.37	10.6	179	104	1.72	3.59	6124	1.95	1.21	3.20	2.97
5	0.9	19	119.2	2.27	10.6	181	107	1.80	3.52	5973	2.07	1.19	3.14	2.97
6	1.0	19	119.2	2.17	10.6	182	110	1.89	3.47	5833	2.18	1.17	3.11	2.99
7	0.5[2]	19	119.2	2.79	10.0	185	98	1.27	3.21	6005	1.70	1.14	3.23	2.73
8	0.8	18[3]	125.9	2.72	11.0	180	98	1.57	3.57	6213	2.02	1.20	3.15	2.79
9	0.9	18	125.9	2.62	11.0	181	100	1.64	3.49	6065	2.14	1.20	3.13	2.82
10	1.0	18	125.9	2.52	11.0	181	102	1.72	3.43	5926	2.25	1.19	3.11	2.85

[1] This case corresponds to the present HEU fuel element.

[2] This case corresponds to the present HEU fuel element where the plates are assumed to be uniformly spaced in the element.

[3] In all 18 plate cases, the fuel plates are uniformly spaced in the element.

Table 6 Meat Thickness vs. Uranium Dencity and Vp/Vc

MEAT THICKNESS (MM)	PLATE THICKNESS (MM)	URANIUM DENSITY* G/CM^3 19 PLATES	18 PLATES	V_P/V_C** 19 PLATES	18 PLATES
0.5	1.27	4.0	4.22	0.313	0.296
0.6	1.37	3.33	3.52	0.337	0.319
0.7	1.47	2.86	3.01	0.362	0.342
0.8	1.57	2.50	2.64	0.386	0.366
0.9	1.67	2.22	2.34	0.411	0.389
1.0	1.77	2.00	2.11	0.436	0.413
1.1	1.87		1.92		0.436

*340 G ^{235}U/ELEMENT.

**
Vp : Volum of fuel plates per element
Vc : Volum of coolant channel per fuel element

Table 7 Anticipated Uranium Loadings, g/cm^3

Fuel System	Existing Techno-logy	Near-Term+ Very Likely	Near-Term+ Some Uncertainty	Long-Term+
UAl_x-Al	1.7	1.5[a]	1.7[a]	3.0[a]
		2.4[b]	2.6[b]	2.8[b]
		2.2[c]	2.4[c]	2.5 - 2.8[c]
U_3O_8-Al	1.7	1.3[a]	2.6[a]	3.3[a]
		2.8[d]	3.0[d]	3.2[d]
		2.7[c]	3.0[c]	-

a NUKEM

b EG&G Idaho

c CERCA

d ORNL

*, rod-type

**, plate-type

+Footnote - near-term 1-3 years, long-term $\geq$ 5 years.

Table 8 Correlations Used in Thermal-Hydraulic Calculations for the JRR-2

	Correlation
Friction Factor	$f = 0.316\ Re^{-0.25}$ $(Re)_{JRR-2} = 30,000$
Single Phase Heat Transfer Coefficient	Dittus-Boelter $Nu = 0.023\ (Re_b)^{0.8}\ (Pr_b)^{0.4}$ This correlation cannot take into account the temperature rise in the film.
Subcool Boiling Heat Transfer Coefficient	Bergles-Rohsenow
Critical Heat Flux	Labuntsov

Table 10 Thermal-Hydraulic Analysis for the JRR-2

	HEU Fuel	LEU Fuel
Fuel Plate Thickness (cm)	0.127	0.176
Water Channel Thickness (cm)	0.30	0.25
Water Channel Area/Element (cm^2)	40.82	38.23
Reactor Thermal Power (MW)	10	10
Primary Flow Rate (m^3/min)	22	22
Core Inlet Water Temperature (°C)	57.1	57.1
Core Outlet Water Temperature (°C)	67.5	68.9
Clad Surface Temperature (Maximum) (°C)	115.8	111.9
Water Velocity (m/sec)	3.8	4.03
Average Heat Flux (W/cm^2)	31.7	31.9
Margin to ONB	1.34	1.46
Margin to DNB	11.6	11.9

Table 9 Data Used in Calculations for the JRR-2

	HEU Fuel	LEU Fuel
Number of Plates	15	15
Meat Thickness (mm)	0.51	1.00
Plate Thickness (mm)	1.27	1.76
Active Fuel Length (mm)	600.0	600.0
Active Fuel Width (mm)	48.6 x 3, 57.6 x 3, 66.5 x 3 75.4 x 3 84.4 x 3	48.6 x 3, 57.6 x 3, 66.5 x 3 75.4 x 3 84.4 x 3
Coolant Channel Thickness (mm)	3.0	2.5
Coolant Channel Area (mm^2)	151.3 x 3, 178.1 x 3, 204.9 x 3 231.6 x 3, 258.4 x 3, 285.4 x 3	126.1 x 3, 148.4 x 3, 170.7 x 3 193.0 x 3, 215.3 x 3, 237.6 x 3
Average Heat Flux (W/cm^2)	31.7	31.9
Hot Spot Factor Nuclear Uncertainty Factor $F_N = F_R \cdot F_Z$		
Axial Peak to Average F_Z	Fig. 19	Fig. 19
Radial Peak to Average F_R	1.49	1.49
Uncertainty Factor for Bulk Water Temperature F_b	1.30	1.30
Uncertainty Factor for Film Temperature F_f	1.66	1.66
Coolant Velocity (m/sec)	3.80	4.03
Coolant Temperature at Core Inlet (°C)	57.1	57.1
Coolant Pressure at Core Inlet (kg/cm^2 A)	1.70	1.70

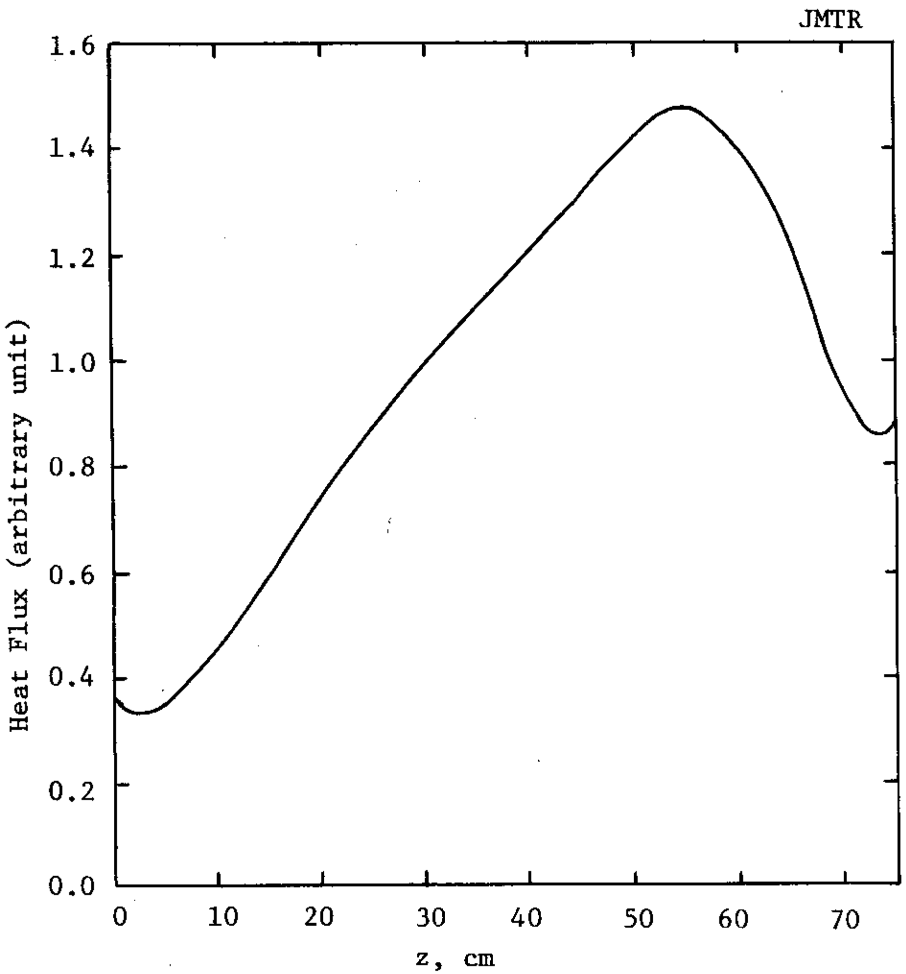


Fig. 1 Heat flux distribution in the JMTR

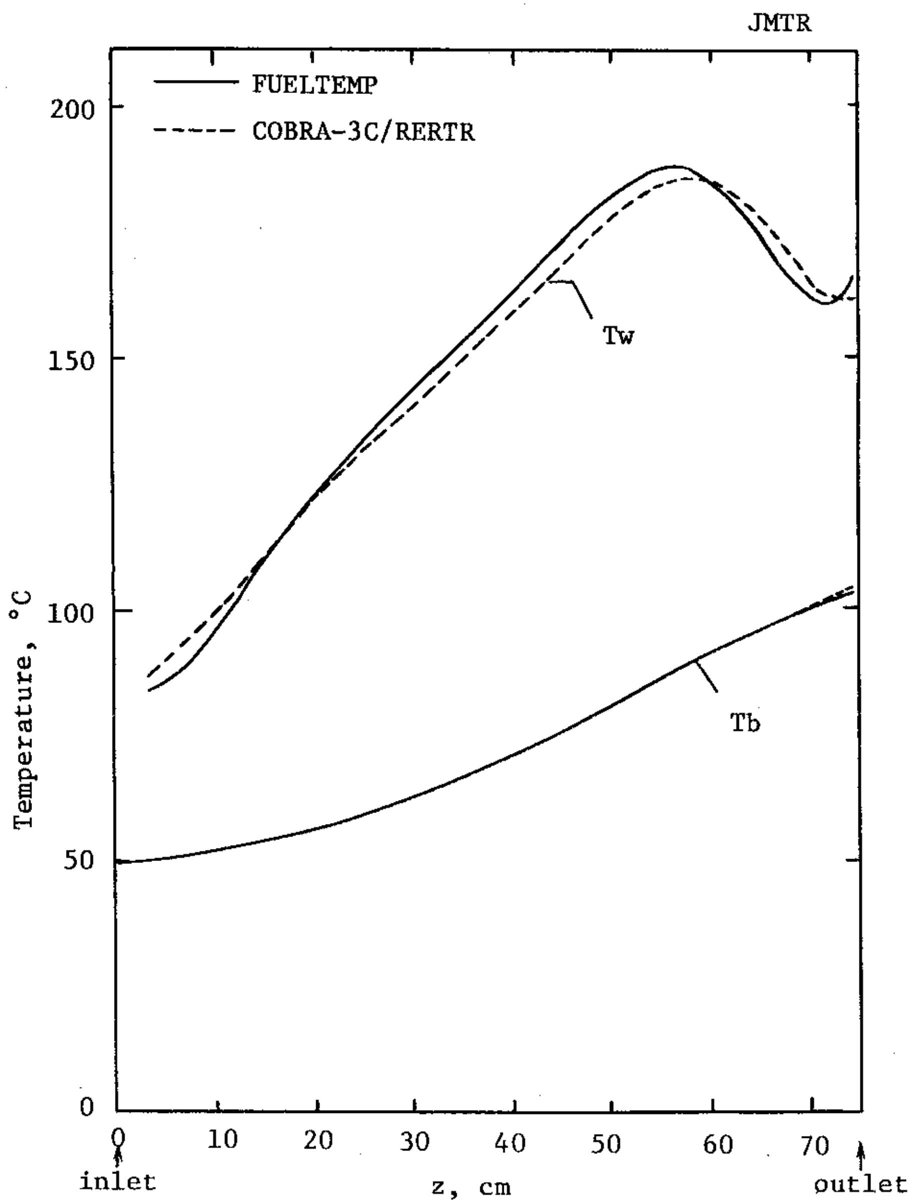


Fig.2 Fuel surface and bulk water temperature distributions calculated with FUELTEMP and COBRA-3C/RERTR

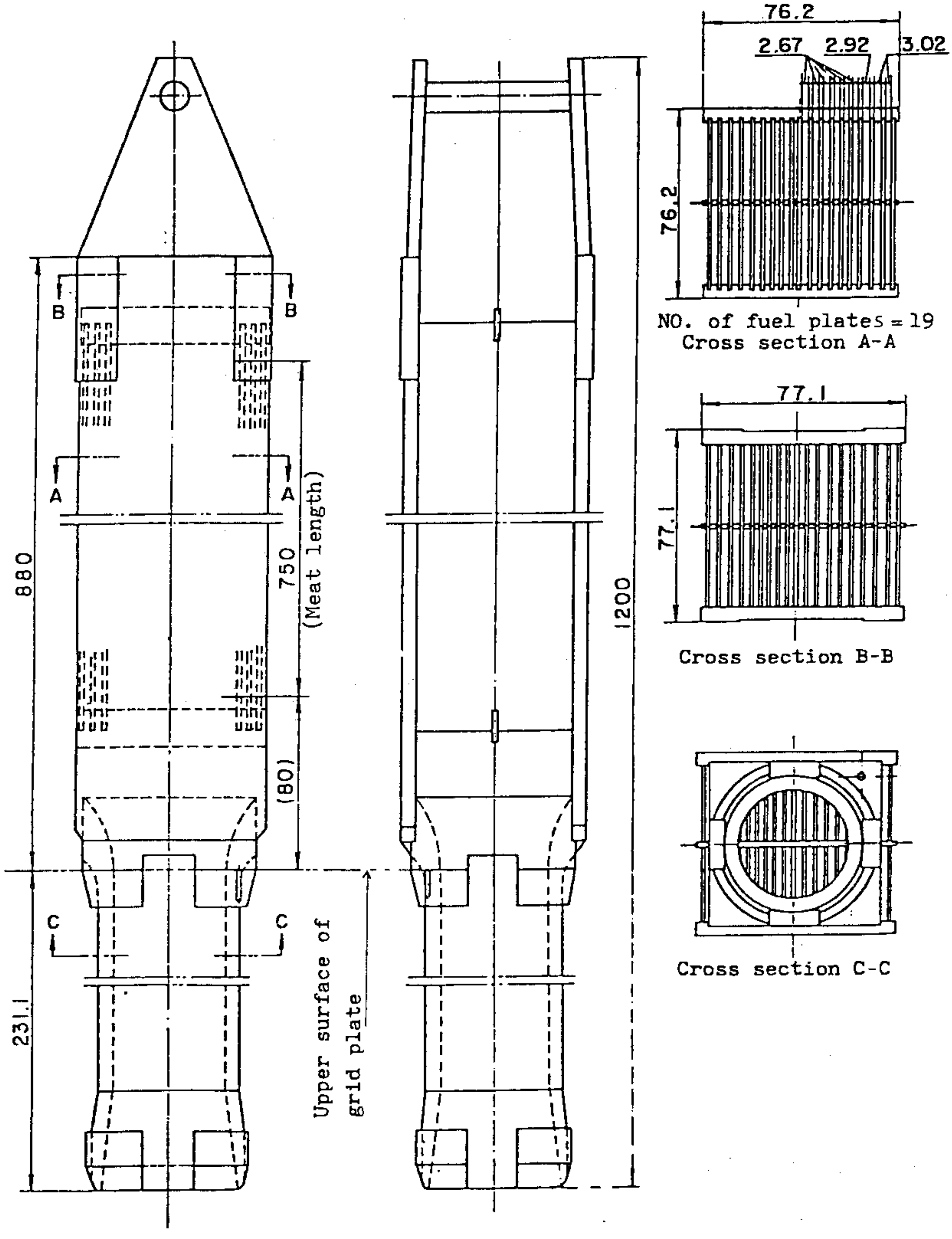


Fig. 3 JMTR Standard Fuel Element

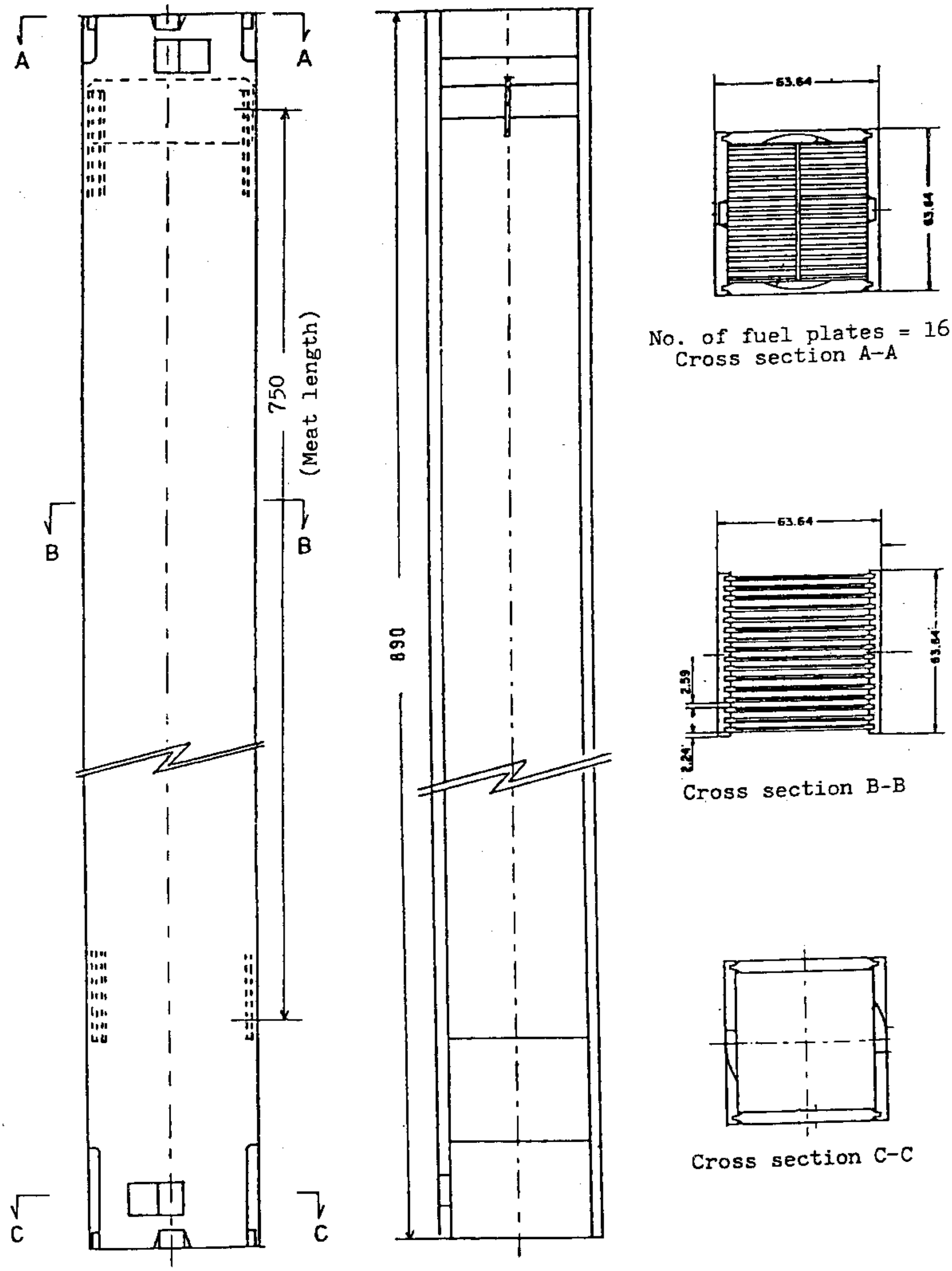


Fig. 4 JMTR Fuel Follower

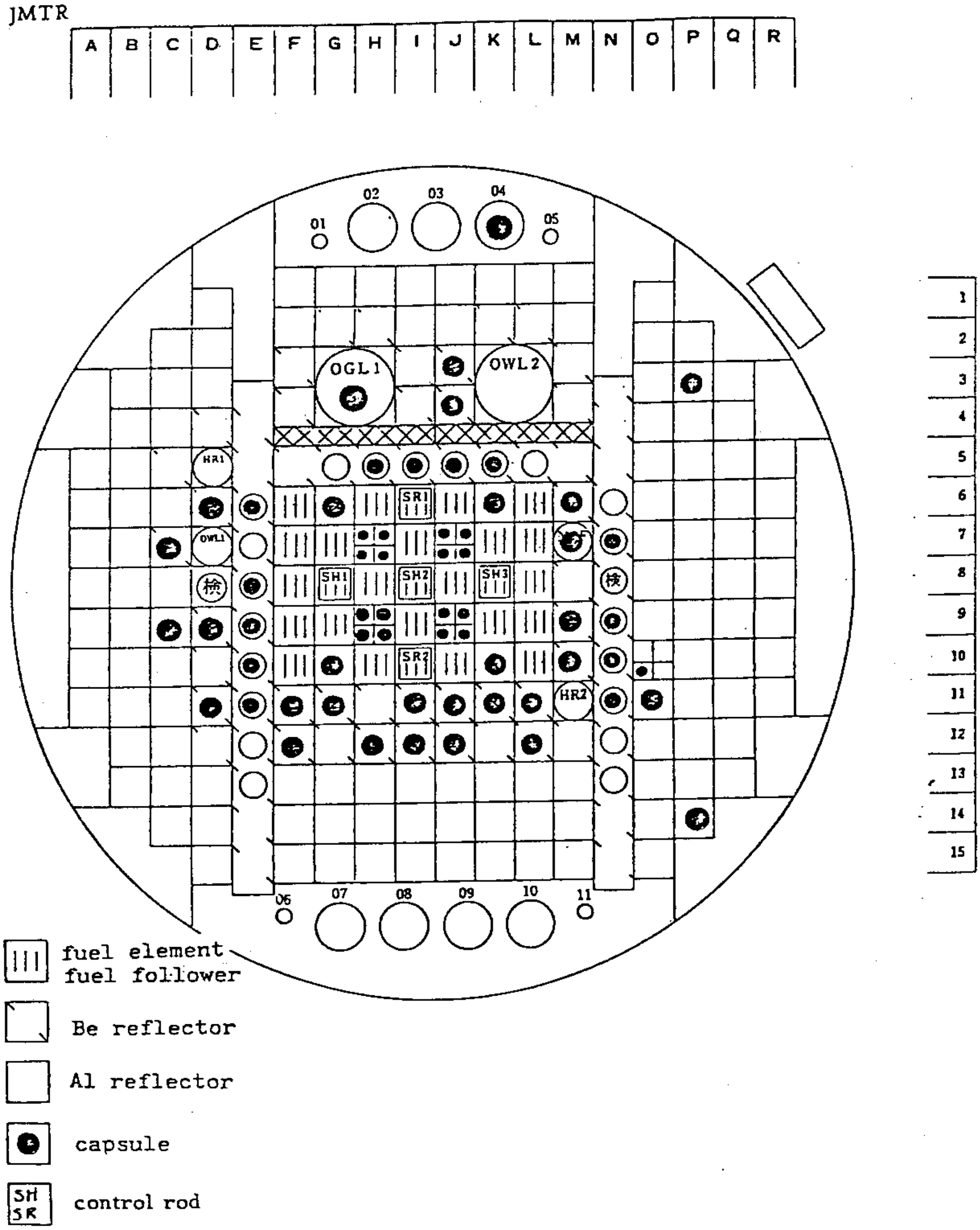


Fig. 5 Present JMTR core configuration

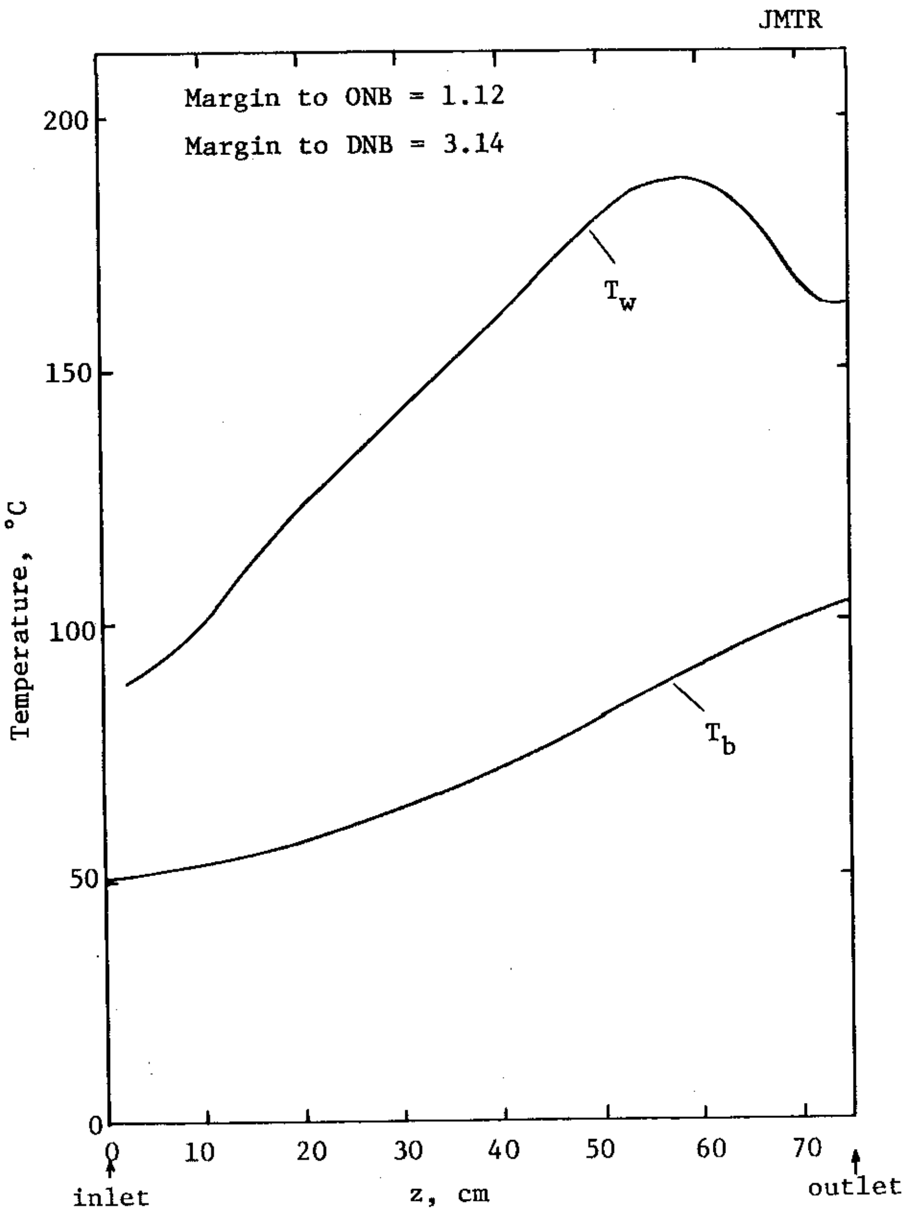


Fig.6 Fuel surface and coolant bulk temperature distributions in the JMTR HEU fuel element (Number of plates=19, Meat thickness=0.5mm)

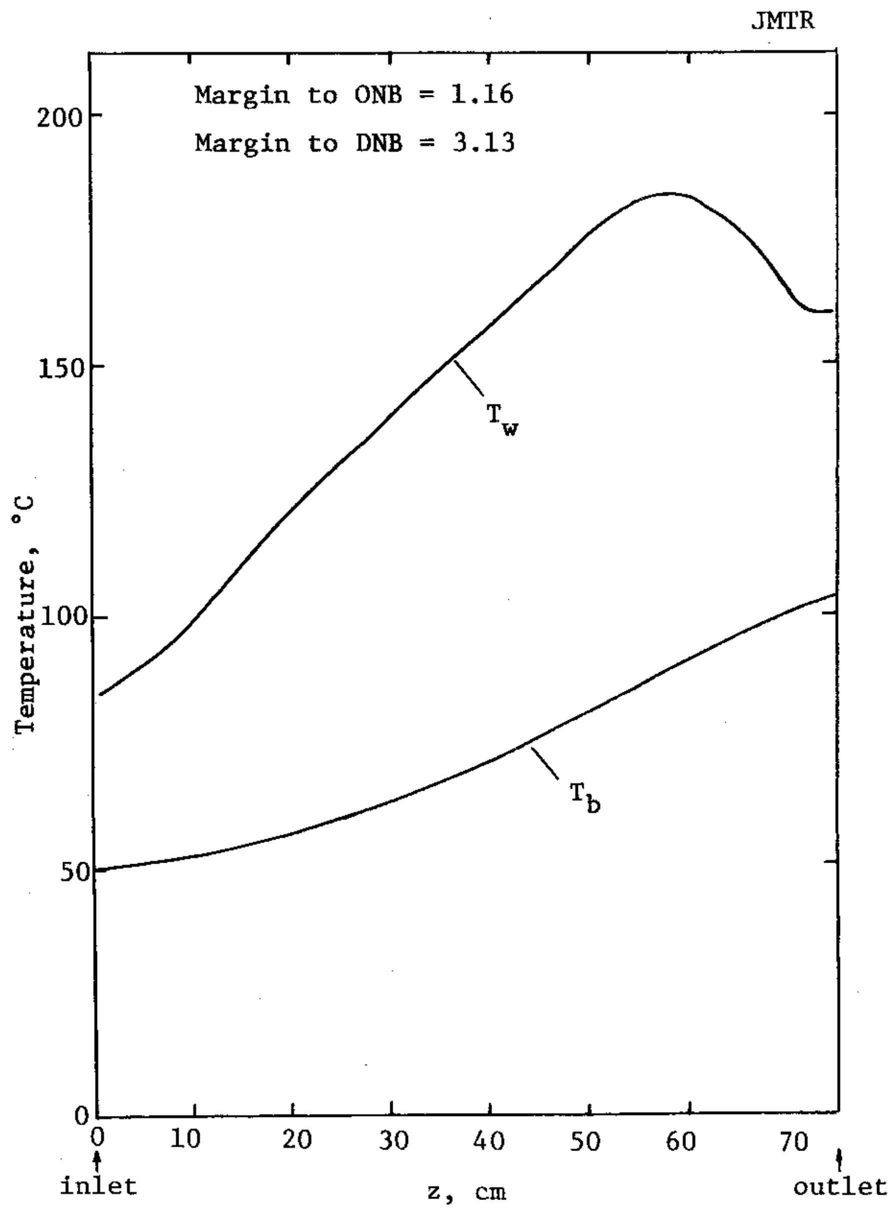


Fig.7 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=19, Meat thickness=0.6mm)

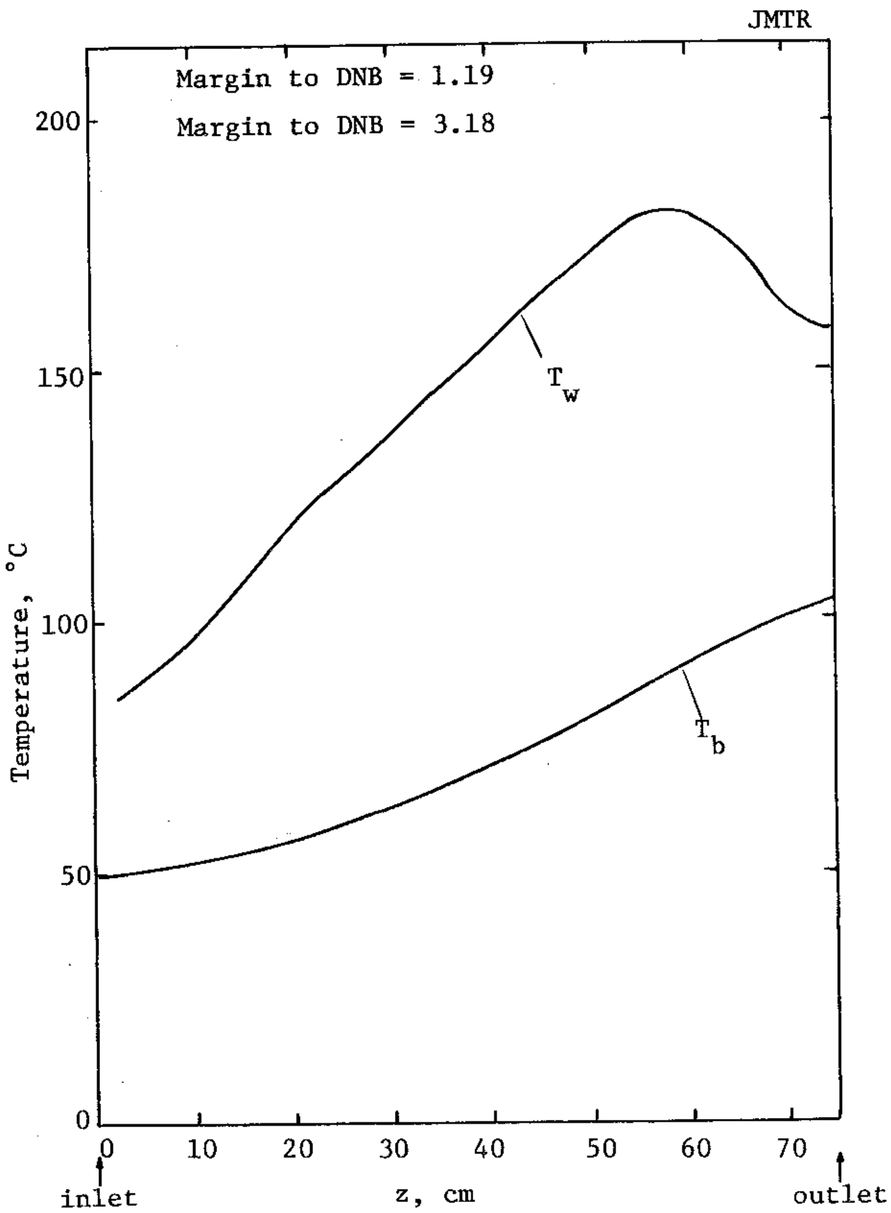


Fig.8 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=19, Meat thickness=0.7mm)

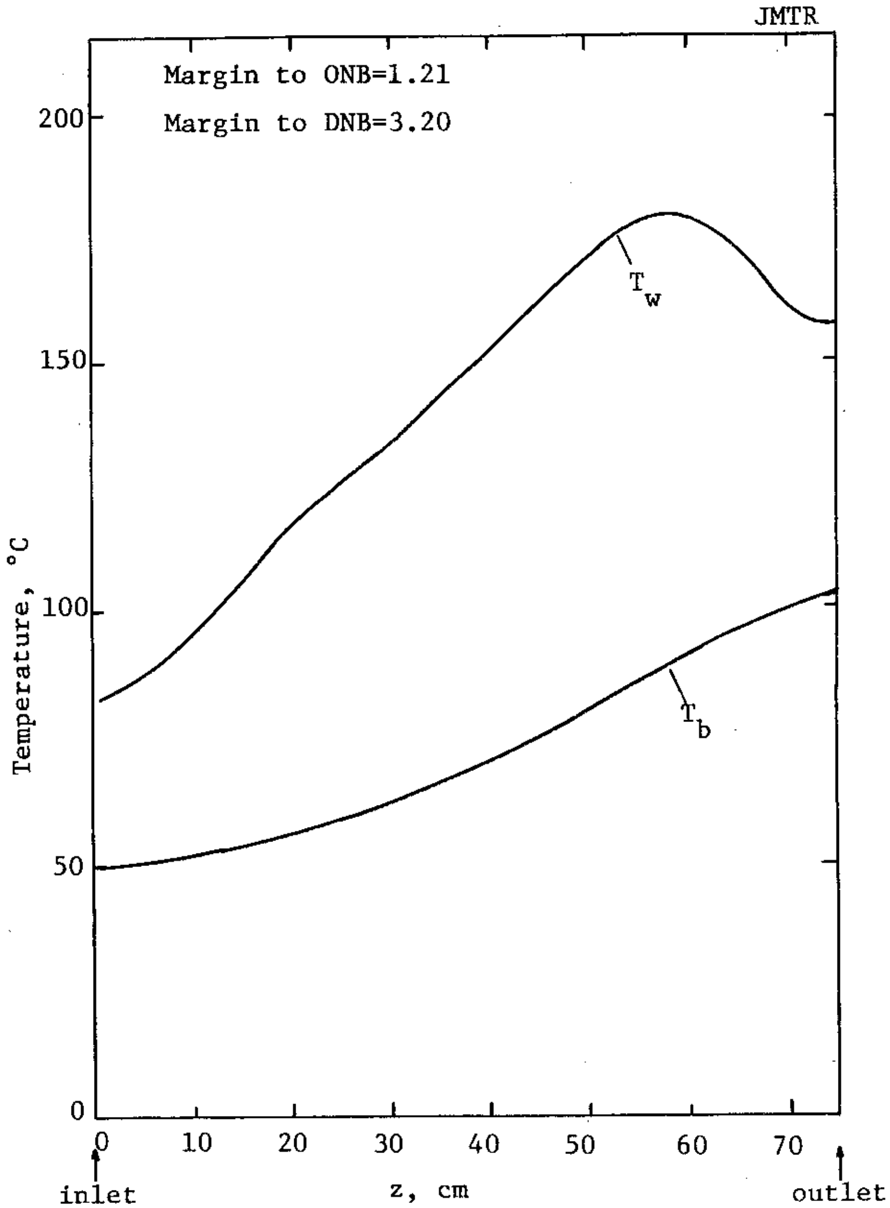


Fig.9 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=19, Meat thickness=0.8mm)

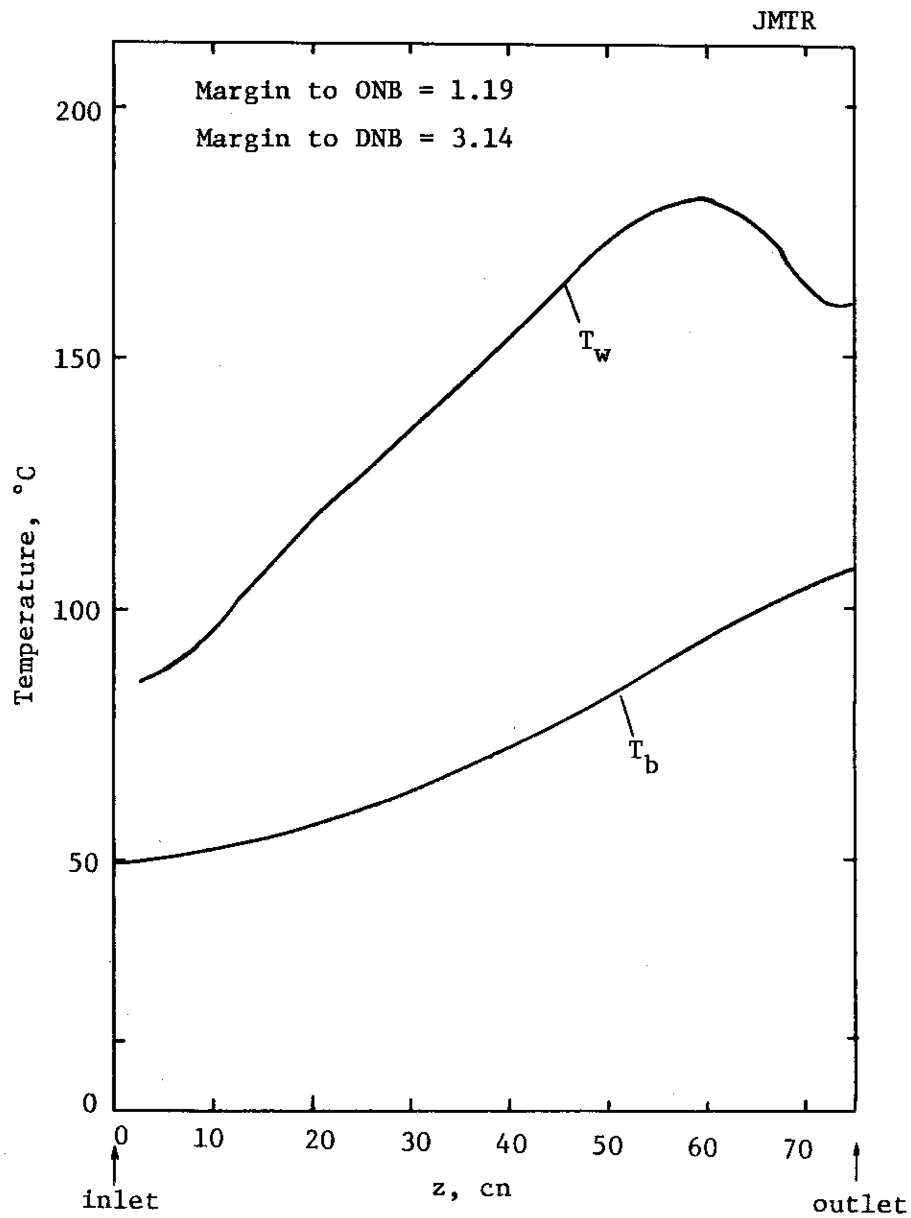


Fig.10 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=19, Meat thickness=0.9mm)

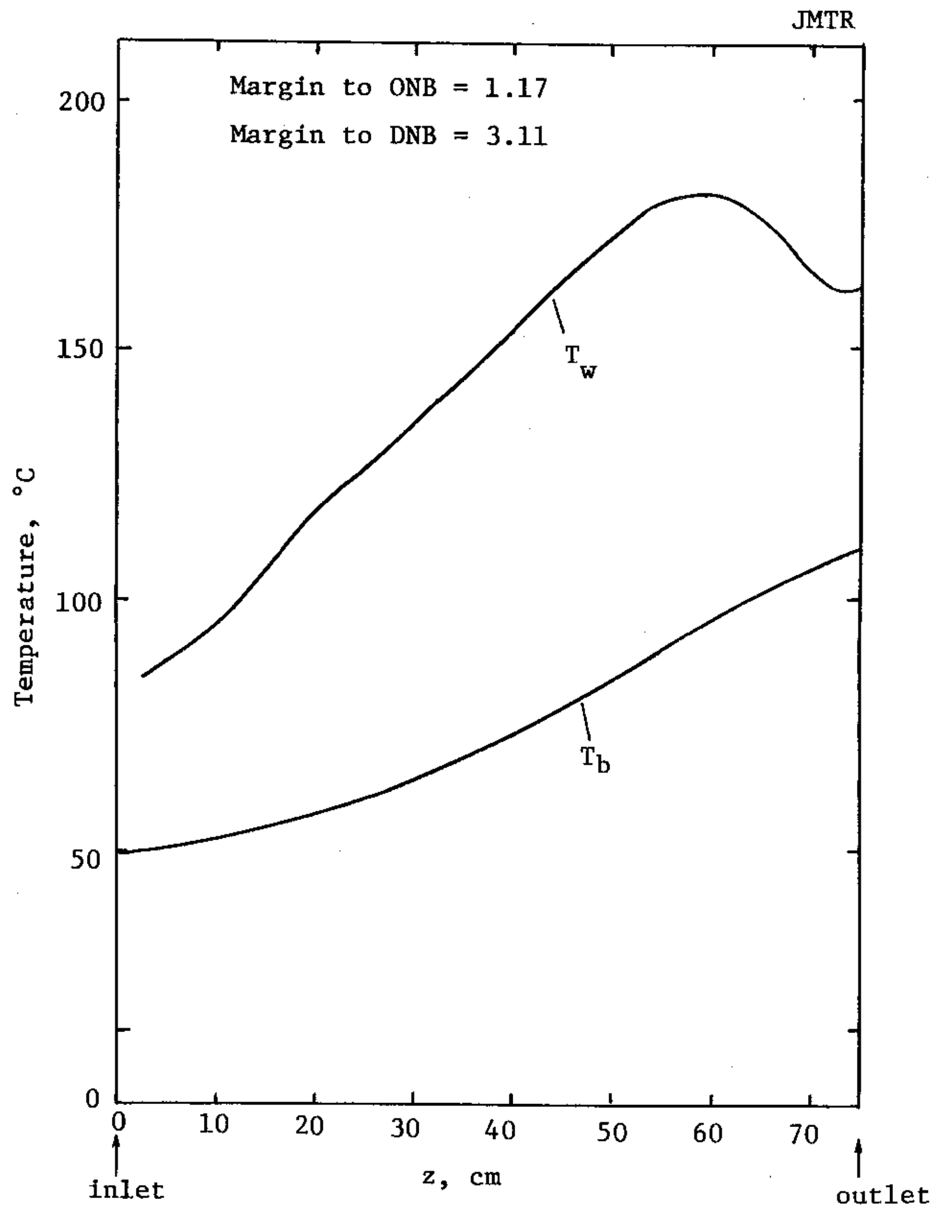


Fig.11 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=19, Meat thickness=1.0mm)

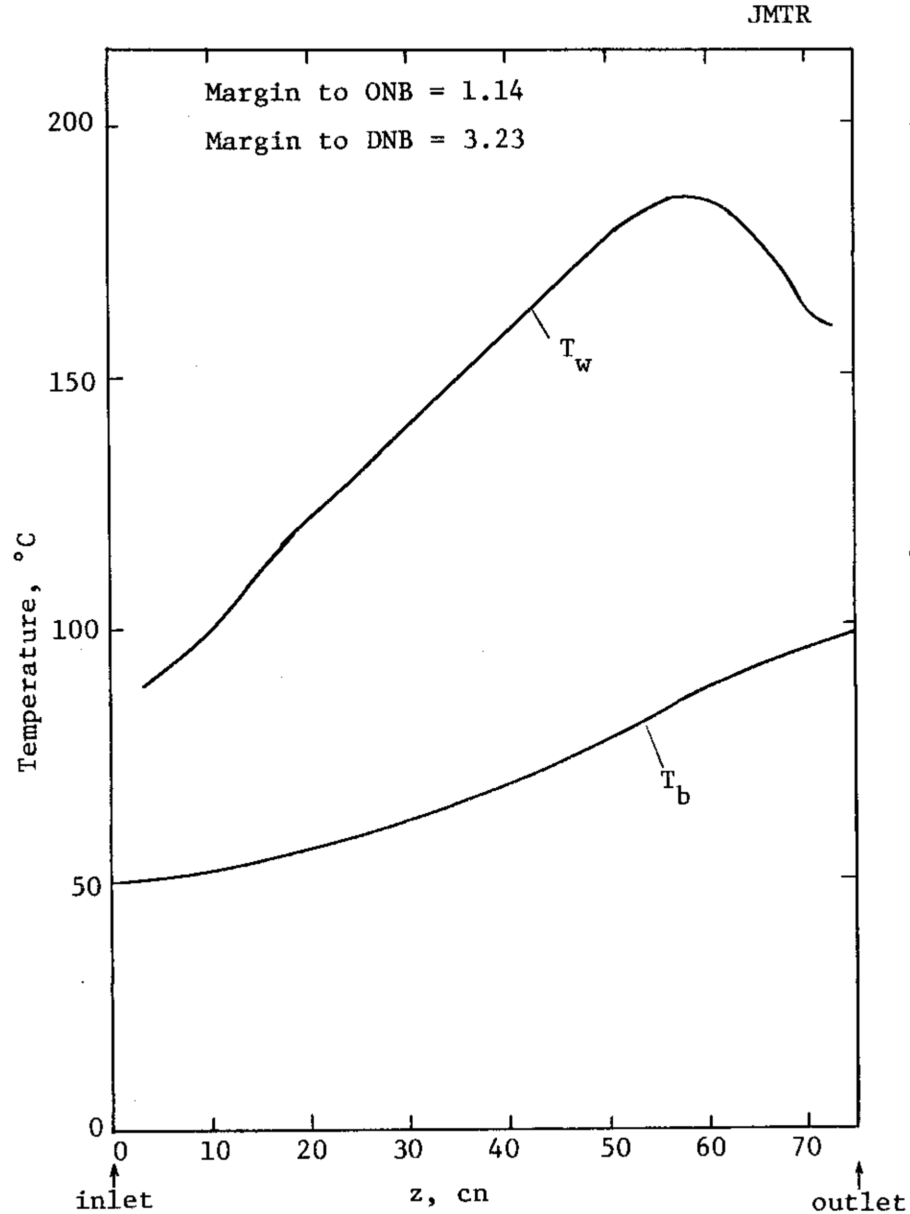


Fig.12 Fuel surface and coolant bulk temperature distributions in the JMTR HEU fuel element with uniformaly spaced fuel plates (Number of plates=19, Meat thickness=0.5mm)

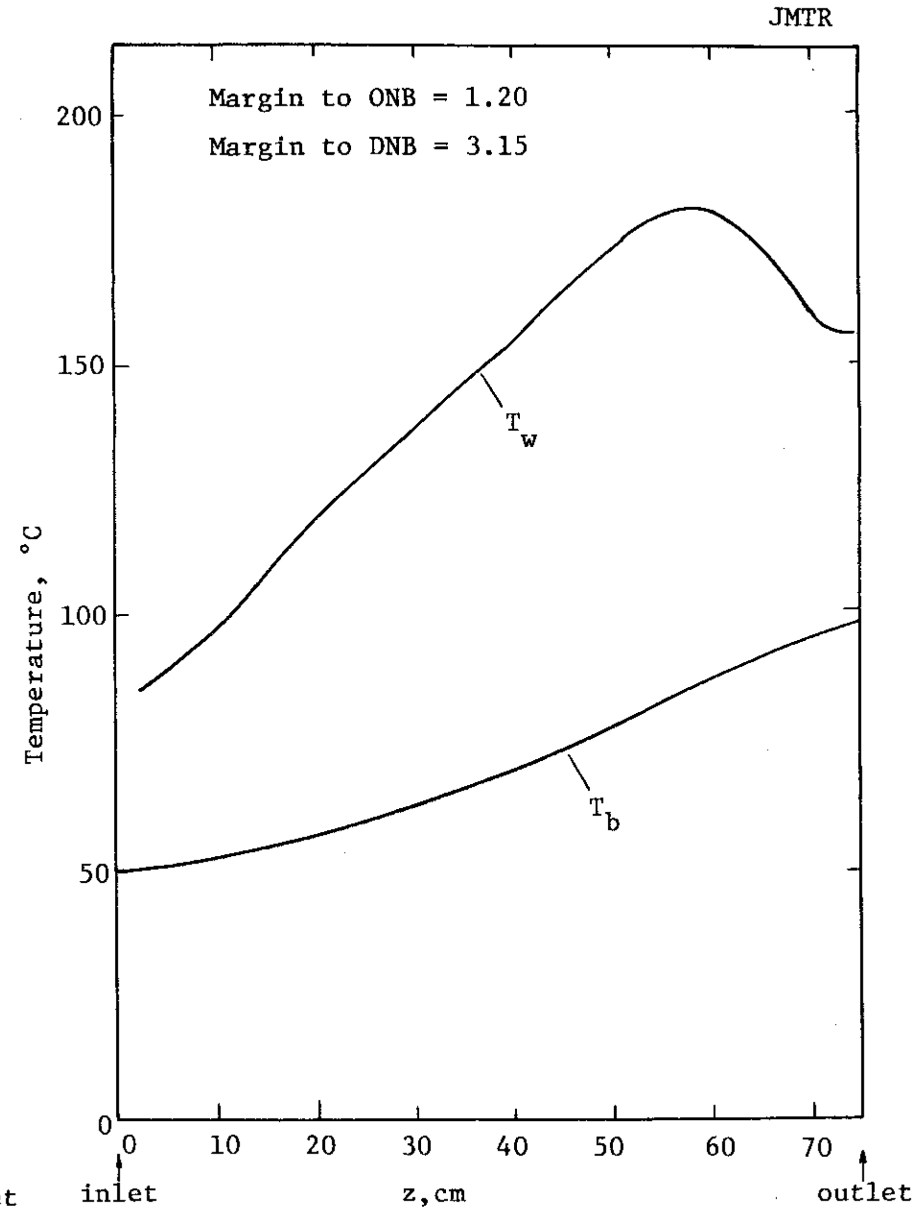


Fig.13 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=18, Meat thickness=0.8mm)

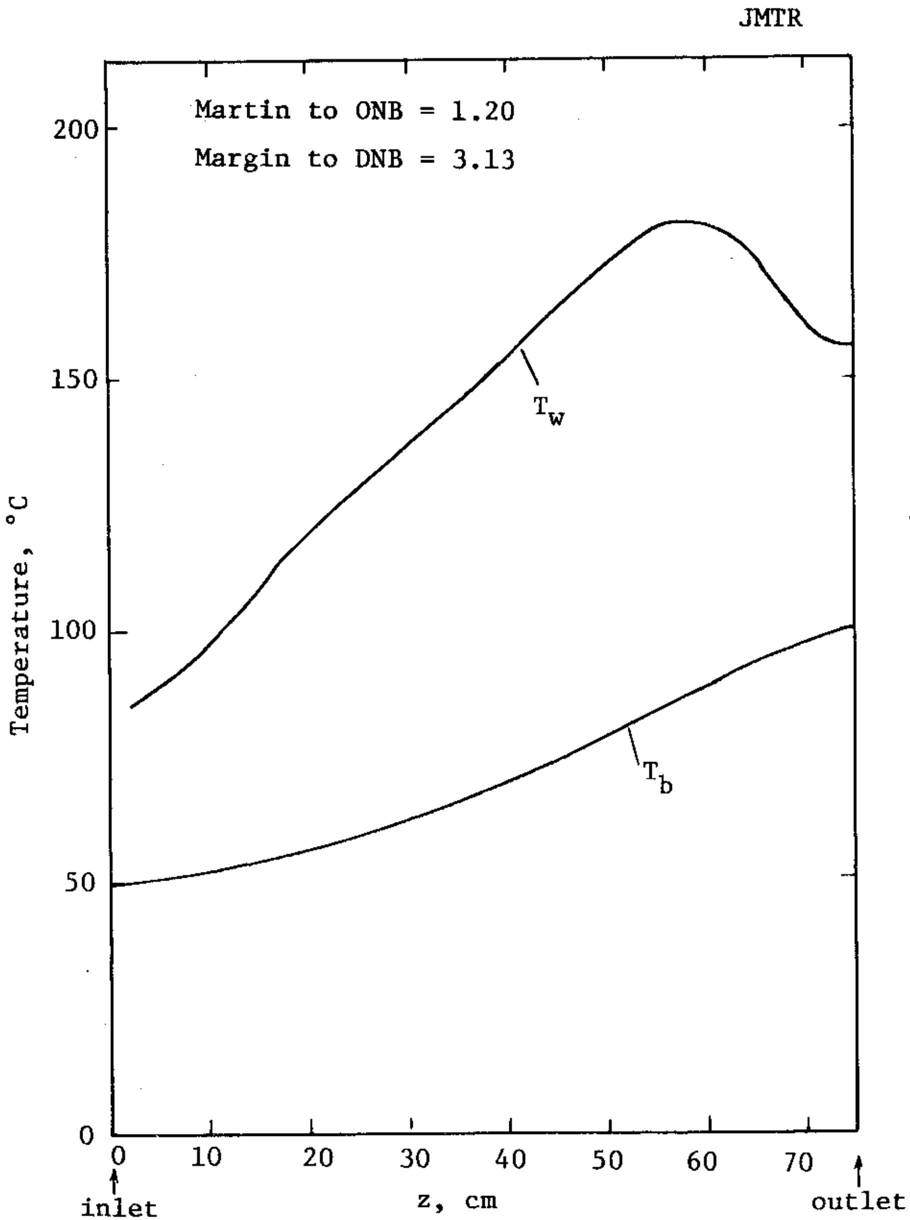


Fig.14 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=18, Meat thickness=0.9mm)

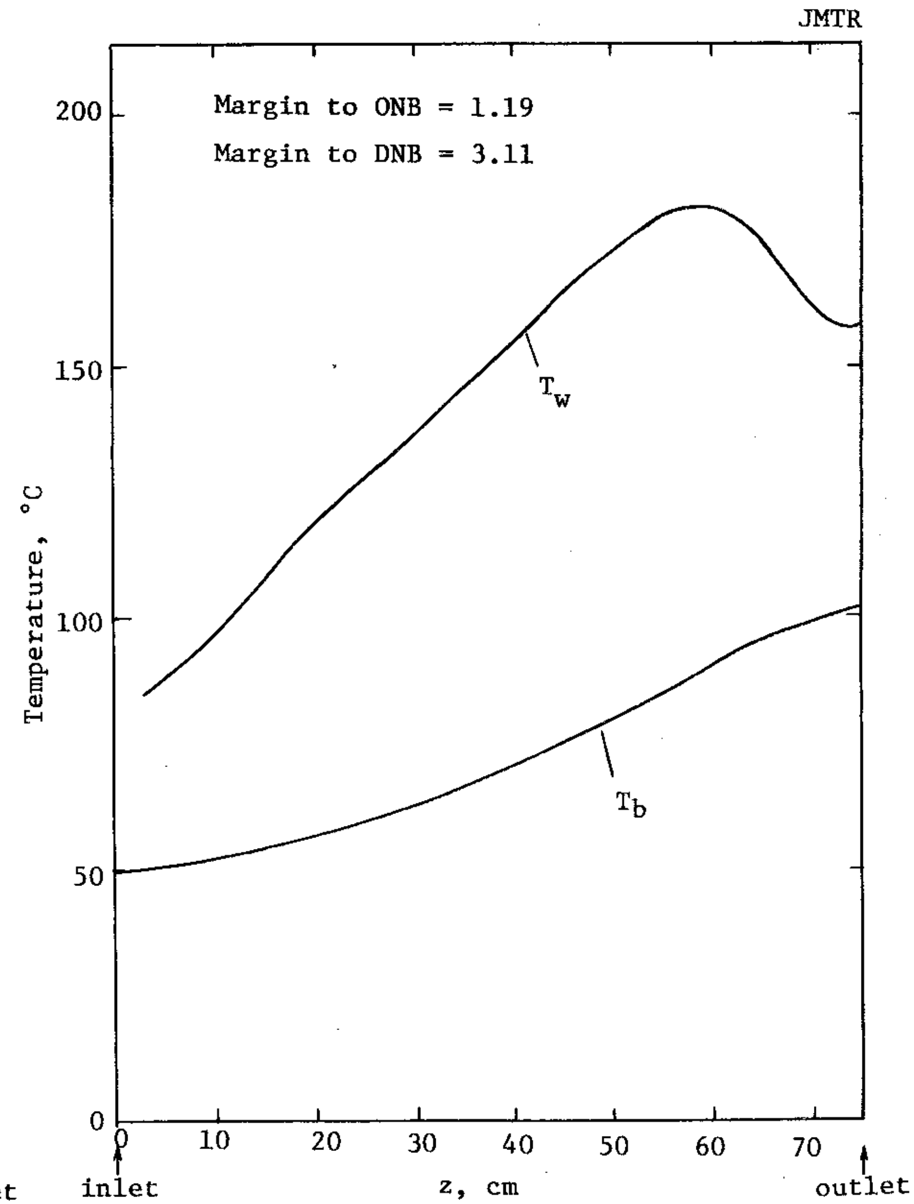
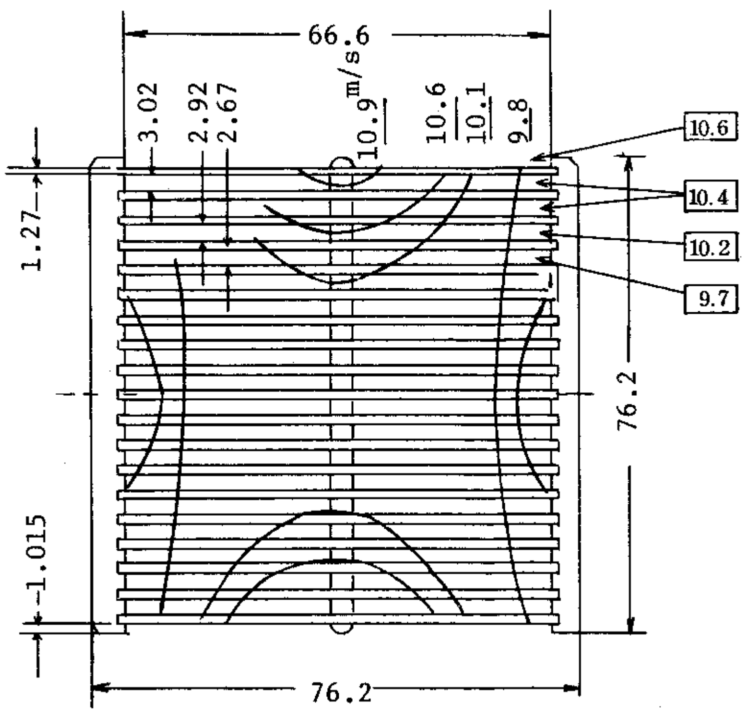


Fig.15 Fuel surface and coolant bulk temperature distributions in the JMTR LEU fuel element (Number of plates=18, Meat thickness=1.0mm)



10.9: measured value (m/sec)

$\boxed{10.6}$: calculated value (m/sec)

Fig.16 Coolant velocity distribution in the present HEU fuel element

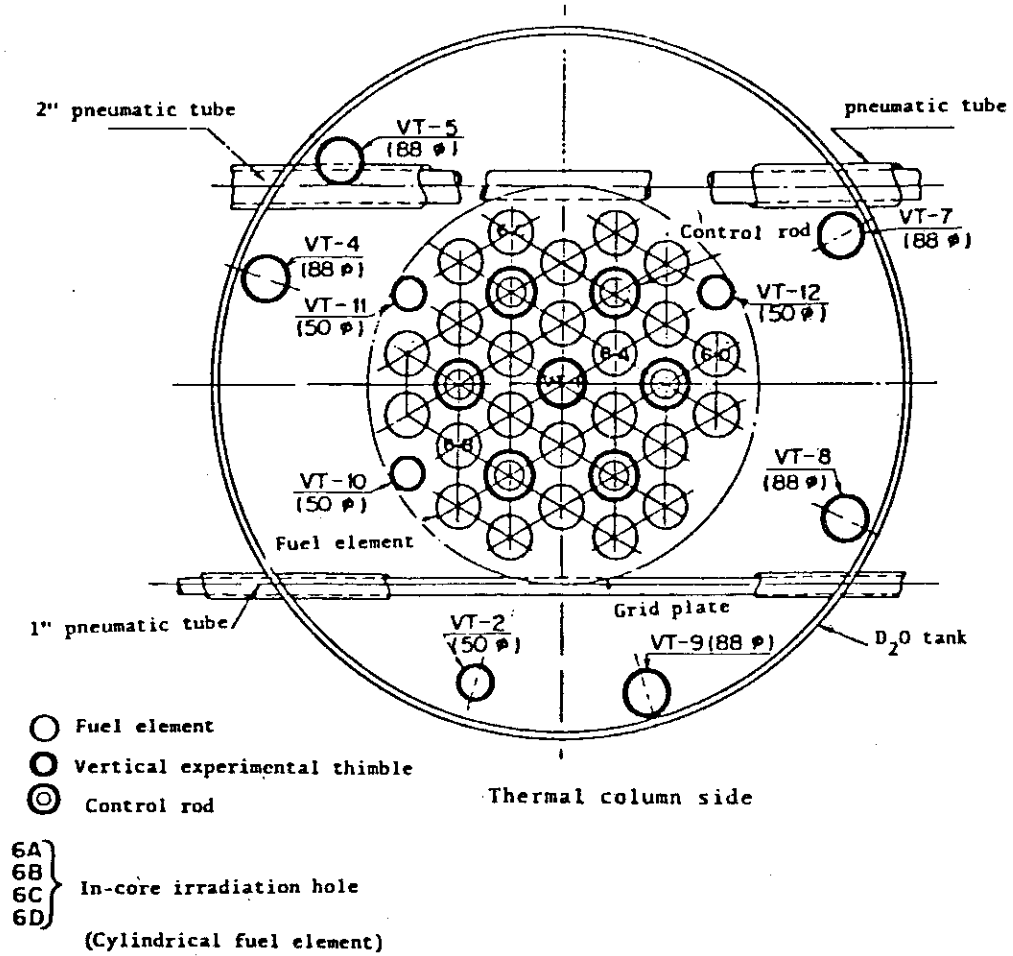
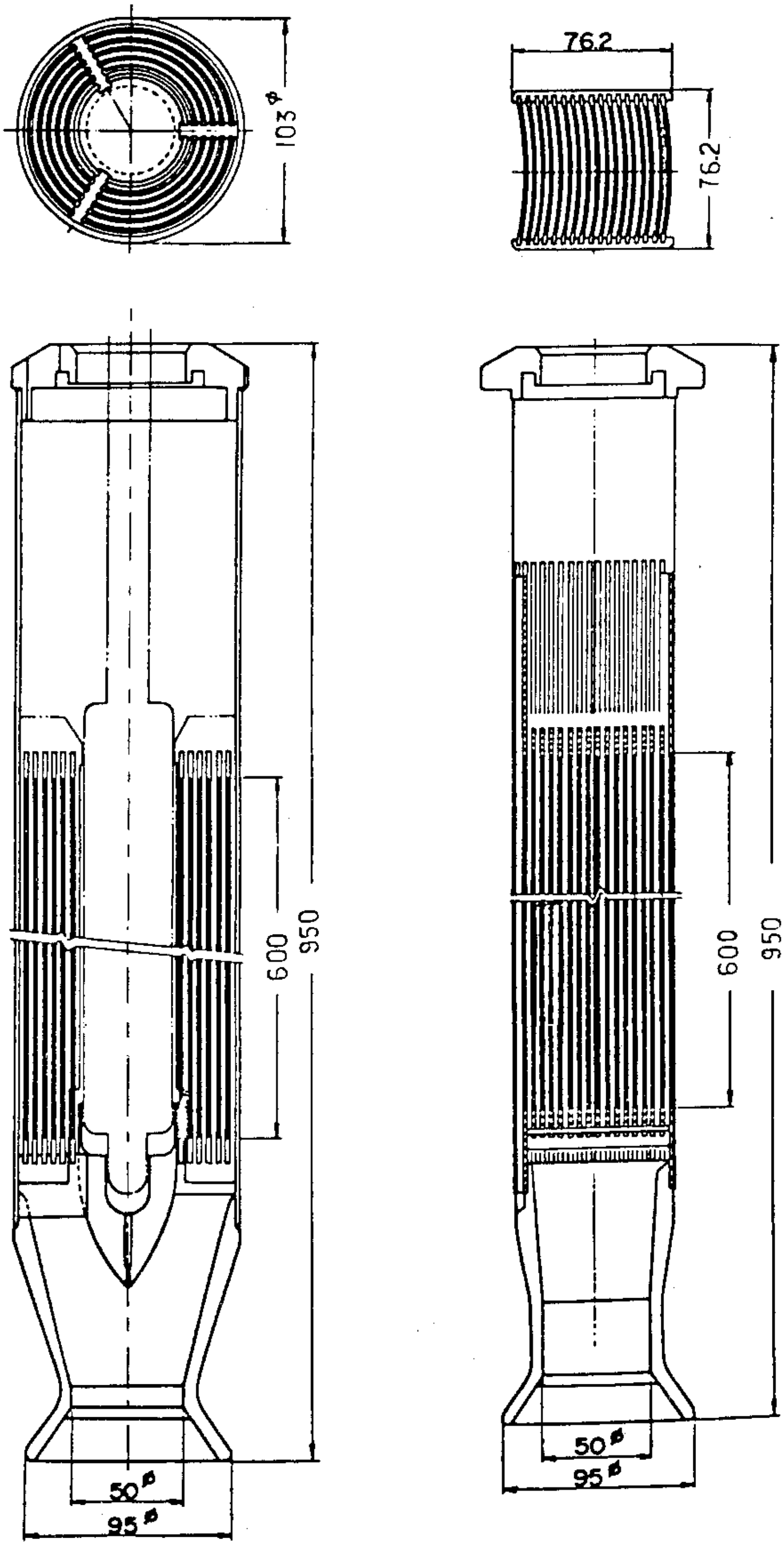


Fig. 18 JRR-2 Core Configuration



Cylindrical Type

MTR Type

Fig.17 JRR-2 Fuel Element

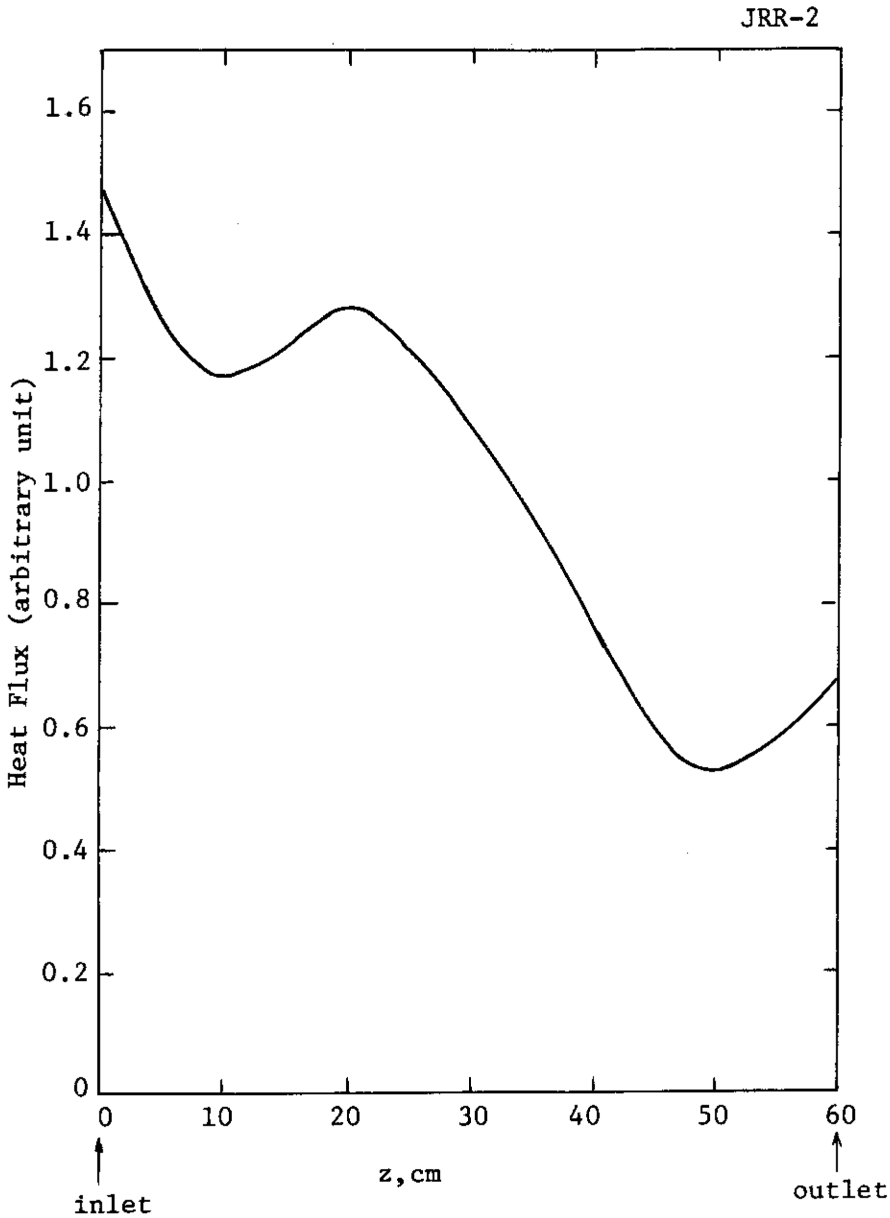


Fig. 19 Heat flux distribution in the JRR-2

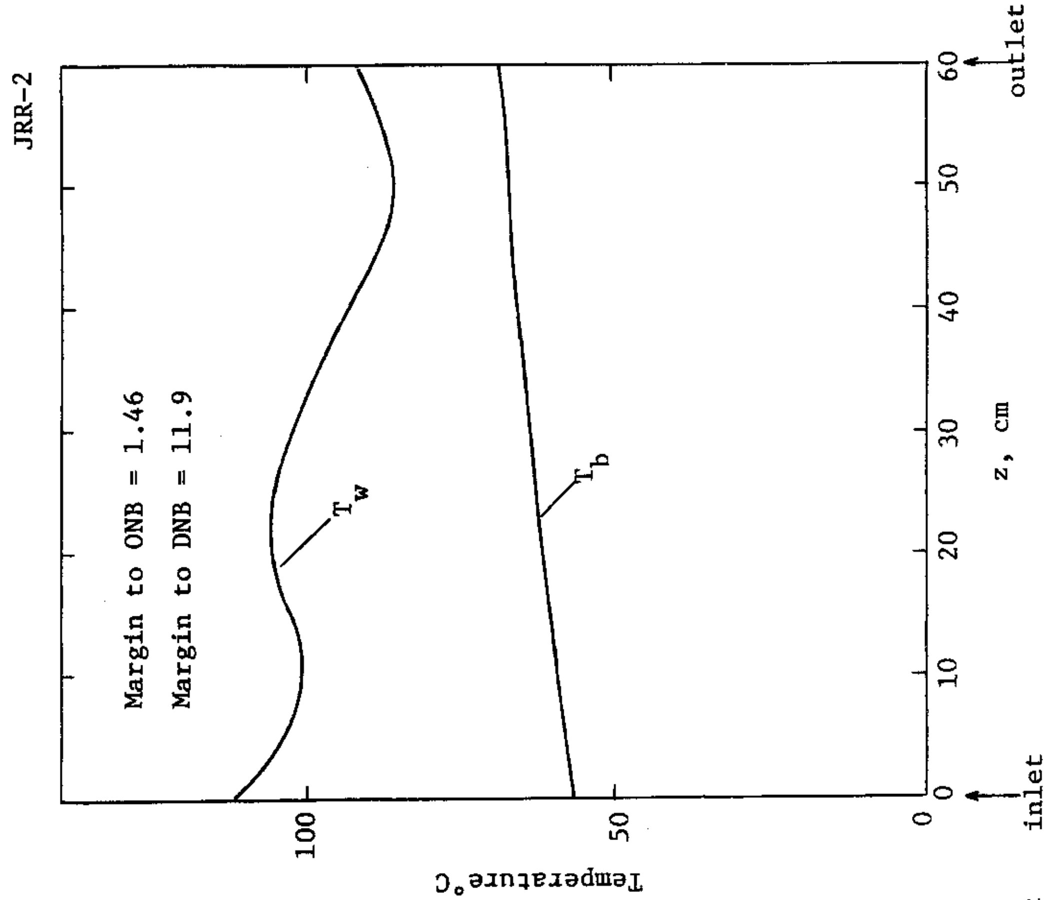


Fig.21 Fuel surface and coolant bulk temperature distributions in the JRR-2 LEU fuel element

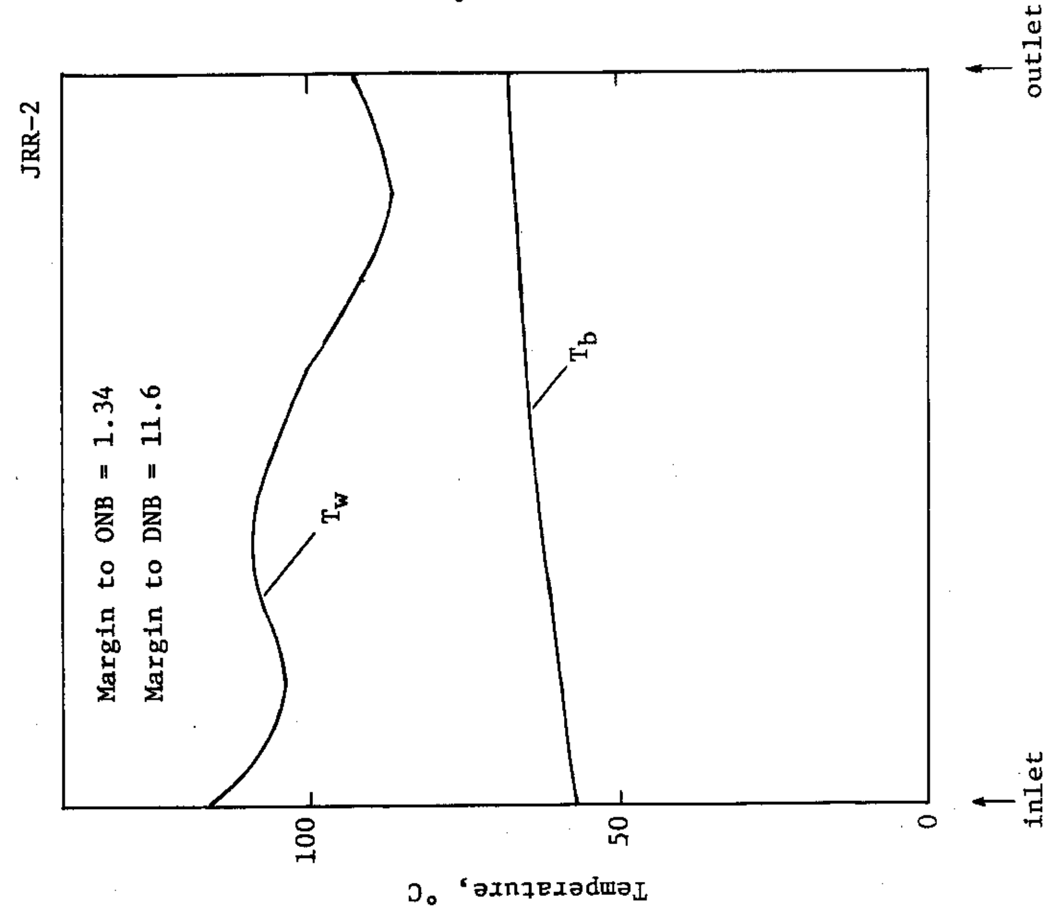


Fig.20 Fuel surface and coolant bulk temperature distributions in the JRR-2 HEU fuel element