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THERMAL-HYDRAULIC CONCEPTUAL DESIGN OF THE MULTIPLE PURPOSE RESEARCH REACTOR MEX-15

February 1994

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Thermal-hydraulic Conceptual Design of the Multiple Purpose Research Reactor MEX-15

Marco Antonio LUCATERO* and Masanori KAMINAGA

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The Multiple Purpose Research Reactor MEX-15 is a 15 MW thermal power, swimming pool type research reactor and will be constructed by the National Institute of Nuclear Research (ININ) of Mexico. Demineralized light water will be used as coolant and moderator. The reactor core will be surrounding by graphite reflectors. The reactor will use 19.75% enriched U_3O_8-Al plate-type fuel (MTR-type).

The core thermal-hydraulic conceptual design of the MEX-15 was performed for two cooling modes, forced convection cooling and natural convection cooling. The key criteria are first to avoid the nucleate boiling anywhere in the core and second to have enough safety margin to the DNB for normal operation conditions. The results of the thermal-hydraulic conceptual design and analysis show that the optimum coolant velocity in the standard fuel element is about 5.6 m/s with the minimum temperature margin against the ONB temperature of about 17°C and the minimum DNBR of 2.58 for the forced-convection cooling mode at a core power of 15 MW with the pressures of 1.43 kg/cm^2 at the core inlet and a core inlet coolant temperature of 35°C.

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It was also determined that the total core power up to about 300 kW can be removed by the natural convection cooling under the condition that nucleate boiling is not allowed anywhere in the core. The minimum temperature margin against ONB temperature and the minimum DNBR at 300 kW are 1.6°C and 6.31, respectively.

The results obtained in this work establishes the preliminary technical specifications for the core thermal-hydraulic design of the Multiple Purpose Research Reactor MEX-15.

Keywords: DNB, Forced Convection, MEX-15, Natural Convection, ONB, Plate-type Fuel, Research Reactor, Steady-state, Thermal-hydraulic Design

多目的研究炉MEX－15の熱水力概念設計

日本原子力研究所東海研究所研究炉部
Marco Antonio LUCATERO*・神永　雅紀

（1994年1月10日受理）

多目的研究炉MEX－15は、メキシコ国立原子力研究所が建設を計画している熱出力15MWの軽水減速・冷却、黒鉛反射体付きのプール型研究炉である。燃料としては、ウラン濃縮度19.75％のU_3O_8－Alを燃料芯材としたMTR型の板状燃料が使用される予定である。

本報告書は、MEX－15の熱水力概念設計について述べたものであり、強制循環冷却及び自然循環冷却の2つの冷却モードについて検討した。設計では、2つの基本方針、すなわち、炉心のいかなる場所においても沸騰を許さない、定常運転状態においてDNBに対して十分な余裕を持つこと、を設定した。熱水力解析結果から、定格出力15MW、炉心入口圧力1.43kg/cm^2、炉心入口温度35℃の強制循環冷却時において、標準型燃料要素に対する最適冷却材流速は約5.6m/sであり、その時の沸騰開始（ONB）温度に対する余裕は約17℃、最小DNBRは2.58であることが明らかとなった。

また、自然循環冷却時においては、熱出力約300kWまでは、炉心のいかなる場所でも沸騰を許さず運転が可能なことが明らかとなった。300kW時のONB温度に対する余裕は1.6℃、最小DNBRは6.31である。

本解析結果は、多目的研究炉MEX－15の暫定的な技術的仕様の作成に用いられる。

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1. INTRODUCTION

Department of Nuclear Systems of the National Institute of Nuclear Research (ININ) of Mexico has been developing the conceptual design of the Multiple Purpose Research Reactor MEX-15. The steady state thermal-hydraulic conceptual design of the reactor core is an important part as well as the neutronic conceptual design.

MEX-15 is a 15 MW thermal power, swimming pool type research reactor. Demineralized light water will be used as coolant and moderator. The reactor core will be surrounding by graphite reflectors and it will be surrounding by light water. The reactor will use 19.75% enriched U_3O_8-Al plate-type fuel (MTR-type). MEX-15 will have two operation modes, one is forced-convection cooling mode for high power operation and the other is natural-convection cooling mode for low power operation.

Thermal-hydraulic conceptual design of the reactor core was, therefore, performed for two cooling modes, forced convection cooling and natural convection cooling. Two major design criteria applied for the JRR-3M were also set up for the MEX-15, first to avoid nucleate boiling anywhere in the core and the second to have enough safety margin against the Departure from Nucleate Boiling (DNB) for normal operation conditions.

This report presents the preliminary core thermal hydraulic characteristics and safety margins for the beginning of cycle (BOC) core configuration for four different core inlet pressures with two different core inlet temperatures and various core flow rates. For the steady-state thermal-hydraulic analysis, COOLOD-N code was applied.

At first, the COOLOD-N code was applied to determine the optimum flow rate for the standard fuel element under forced convection cooling mode at the nominal power (15 MW) operation and at the same time, the core thermal hydraulic characteristics and safety margins were determined. Then, the core thermal hydraulic characteristics and safety margins under natural convection cooling mode were determined.

This work have been carried out at Tokai Research Establishment, JAERI under the personnel assignment arrangement between the National Institute of Nuclear Research (ININ) of Mexico and JAERI.

2. DESCRIPTION OF THE REACTOR

The MEX-15 reactor core conceptual design will utilize 19.75% enriched uranium fuel in the form of U_3O_8-Al. The core will comprise of 23 standard MTR type fuel elements and 6 control fuel elements in 6 x 5 patten on a 8 x 7 grid. Each standard fuel element will contain 21 fuel plates and each control fuel element will contain 15 fuel plates plus 4 aluminum dummy plates forming a passage for fork type control absorber. Fuel meat dimensions for the proposed fuel are 63.0 mm x 0.51 mm x 600.0 mm. The cooling channel gap between fuel plates will have a thickness of 2.563 mm. Figure 1 shows the core layout of the MEX-15 research reactor.

The reactor will be operated at a thermal power of 15 MW and will use demineralized light water as coolant. The heat generated in the core will be removed by a primary cooling system. For the core thermal hydraulic conceptual design of MEX-15, two cooling modes are specified, one is a forced-convection cooling mode for high power operation and the other is a natural convection cooling mode for low power operation, in which the core flow direction is upward. As flow direction in the core for forced-convection cooling mode, downward flow is adopted in the view pint of attenuation of ^{16}N. Figure 2 shows a schematic diagram of the primary cooling system.

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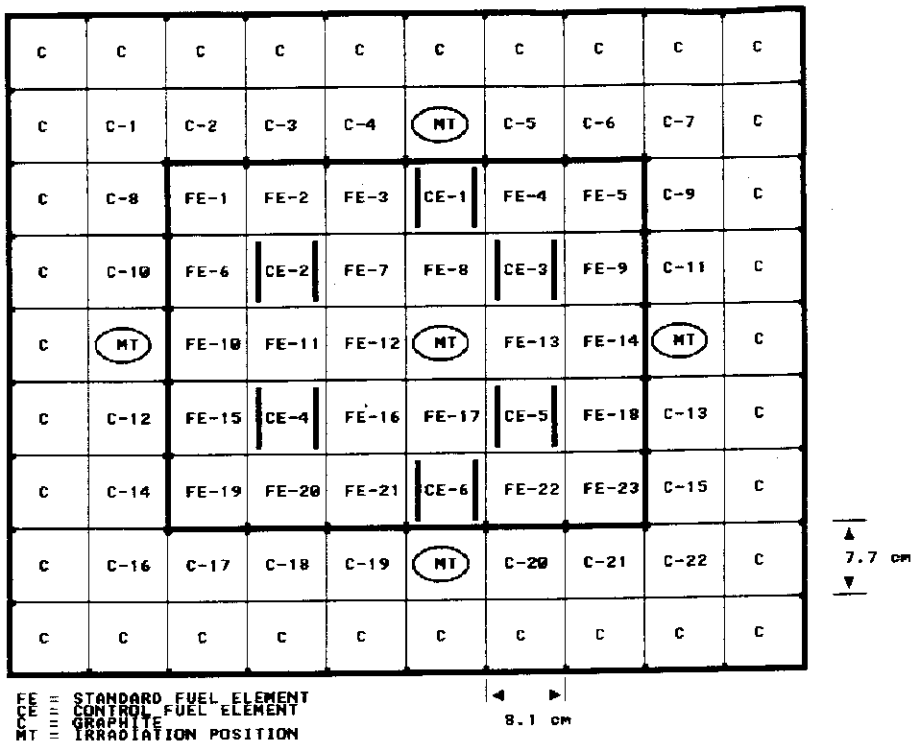

Figure 1 Core layout of the Multiple Purpose Research Reactor MEX-15

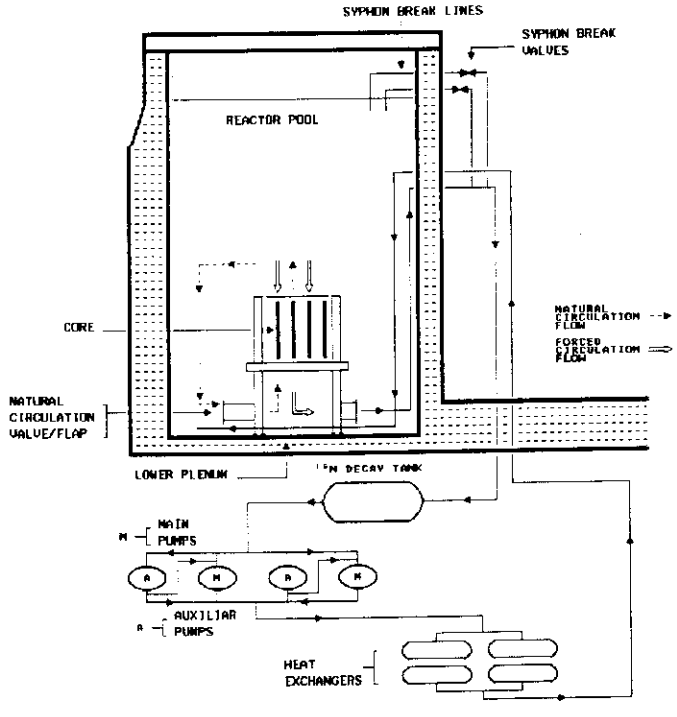

Figure 2 Schematic diagram of the primary cooling system

3. CORE THERMAL HYDRAULIC CONCEPTUAL DESIGN

For the core thermal hydraulic conceptual design of MEX-15, the design bases and the procedure applied for the JRR-3M were also adopted[1].

Two major design criteria were set up for the core thermal hydraulic design of MEX-15 so that fuel plates may have enough safety margins for the conditions of normal operation. One is to avoid nucleate boiling of coolant anywhere in the core in order to give enough allowance against the burnout of the fuel plate even at the hottest spot in the core, to avoid any flow instability induced by partial boiling in the core and obtain stable neutron fluxes for experiments. For this criterion, the allowance in surface temperature of the fuel plate for the onset of nucleate boiling (ONB) temperature was evaluated at the hottest spot in the core, using hot channel model.

The other is to give enough margin against the burnout itself of the fuel plate under the normal operation condition so that there may be enough margin also for operational transients. The departure from nucleate boiling ratio (DNBR) was decided to be not less than 1.5 to meet the latter criterion. Where DNBR is the ratio of the heat flux at the departure from nucleate boiling (DNB) to the maximum heat flux.

With respect to the latter criterion, the following situation should be mentioned here which was a major feature of the JRR-3M reactor core as well as MEX-15 reactor core. Core flow in the forced-convection cooling mode is downward during normal operation and core flow does, therefore, decreases to zero velocity and flow direction reverses to an upward flow when abnormal transients such as loss-of-flow occurred. Downward flow in normal operation condition is more severer than upward flow for heat removal in case of the operational transients or accident conditions. Taking into account the situation described above, DNB heat flux correlations were developed by JAERI, which are applicable to the condition at very low coolant velocities including zero velocity as well as to upward and downward flows at higher velocities[2].

The following design procedure was adopted to the MEX-15 thermal-hydraulic conceptual design.

1. The coolant velocity in the standard fuel element which has the hottest spot is determined to give the maximum temperature margin of the fuel plates against the ONB. At the same time the DNBR is calculated under the condition of coolant velocity thus obtained and it is confirmed that the DNBR is lager than 1.5. A core inlet pressure and a core inlet temperature are selected as design parameters.

2. The coolant velocity in control elements, irradiation holes and so on are calculated so that the pressure drops there my be balanced with that in the standard fuel element. It is then confirmed that coolant velocities determined are enough for cooling each component.

3. The core flow distribution and total core flow rates can be obtained with the coolant velocities determined as described above for each component.

This report does not describe analysis results of coolant velocities in irradiation elements, graphite reflectors and so on. Because up to now, only specifications for the standard fuel elements and the control elements ,core layout (23 standard fuel elements and 6 control fuel elements in 6 x 5 patten on a 8 x 7 grid) and a thermal power (15MW) were fixed. The core flow distribution is, therefore, not determined at this moment. Table 1 shows fuel element specifications for MEX-15.

Table 1 Fuel element specifications

1. Fuel meat:	
- width (mm)	63.00
- length (mm)	600.00
- thickness (mm)	0.51
2. Cladding thickness (mm)	0.38
3. Distance between top of plate and top of fuel meat (mm)	12.50
4. Distance between bottom of fuel meat and bottom of fuel plate (mm)	12.50
5. Fuel plate length (mm)	625.00
6. Cooling channel	
- width (mm)	66.50
- thickness (mm)	2.563

4. COMPUTER CODE USED IN THE CALCULATIONS AND ANALYSIS

The thermal hydraulic calculations and analysis were carried out using the computer code COOLOD-N[3]. The COOLOD-N calculates local bulk temperature of coolant and the fuel plate surface temperature as a function of a distance from inlet of subchannel with calculations of local pressures, local heat fluxes, local coolant velocities and an outlet coolant temperature of the subchannel in which the hottest spot exists. It also calculates the ONB temperature, the DNB flux, saturation temperature and saturation pressure of coolant at every distance from the inlet of subchannel, to evaluate the temperature margin against the ONB and heat flux margin against the DNB along the fuel plate which includes the hottest spot. The COOLOD[4] code has been developed by JAERI for the steady-state thermal hydraulic analysis of research reactors in which plate-type fuel is employed. Thermal hydraulic analysis of JRR-2, JRR-3M and JRR-4 and so on have been performed using the COOLOD code. The COOLOD-N code is a revised version of COOLOD code. In the COOLOD-N code a function to calculate flow rates under a natural circulation cooling mode and a "Heat Transfer Package"[5] which was especially developed for research reactors which are operated under low pressure and low temperature conditions using plate-type fuel, based on heat transfer experiments have been newly added to the COOLOD code.

5 CALCULATION MODEL AND INPUT DATA

5.1 Temperature Calculation Model

Fuel plates temperatures are calculated considering one dimensional heat conduction in radial direction of fuel plates and assuming that the heat generation in fuel meat is constant along the radial direction.

5.1.1 Temperature calculation model for forced convection cooling

The temperature calculation model for forced convection cooling mode is shown in the Figure 3. In this calculation model, a coolant flow rate is given by input data. Fuel plates and coolant channel are divided into 15 segments in axial direction. The coolant is heated from both sides, and the flow direction is downward.

5.1.2. Temperature calculation model for natural circulation cooling

Figure 4 shows a temperature calculation model for a natural circulation cooling mode. In this calculation model, a coolant flow rate is calculated by the COOLOD-N code. In the natural circulation

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cooling, the core flow direction is upward, which supplied by the downward flow through the natural circulation flaps, as shown in Figure 4. The driving force for the natural circulation is calculated by the difference between the outlet water density of the core flow heated by the core power and the inlet water density through the natural circulation flaps[3]. Fuel plates and coolant channel are divided into 15 segments in axial direction, as well as forced convection cooling mode. The coolant is heated from both sides by the fuel plates.

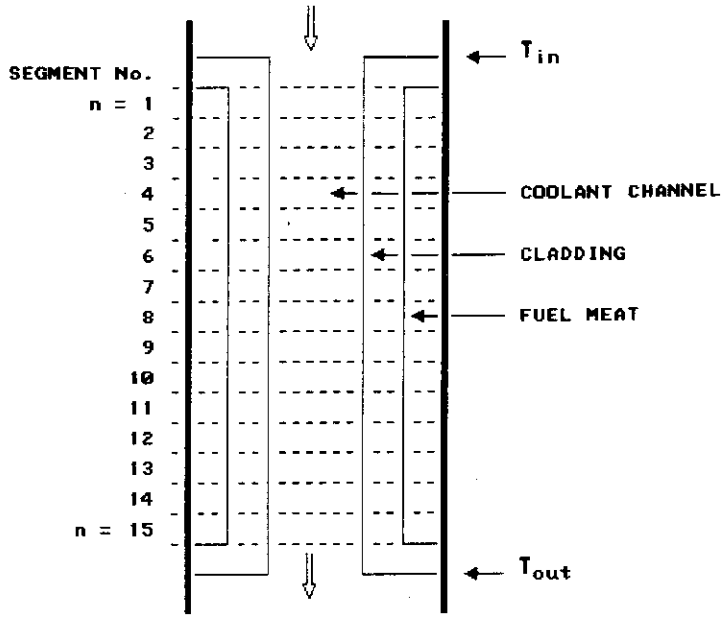


Figure 3 Temperature calculation model under forced convection cooling

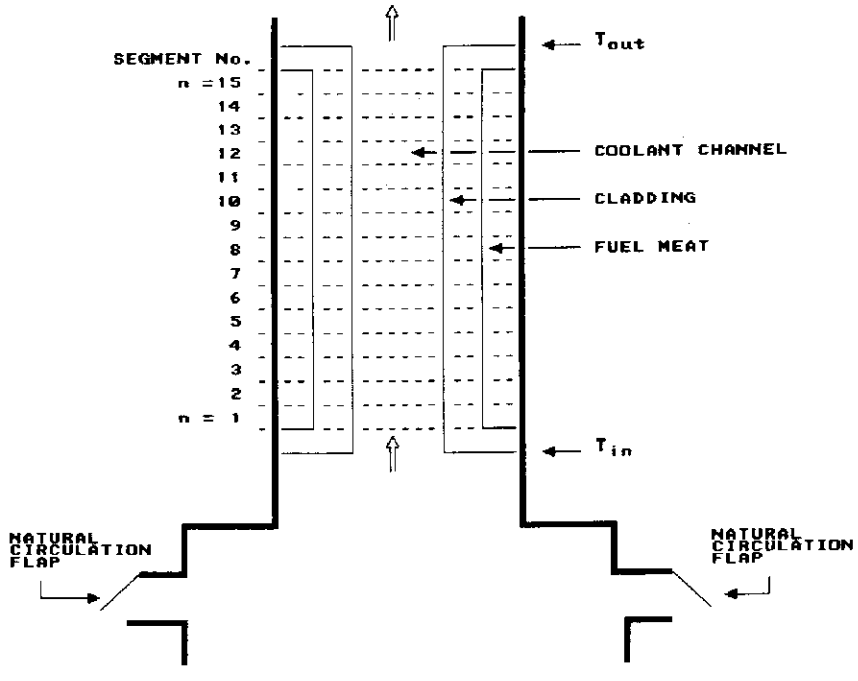


Figure 4 Temperature calculation model under natural circulation cooling

5.2 Pressure Drop Calculation Model

5.2.1. Pressure drop calculation model for forced convection cooling

Pressure drop calculation model of the fuel element for a forced convection cooling mode is shown in Figure 5. The distance between the top (49.5 mm) and the top of the fuel plates region and the distance between the bottom (46.0 mm) and the bottom of the fuel plates region were taken equal to the fuel element of RSG-GAS of Indonesia[6], because, MEX-15's fuel element specification is based on the standard fuel element of the RSG-GAS. The length of 149.5 mm of the fuel element nozzle was taken equal to the fuel element of the RSG-GAS, too. Its geometrical form and dimensions of 65.0 mm x 62.17 mm were chosen in basis to calculation of the pressure at fuel element nozzle inlet, pressure loss and coolant velocity at nozzle. The pressures of 14, 15, 16 and 17 bar abs. at core inlet were used for these calculations. The calculation results are shown in Appendix A. Table 2 shows flow area, equivalent hydraulic diameter, Cb factor and resistance coefficient due to geometry change (contraction and expansion) of each region of the fuel element.

Table 2 Pressure drop calculation data for forced convection cooling (Fuel element)

Region No. (see Fig. 5)	Flow area (cm^2)	Equivalent hydraulic diameter (cm)	Factor Cb*	Resistance coefficient due to geometry change
1	53.53	7.28	0.0	0.5
2	53.53	7.28	96.0	-
3	53.53	7.28	96.0	-
4	40.41	6.36	0.0	0.123
5	40.41	6.36	64.0	-
6	40.41	6.36	0.0	1.0

* Factor Cb is used to calculate the fraction loss coefficient under laminar flow

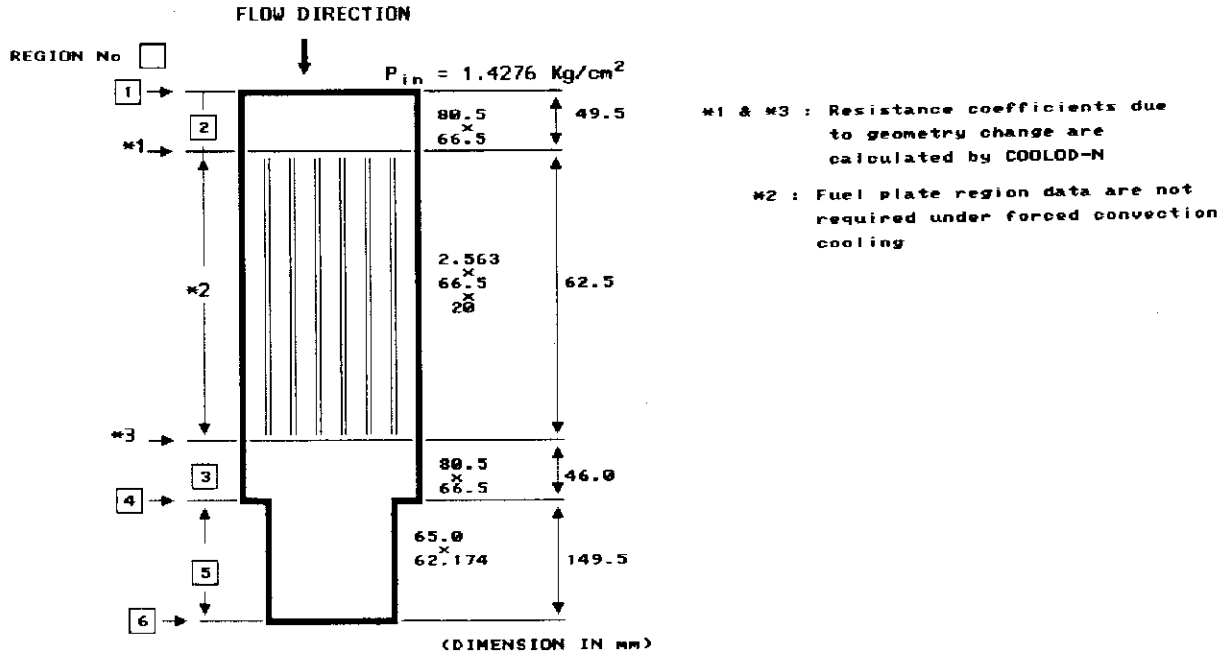


Figure 5 Pressure drop calculation model under forced convection cooling

5.2.2. Pressure drop calculation model for natural circulation cooling

Figure 6 shows the pressure drop calculation models of the fuel element for a natural circulation cooling mode and that of the natural circulation flap. The pressure of 1.5143 kg/cm^2 abs. at core inlet was used in all the calculations. Table 3 and Table 4 show the flow area, equivalent hydraulic diameter, Cb factor and resistance coefficient due to geometry change of each region of the fuel element and the natural circulation flap, respectively.

Table 3 Pressure drop calculation data for natural circulation cooling (Fuel element)

Region No. (see Fig. 6)	Flow area (cm^2)	Equivalent hydraulic diameter (cm)	Factor Cb	Resistance coefficient due to geometry change
1	40.41	6.36	-	0.5
2	40.41	6.36	64.0	-
3	40.41	6.36	-	0.123
4	53.53	7.28	90.0	-
5	34.09	0.49	96.0	-
6	53.53	7.28	90.0	-
7	53.53	7.28	-	1.0

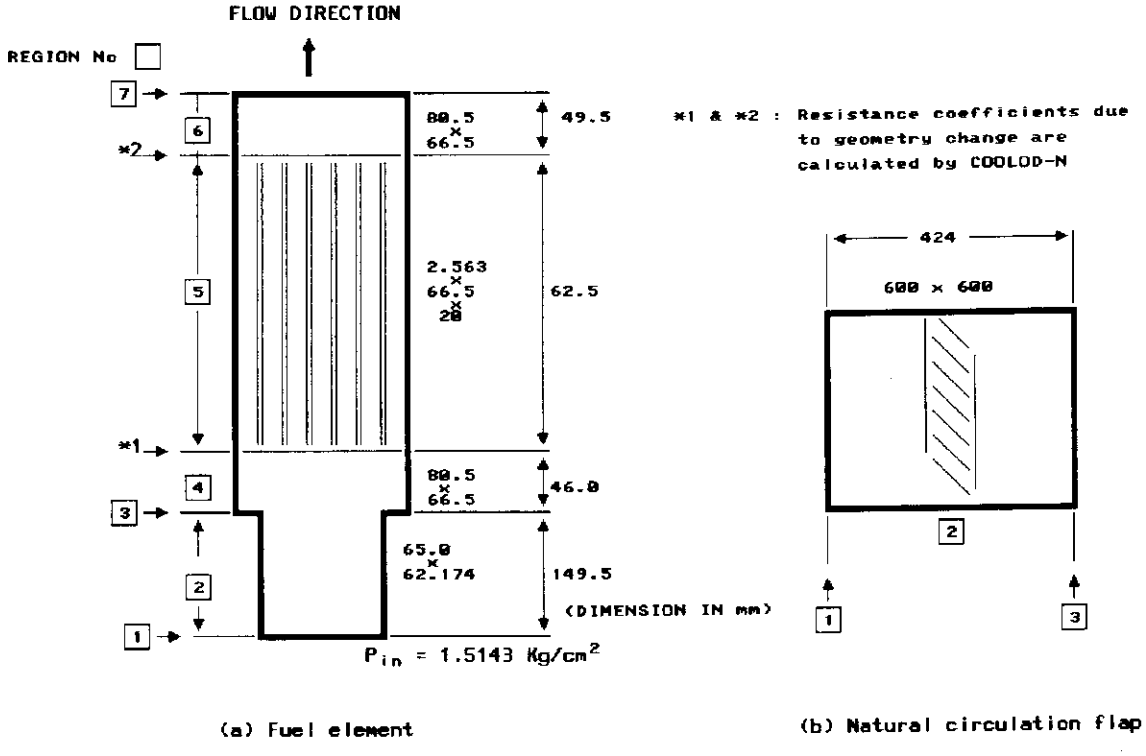


Figure 6 Pressure drop calculation model under natural circulation cooling

Table 4 Pressure drop calculation data for natural circulation cooling (Fuel element)

Region No. (see Fig. 6)	Flow area (cm^2)	Equivalent hydraulic diameter (cm)	Factor Cb	Resistance coefficient due to geometry change
1	3600	60.0	-	0.5
2	3600	60.0	56.9	-
3	3600	60.0	-	1.0

In the above pressure drop calculation models, following correlation were adopted to calculate pressure loss coefficients due to geometry change:

a. Pressure loss coefficient due to geometry change.- Contraction (Large area to small area)

$$\xi = 0.5\left[1.0-(A_S/A_L)\right]$$
$$= 0.5\left[1.0-(V_L/V_S)\right]$$

b. Pressure loss coefficients due to geometry change.- Expansion (Small area to large area)

$$\xi = \left[1.0-(A_S/A_L)\right]^2$$
$$= \left[1.0-(V_L/V_S)\right]^2$$

Where ξ : Pressure loss coefficient due to geometry change,
A_S/A_L : Ratio of large area to small area, and
V_L/V_S : Ratio of velocity in small area to velocity in large area.

5.3 Major Input Data

5.3.1 Radial and axial peaking factors

The radial peaking factors were obtained from neutronic calculation using the CITATION code[7] and the maximum radial peaking factor is at the position of standard fuel element SE-12, and this position was selected as the hottest channel. Figure 7 shows radial peaking factor calculated for BOC core configuration.

Axial peaking factors in fuel elements were also determined by neutronic calculation. The axial peaking factors were calculated based on the power distribution at the fueled region SE-12 of the BOC core configuration. The Table 5 shows the axial peaking factors used in the thermal hydraulic calculations and the Figure 8 shows the axial peaking factors distribution.

5.3.2 Flow rate through fuel element

The performances of the proposed core under 15 MW operating condition, were evaluated under various flow rates through core ranging from 188.2 kg/s to 846.9 kg/s, 187.9 kg/s to 845.1 kg/s and 187.1 kg/s to 842.2 kg/s which correspond to the core inlet temperatures of 20 °C, 30 °C and 40 °C, respectively, to determine optimum effective flow rate through the standard fuel elements.

For the thermal hydraulic calculations, it was assumed that the flow rate through standard and control fuel elements to be the same and it was also assumed that the effective flow rate through the core to be produced only by the flow rates through all standard and control fuel elements in the core.

RADIAL POWER PEAKING FACTORS DISTRIBUTION, BOC CORE

C	C	C	C	C	C	C	C	C	C
C	C-1	C-2	C-3	C-4	MT	C-5	C-6	C-7	C
C	C-8	(4) FE-1 0.7808	(3) FE-2 0.9469	(3) FE-3 1.0391	(5) CE-1 .884	(3) FE-4 0.8964	(4) FE-5 0.7072	C-9	C
C	C-10	(3) FE-6 0.9353	(5) CE-2 .957	(2) FE-7 1.2566	(2) FE-8 1.3913	(5) CE-3 .894	(4) FE-9 0.8387	C-11	C
C	MT	(3) FE-10 1.0227	(2) FE-11 1.2295	(2)(1) FE-12 1.4855	MT	(2) FE-13 1.2909	(3) FE-14 0.9171	MT	C
C	C-12	(3) FE-15 0.9353	(5) CE-4 .957	(2) FE-16 1.2566	(2) FE-17 1.3913	(5) CE-5 .894	(4) FE-18 0.8387	C-13	C
C	C-14	(4) FE-19 0.7808	(4) FE-20 0.9469	(3) FE-21 1.0391	(5) CE-6 .884	(4) FE-22 0.8964	(4) FE-23 0.7072	C-15	C
C	C-16	C-17	C-18	C-19	MT	C-20	C-21	C-22	C
C	C	C	C	C	C	C	C	C	C

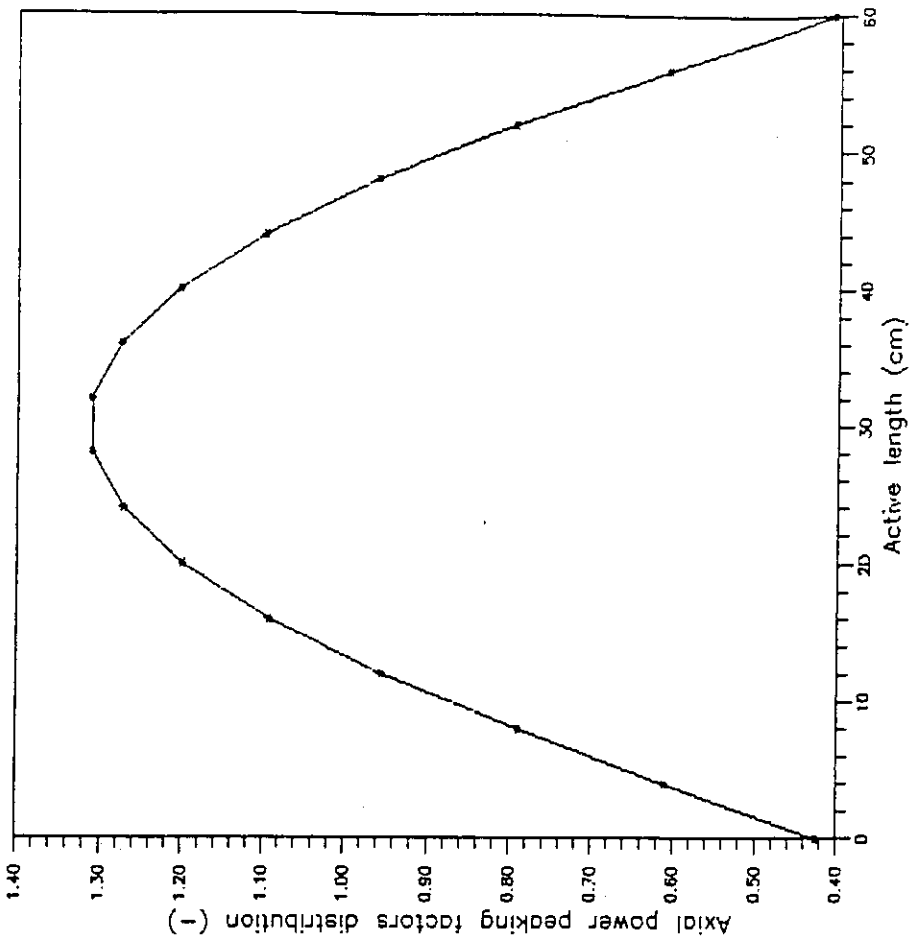
FE = STANDARD FUEL ELEMENT, CE = CONTROL FUEL ELEMENT
C = GRAPHITE, MT = IRRADIATION POSITION
( ) Channel number for the EUREKA-2 analysis model

Figure 7 Radial peaking factors distribution at BOC core configuration

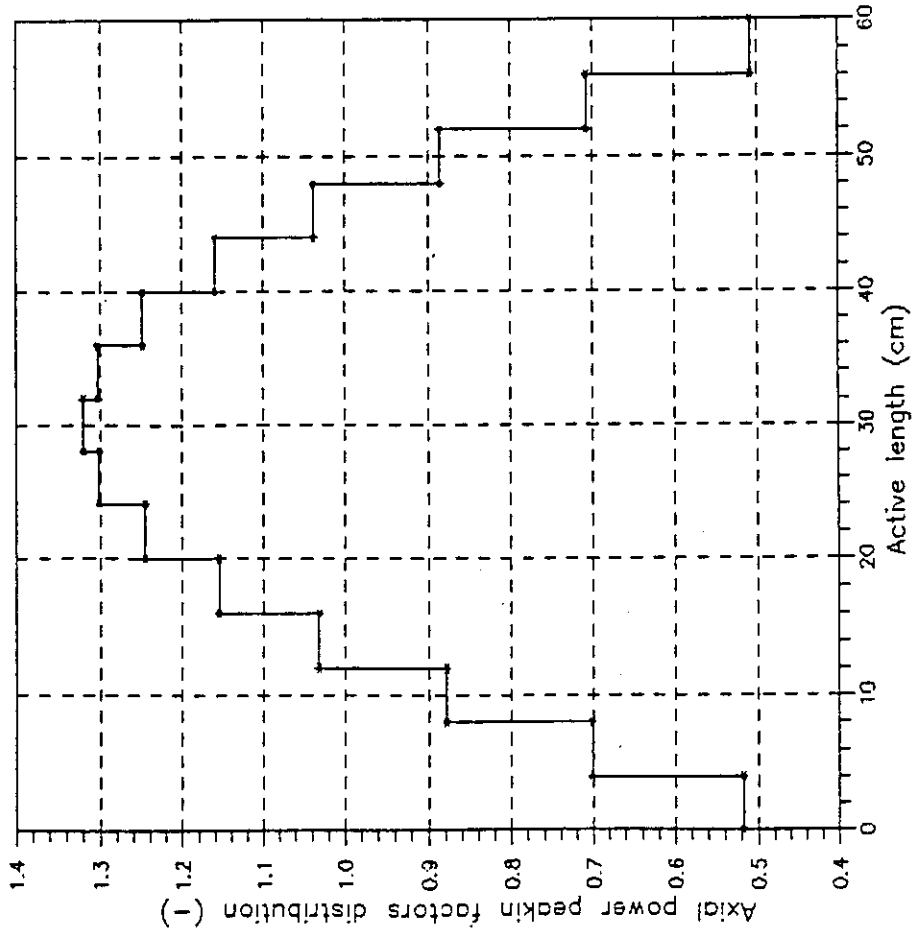
Table 5 Axial peaking factors

Segment No.	Axial power peaking factor*	Point No.	Axial power peaking factor**
-	-	1	0.426
1	0.518	2	0.610
2	0.701	3	0.790
3	0.873	4	0.955
4	1.031	5	1.093
5	1.155	6	1.200
6	1.246	7	1.273
7	1.301	8	1.311
8	1.320	9	1.312
9	1.303	10	1.276
10	1.249	11	1.204
11	1.160	12	1.099
12	1.038	13	0.962
13	0.886	14	0.796
14	0.707	15	0.608
15	0.508	16	0.408

* From neutronic calculations[7]
** Calculated by COOLOD-N Code



b) For thermal hydraulic calculation (For COOLOD-N)



a) From neutronic calculation

Figure 8 Axial peaking factors distribution used in the thermal hydraulic calculations

5.3.3 Temperature and pressure at the core inlet

Pressures of 1.43 kg/cm^2 abs., 1.53 kg/cm^2 abs., 1.63 kg/cm^2 abs. and 1.73 kg/cm^2 abs. and the temperatures of 20 °C, 30 °C and 40 °C at the core inlet were used in the calculation under the forced convection cooling mode. On the other hand, a pressure (at the core bottom) of 1.51 kg/cm^2 abs. at the core inlet and the core inlet temperature of 35°C was used in the calculations under natural circulation cooling mode.

5.3.4 Thermal power and hot channel factors

The thermal power was fixed to 15 MW by the neutronic calculations. For these thermal hydraulic calculations, it is assumed that all the thermal power is produced only by the fuel elements. The average power generated per fuel element was calculated as the ratio of the net heat generated in the core to number of effective fuel elements in the core. In Table 6 shows the net heat generated in the core and the average power generated per fuel element. The number of effective fuel elements in the core was calculated as follows:

$$N_{eff} = \frac{N_s N_{ps} + N_c N_{pc}}{N_{ps}}$$

Where N_{eff} : Number of effective fuel elements
N_s : Number of standard fuel elements
N_c : Number of control fuel elements
N_{ps} : Number of fuel plates in the standard fuel element
N_{pc} : Number of fuel plates in the control fuel element

The number of effective fuel elements used in the calculation is shown in Table 7.

The nuclear factors, radial (F_R) and local (F_L) were determined by the neutronic calculations. The uncertainty nuclear factor (F_E) was not calculated and it was taken equal to the neutronic design of the upgraded JRR-3 reactor[1]. The engineering hot channel factors which are determined with manufacturing tolerance of the fuel element, error of heat transfer correlation adopted and uncertainties of parameter affecting the bulk temperature rise of coolant and the film temperature rise, were assumed equal to engineering hot channel factors of the RSG-GAS reactor[6]. Table 8 shows the nuclear factors and engineering hot channel factors used in the calculations.

Table 6 Thermal power, net heat generated in the core and average power generated per fuel element

Thermal power (MW)	15.0
Net heat generatedin the core (MW)	15.0
Average power generated per fuel element (MW)	0.550

Table 7 Number of fuel elements in the core

Number of fuel elements:	
- standard fuel	23
- control fuel	6
Number of effective fuel elements in the core	27.29

Table 8 Hot channel factors

Nuclear factors:	
Radial (F_R)	1.485
Local (F_L)	1.499
Uncertainty (F_E)	1.180
Engineering hot channel factors:	
for coolant temp. rise (F_b)	1.167
for film temp. rise (F_f)	1.200

5.3.5 Thermal conductivities of fuel meat and cladding

Table 9 shows the thermal conductivities of the fuel meat and cladding.

Table 9 Thermal conductivities

	Fuel meat material (U_3O_8+Al)	Cladding (Al)
Thermal conductivity (W/cm K)	0.13	1.80

6. RESULTS AND DISCUSSION

Thermal hydraulic calculations were carried out for the forced convection cooling mode at the reactor thermal power of 15 MW with four different core inlet pressures of 1.43, 1.53, 1.63 and 1.73 kg/cm^2 abs. and two different core inlet temperatures of 20 and 40°C under various flow rates through the core ranging from 188.2 kg/s to 846.9 kg/s. Calculation results of the fuel plate surface temperature (T_W) and ONB temperature (T_{ONB}) at the hottest spot where the difference between T_{ONB} and T_W is a minimum in the hot channel at the core inlet temperature of 20 °C, are given in Figure 9 as a function of coolant velocity taking the core inlet pressure as a parameter and Figure 10 shows calculation results of those at the core inlet temperature of 40 °C. These temperatures are shown as a function of the coolant velocity because the coolant velocity is only the dominant variable to fuel plate surface temperature, once the pressure and temperature at the core inlet are fixed. Table 10 shows the coolant velocities range which the T_{ONB} is higher than T_W.

In the ranges of coolant velocities shown in Table 10, no boiling occurs in the hot channel. If the coolant velocities are greater than approximately 8.7 to 9.2 m/s, however, the pressure at fuel element nozzle becomes negative. On the other hand, two-phase flow occurs with nucleate boiling at coolant velocities less than approximately 3.2 m/s and 3.8 m/s for the core inlet pressure of 1.43 kg/cm^2 with the core inlet temperatures of 20 °C and 40 °C, respectively, 3.2 m/s and 3.5 m/s for the core inlet pressure of 1.53 kg/cm^2 with core inlet temperatures of 20 °C and 40 °C, respectively, 3.0 m/s and 3.4 m/s for the core inlet pressure of 1.63 kg/cm^2 with the core inlet temperatures of 20 °C and 40 °C, respectively, 3.0 m/s and 3.4 m/s for the core inlet pressure of 1.73 kg/cm^2 with the core inlet temperatures of 20 °C and 40 °C, respectively.

The coolant saturation temperature, the ONB temperature (T_{ONB}) and fuel plate surface temperature as a function of the coolant velocity are shown in Figure 11 through Figure 14 for each of the four core inlet pressures with the core inlet temperatures of 20 °C and 40 °C, respectively. In these figures, it can be observed that the coolant saturation temperature and T_{ONB} decrease with an increase of the coolant velocity because an increase in the coolant velocity gives lower local pressure according to the increase of pressure loss. On the other hand, the fuel plate surface temperature decreases with an increase of the coolant velocity.

Figure 15 shows the characteristics of the pressure at the fuel plates exit as a function of coolant velocity, taking the core inlet pressure and temperature as parameters. The pressure at the fuel plate exit decreases with an increase of the coolant velocity.

The temperature margin against the ONB as a function of the coolant velocity is shown in Figure 16 for each of the four pressures at the core inlet with core inlet temperatures of 20 °C and 40 °C, respectively. Figure 16 shows that an increase of the coolant velocity gives a higher temperature margin against the ONB temperature ($\Delta T_{ONB} = T_{ONB} - T_W$).

Figure 17 shows the minimum safety margin against DNB and OFI, i.e., DNBR and OFIR as a function of the coolant velocity. In Figure 17, it can be observed that an increase of the coolant velocity gives higher safety margins of DNBR and OFIR. Table 11 shows the ranges of minimum DNBR and minimum OFIR corresponding to the ranges of coolant velocity for each of the four pressures at the core inlet with core inlet temperatures of 20 °C and 40 °C, respectively.

The pumping power required to remove the heat flux produced only by the fuel elements can be calculated[8] as follows:

$$\textit{Pumping power} = \textit{Pressure drop} \times \textit{Volume flow rate}$$

Where the pressure drop used in the calculations was equal to total pressure loss per fuel element times the number of effective fuel elements in the core. Figure 18 shows the behavior of the pumping power and the total pressure loss per fuel element as a function of the coolant velocity taking the core inlet temperature as a parameter. In the figure it can be seen that the pumping power increases according to increase of pressure drop and an increase of the coolant velocity.

Table 10 Range of coolant velocities in which $T_{ONB} > T_W$

Core inlet pressure (kg/cm^2 abs.)	Core inlet temperature (°C)	Range of coolant velocities which $T_{ONB} > T_W$ (m/s)
1.43	20	3.56 to 8.12
	40	3.58 to 8.14
1.53	20	3.56 to 8.63
	40	3.58 to 8.64
1.63	20	3.06 to 8.63
	40	3.58 to 9.14
1.73	20	3.06 to 9.14
	40	3.56 to 9.15

Table 11 Minimum safety margins against the DNB and OFI

Pressure (Temperature (°C)) at the core inlet (kg/cm^2)	Range of coolant velocities (m/s)	Range of minimum DNBRs (-)	Range of minimum OFIRs (-)
1.43 (20)	3.56 to 8.12	2.03 to 2.95	1.70 to 3.88
(40)	4.08 to 8.14	1.84 to 2.96	1.50 to 3.00
1.53 (20)	3.56 to 8.63	2.05 to 3.05	1.74 to 4.22
(40)	3.56 to 8.64	1.65 to 3.07	1.35 to 3.28
1.63 (20)	3.06 to 8.63	2.05 to 3.05	1.74 to 4.22
(40)	3.58 to 9.15	1.70 to 3.16	1.39 to 3.56
1.73 (20)	3.06 to 9.14	1.90 to 2.92	1.55 to 3.36
(40)	3.56 to 9.15	1.74 to 3.25	1.42 to 3.65

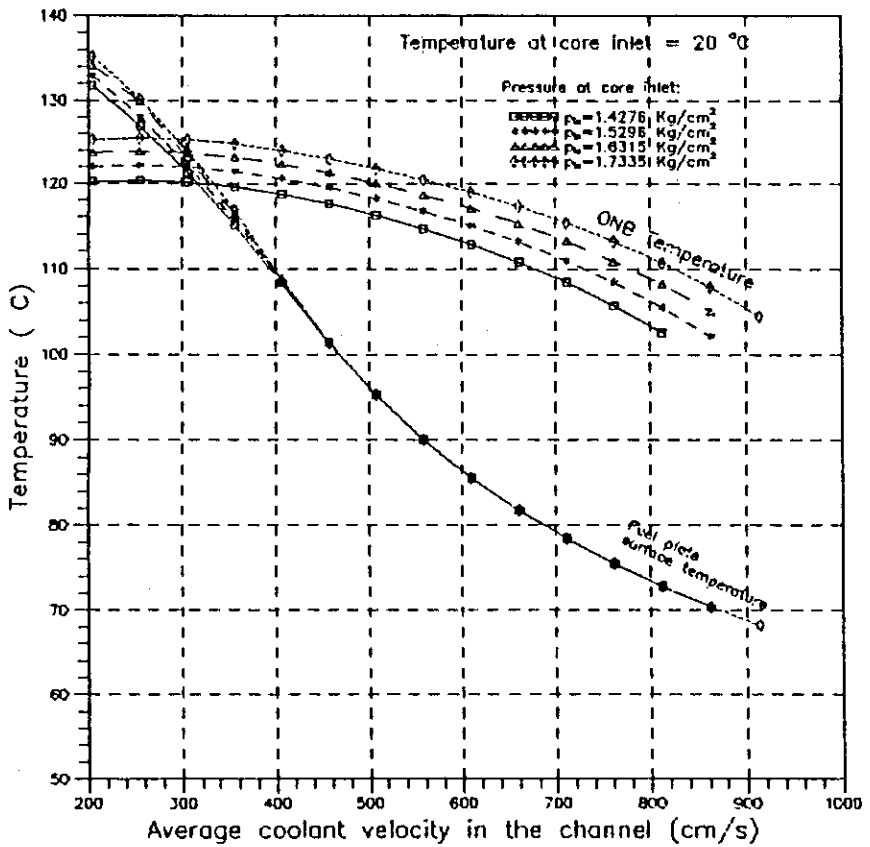


Figure 9 Calculation results of ONB temperature, fuel surface temperature vs. coolant velocity for the standard fuel element for the condition of core inlet coolant temperature of 20 °C at 15 MW of thermal power. (Hot channel)

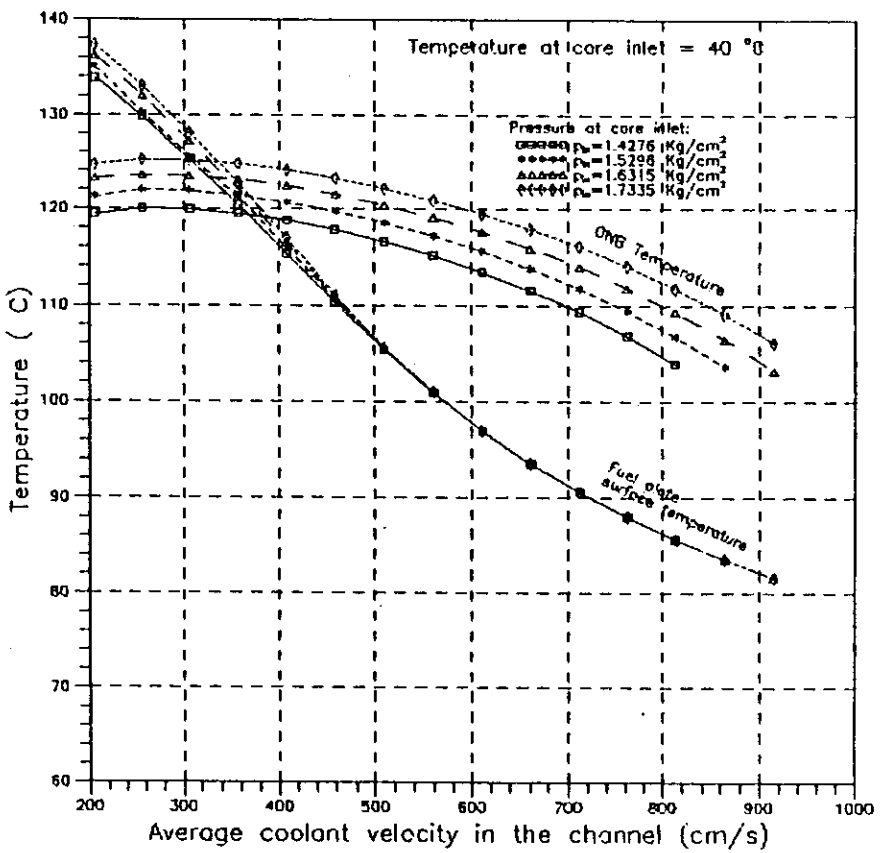


Figure 10 Calculation results of ONB temperature, fuel surface temperature vs. coolant velocity for the standard fuel element for the condition of core inlet coolant temperature of 40 °C at 15 MW of thermal power. (Hot channel)

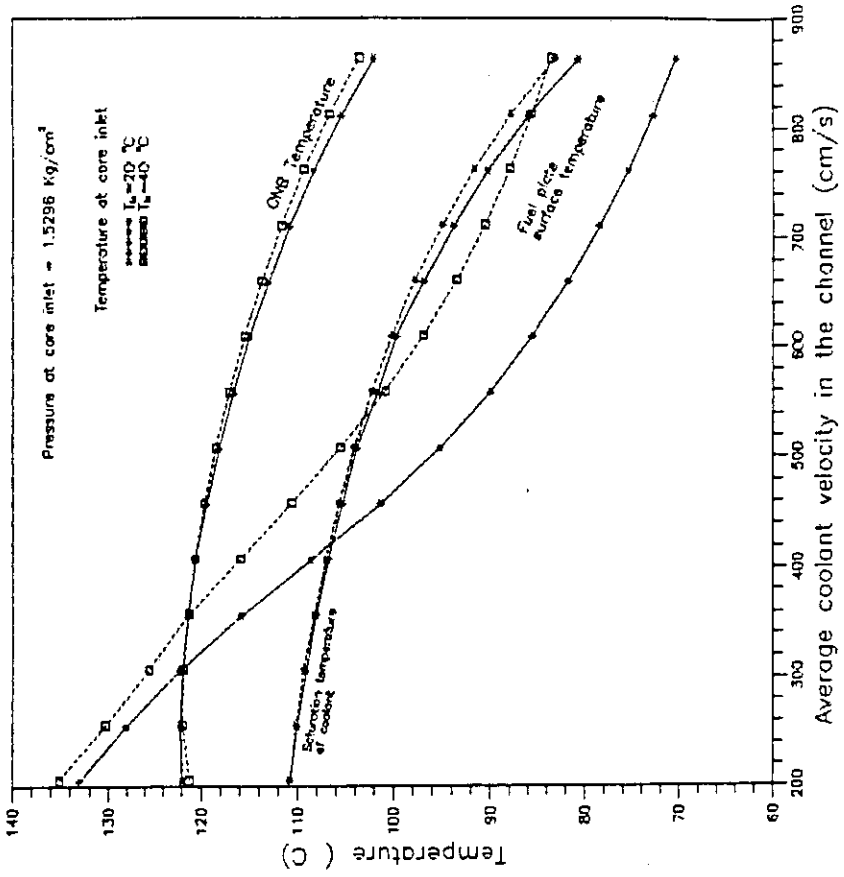


Figure 12 Calculation results of ONB temperature, fuel surface temperature vs. coolant velocity for the standard fuel element for the condition of core inlet pressure of 1.5296 kg/cm^2 at 15 MW of thermal power. (Hot channel)

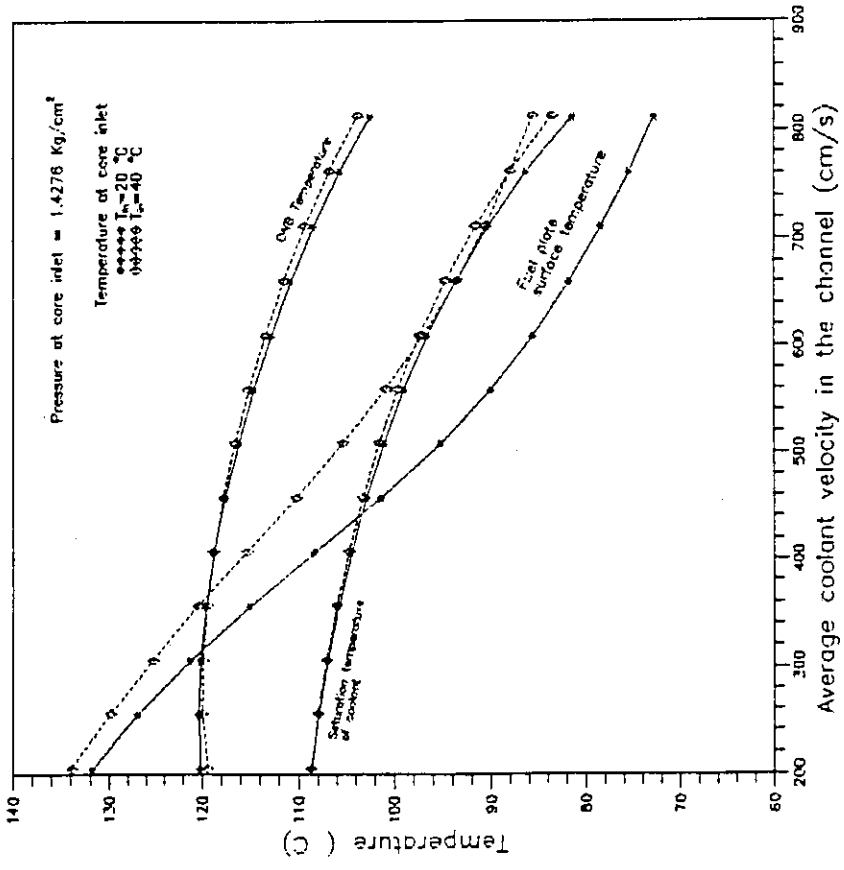


Figure 11 Calculation results of ONB temperature, fuel surface temperature vs. coolant velocity for the standard fuel element for the condition of core inlet pressure of 1.4276 kg/cm^2 at 15 MW of thermal power. (Hot channel)

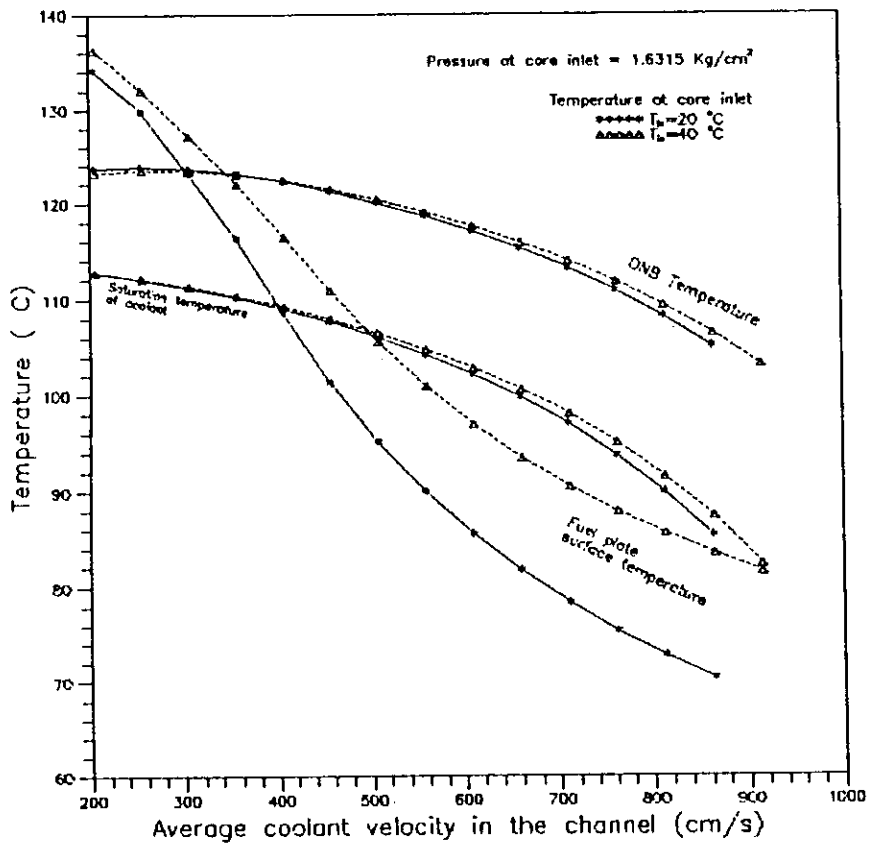


Figure 13 Calculation results of ONB temperature, fuel surface temperature vs. coolant velocity for the standard fuel element for the condition of core inlet pressure of 1.6315 kg/cm^2 at 15 MW of thermal power. (Hot channel)

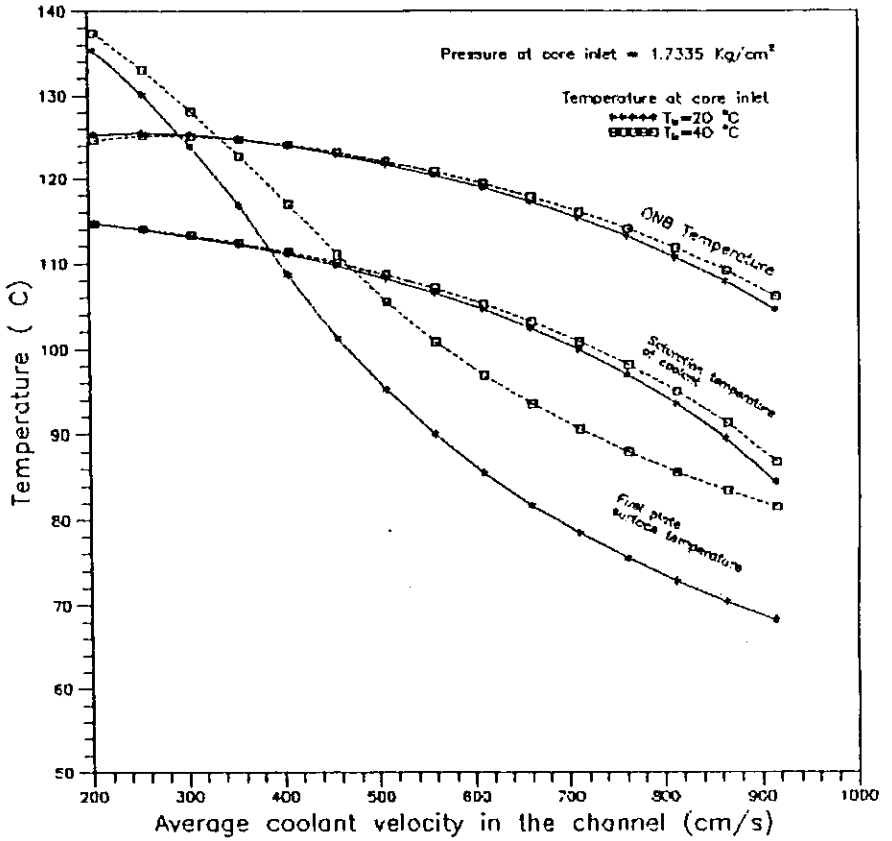


Figure 14 Calculation results of ONB temperature, fuel surface temperature vs. coolant velocity for the standard fuel element for the condition of core inlet pressure of 1.7335 kg/cm^2 at 15 MW of thermal power. (Hot channel)

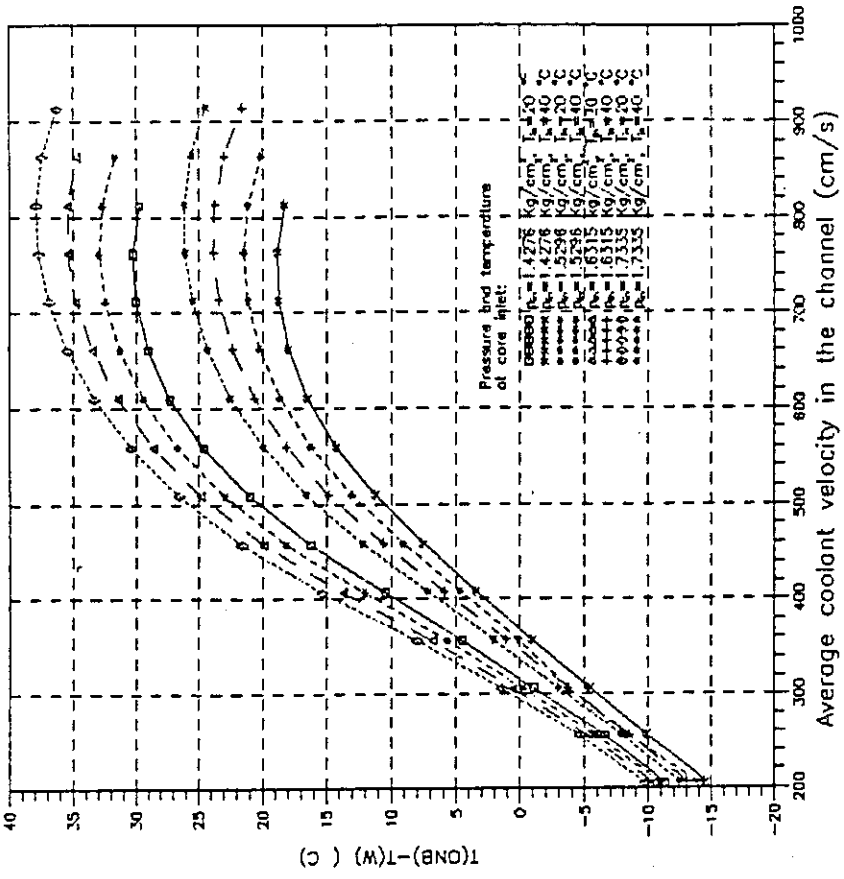


Figure 15 Calculation results of pressure at fuel plate exit vs. coolant velocity for the standard fuel element at 15 MW of thermal power. (Hot channel)

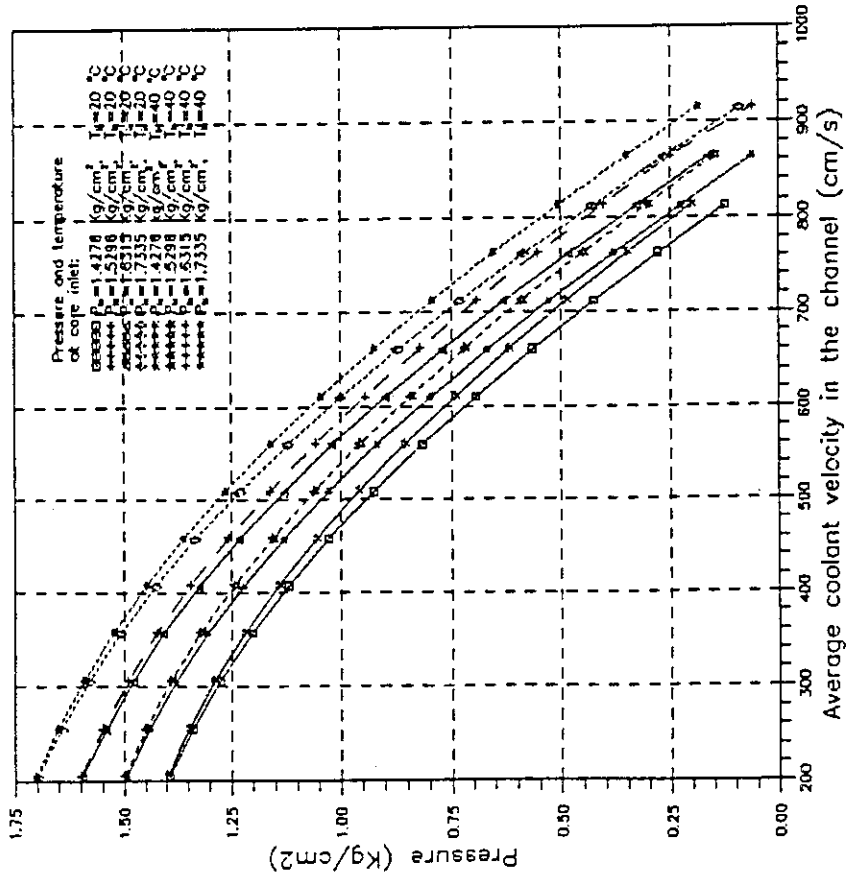


Figure 16 Calculation results of temperature margin against the ONB temperature vs. coolant velocity for the standard fuel element at 15 MW of thermal power. (Hot channel)

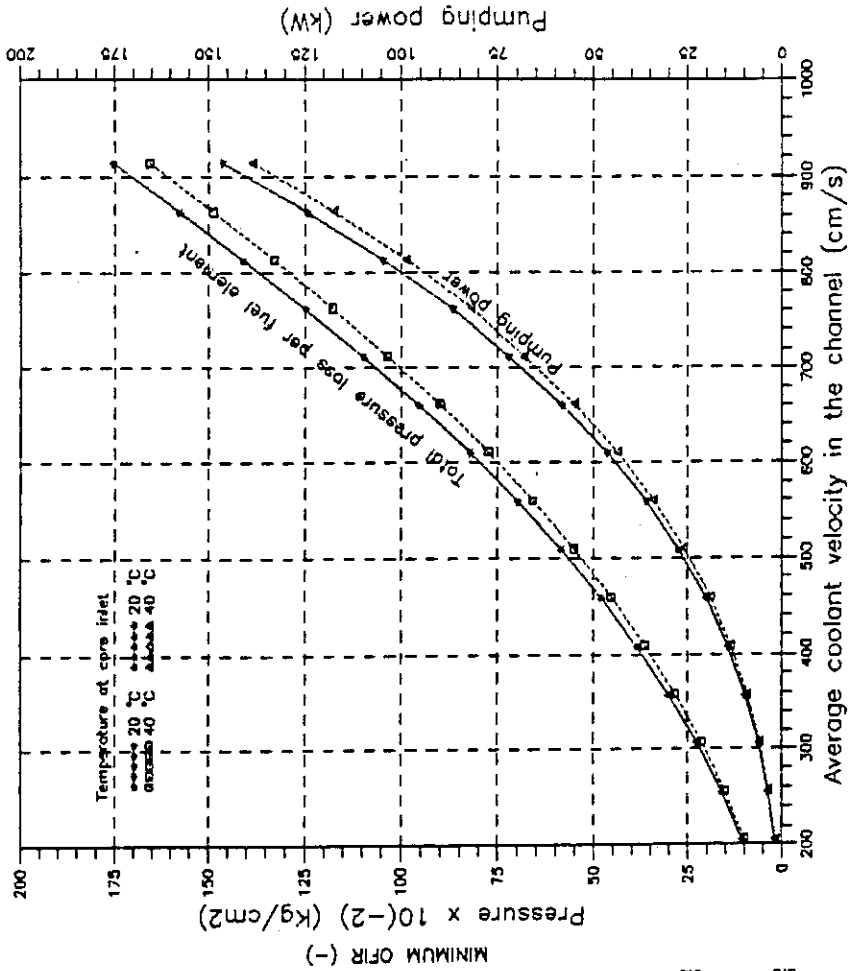


Figure 18 Calculation results of total pressure loss per fuel element and the pumping power vs. coolant velocity at 15 MW of thermal power. (Hot channel)

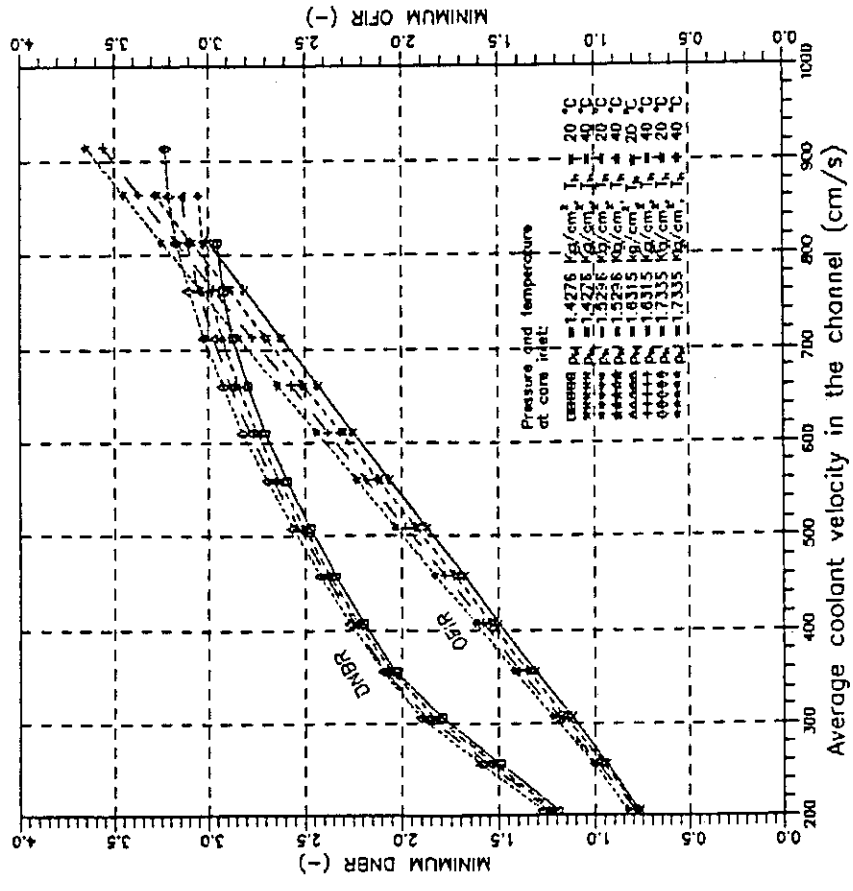


Figure 17 Calculation results of DNBR and OFIR vs. coolant velocity for the standard fuel element at 15 MW of thermal power. (Hot channel)

6.1 Core thermal-hydraulic conceptual design parameters and analysis results for forced convection cooling condition

Generally the research reactors with plate-type fuels are operated under the condition of no nucleate boiling of the coolant anywhere in the core in order to give enough allowance against the burnout of the fuel plate even at the hottest spot in the core, to avoid any flow instability induced by partial boiling in the core and to obtain stable neutron fluxes for experiments.

Onset of nucleate boiling (ONB) is taken as a limit for single-phase cooling and is not a limiting criterion in the design of a fuel element. The heat transfer regime should be clearly identified for proper hydraulic and heat transfer considerations i.e. single-phase flow versus two-phase flow. The nucleate boiling occurs at a wall temperature (T_W) over ONB temperature (T_{ONB})by a quantity T_W - T_{ONB}.

For the reactor design purpose, acceptable data on burnout heat flux are needed since departure from nucleate boiling (DNB) is potentially a limiting design constraint.

The term "Flow Instability" referees to flow oscillations of constant or variable amplitude that are analogous in mechanical system. In this connection the relationship between flow rate and pressure drop plays an important role. Flow oscillations may be aggravated when there is thermohydrodynamic coupling between heat transfer, void, flow pattern and flow rate; however oscillations can occur even when the heat source is held constant.

Flow oscillations are undesirable for several reasons: First, sustained flow oscillations may cause undesirable forced mechanical vibration of components, second, flow oscillations may cause system control problems, which are of particular importance in water cooled reactors where the coolant also acts as moderator; third, flow oscillations affect the local heat transfer characteristics and the boiling crisis. The burnout heat flux under unstable flow conditions may be well below the burnout heat flux under stable flow conditions. Thus for plate type fuel design, the critical heat flux that leads to the onset of flow instability (OFI) may be more limiting than of stable burnout[9].

Operational transients and accidents analysis should be carried out based on steady-state thermal hydraulic calculations results. During an operational transient, DNBR and OFIR will decrease due to a decrease of coolant flow rate and/or an increase of the reactor power. So, the steady-state condition of the research reactors should have enough safety margins.

From the calculation results for the hot channel described before, the coolant velocity of 5.6 m/s is proposed as a conceptual design velocity for the proposed fuel elements. As seen in Figure 16, higer coolant velocity gives higher temperature margin against ONB temperature(ΔT_{ONB}), up to coolant velocity range of 7.2 to 8.0 m/s (corresponding coolant velocity depends on the core inlet pressure as shown in Figure 16).ΔT_{ONB} increases linearly with increase of coolant velocity up to 5.6 m/s for the case of the core inlet pressure of 1.43 kg/cm^2 and after that ΔT_{ONB} increase ratio is decreases because of increase of core pressure drop and has maximum at coolant velocity of about 7.2 m/s. The coolant velocity of 5.6 m/s is the maximum coolant velocity in the linear part of ΔT_{ONB} vs. coolant velocity and that is why 5.6 m/s is selected as the conceptual design velocity for the standard fuel elements. The coolant velocity of 5.6 m/s corresponds to a flow rate of 515.69 kg/s (1867.1 m^3/h). At the coolant velocity of 5.6 m/s, ΔT_{ONB}s are calculated to be 24 °C and 14 °C for the core inlet pressure of 1.43 kg/cm^2 with the core inlet temperatures of 20 °C and 40°C, respectively, and 27 °C and 16 °C for the core inlet pressure of 1.53 kg/cm^2 with the core inlet temperatures of 20 °C and 40°C, respectively. These ΔT_{ONB}s are enouh for the steady state condition of MEX-15 from safty point of view.

The minimum DNBRs obtained with the coolant velocity of 5.6 m/s are 2.59 and 2.52 for the core inlet pressure of 1.43 kg/cm^2 with the core inlet temperatures of 20 °C and 40°C, respectively, and 2.63 and 2.60 for the core inlet pressure of 1.53 kg/cm^2 with the core inlet temperatures of 20 °C and 40°C, respectively.

A core inlet pressure of 1.43 kg/cm^2 abs. is proposed as a conceptual design core inlet pressure. With the core inlet pressure of 1.43 kg/cm^2 abs., the top of core will be located at 7 m below the surface of the water of the reactor pool and the reactor pool depth will be about 10 m, and the location of the Nuclear Center of Mexico, 3074 m over sea level is taken into account. Also, a core inlet pressure of 1.53 kg/cm^2 is

proposed. With this pressure, the top of core will be located at about 8 m below the surface of the water of the reactor pool and the reactor pool depth will be about 11 m. The core inlet pressure will be fixed based on the shielding design of MEX-15 in the near future.

A maximum core inlet coolant temperature of 35 °C is proposed as a conceptual design core inlet temperature considerring maximum core inlet temperatures of similer research reactors such as JRR-3M. With the core inlet coolant temperature of 35 °C and with the two proposed core inlet pressures, enough safety margins against DNB (2.58 and 2.62) and OFI (2.21 and 2.27), and enough temperature margin against ONB (17 °C and 19 °C) were obtained for nominal power of 15MW.

The thermal-hydraulic calculations for the proposed control fuel element were also carried out for the two core inlet pressures of 1.43 kg/cm^2 and 1.53 kg/cm^2 with the core inlet temperature of 35°C and a core flow rate of 515.69 Kg/s (1867.1 m^3/h) using axial power distribution factors of the control fuel element CE-2, a radial power peaking factor of 0.963 and an uncertainty factor of 1.180.

The major thermal hydraulic analysis results calculated by the COOLOD-N code for the standard and control fuel elements under the proposed core inlet pressures, the core inlet temperature of 35°C and the core flow rate of 515.69 kg/s are given in Table 12 for the average and hot channel cases at the nominal power and over power of 114 %. The distribution of fuel plates surface temperature, fuel meat temperature, ONB temperature, saturation temperature of the coolant and bulk coolant temperature along the fuel plate for the hot channel at the nominal power condition are shown in Figure 19 for the standard fuel element, and in Figure 20 for the control fuel element. Figure 21 and Figure 22 show the results for an over power of 114 % condition for the standard and control fuel element, respectively.

The major conceptual design parameters proposed for the core thermal hydraulic design of the Multiple Purpose Research Reactor MEX-15 are given in Table 13.

Table 12a Thermal hydraulic analysis results for the average and hot channel for standard and control fuel elements at the core inlet pressure of 1.43 kg/cm^2, core inlet temperature of 35 °C under nominal power and over power (114%) conditions

Parameters	Standard fuel element				Control fuel element	
	Nominal power		Over power		Nominal power	Over power
	Average	Hot	Average	Hot	Hot	Hot
Coolant velocity (m/s)	5.6	5.6	5.6	5.6	4.0	4.0
Total pressure drop across the core (kg/cm^2)	0.647	0.645	0.647	0.644	0.350	0.350
Pressure at fuel plate exit (kg/cm^2)	0.842	0.843	0.843	0.845	1.141	1.142
Saturation temperature at fuel plate exit (°C)	94.4	94.4	94.4	94.5	102.8	102.8
Peak clad temperature (°C)	88.3	98.0	95.2	105.5	64.8	68.8
Peak fuel meat temperature (°C)	102.7	112.3	111.6	121.9	70.6	75.4
Average heat flux (W/cm^2)	87.1	104.5	91.1	119.2	36.0	41.1
Peak heat flux (W/cm^2)	119.3	143.2	136.0	163.3	48.8	55.6
Temperature margin against ONB (ΔT_{ONB}) (°C)	25.4	17.0	19.4	10.4	48.3	44.9
Safety margin :						
- DNBR	3.10	2.58	2.72	2.27	5.68	4.99
- OFIR	2.65	2.21	2.33	1.94	4.77	4.07

Table 12b Thermal hydraulic analysis results for the average and hot channel for standard and control fuel elements at the core inlet pressure of 1.53 kg/cm^2, core inlet temperature of 35 °C under nominal power and over power (114%) conditions

Parameters	Standard fuel element				Control fuel element	
	Nominal power		Over power		Nominal power	Over power
	Average	Hot	Average	Hot	Hot	Hot
Coolant velocity (m/s)	5.6	5.6	5.6	5.6	4.0	4.0
Total pressure drop across the core (kg/cm^2)	0.647	0.645	0.646	0.644	0.350	0.350
Pressure at fuel plate exit (kg/cm^2)	0.944	0.945	0.945	0.946	1.243	1.244
Saturation temperature at fuel plate exit (°C)	97.5	97.5	97.5	97.6	105.3	105.3
Peak clad temperature (°C)	88.3	98.0	95.2	105.5	64.8	68.8
Peak fuel meat temperature (°C)	102.7	112.3	111.6	121.9	70.6	75.4
Average heat flux (W/cm^2)	87.1	104.5	91.1	119.2	36.0	41.1
Peak heat flux (W/cm^2)	119.3	143.2	136.0	163.3	48.8	55.6
Temperature margin against ONB (ΔT_{ONB}) (°C)	27.5	18.9	21.5	12.1	50.4	46.9
Safety margin :						
- DNBR	3.14	2.62	2.76	2.30	5.84	5.12
- OFIR	2.73	2.21	2.27	1.99	4.64	4.18

Table 13 Core thermal-hydraulic conceptual design data

Steady-state power level (MW)	15.0
Fuel meat dimensions:	
- thickness (mm)	63.0
- width (mm)	0.51
- length (mm)	600.0
Fuel clad thickness (mm)	0.38
Coolant channel thickness (mm)	2.563
Primary-loop flow rate* (kg/s),(m^3/h)	567.3 (2054)
Flow direction	Downward
Radial power peaking factor** (max.)	1.485
Axial power peaking factor** (max.)	1.320
Coolant inlet pressure (kg/cm^2)	1.43 (1.53)
Safety margin:	
- MDNBR	2.58 (2.62)
- MOFRI	2.21 (2.27)
Hydraulic diameter of subchannel (Equivalent)(mm)	4.94
Total heat transfer area in the core (m^2)	43.3

* It includes 10 % of the core bypass flow
** From neutronic calculations[7]

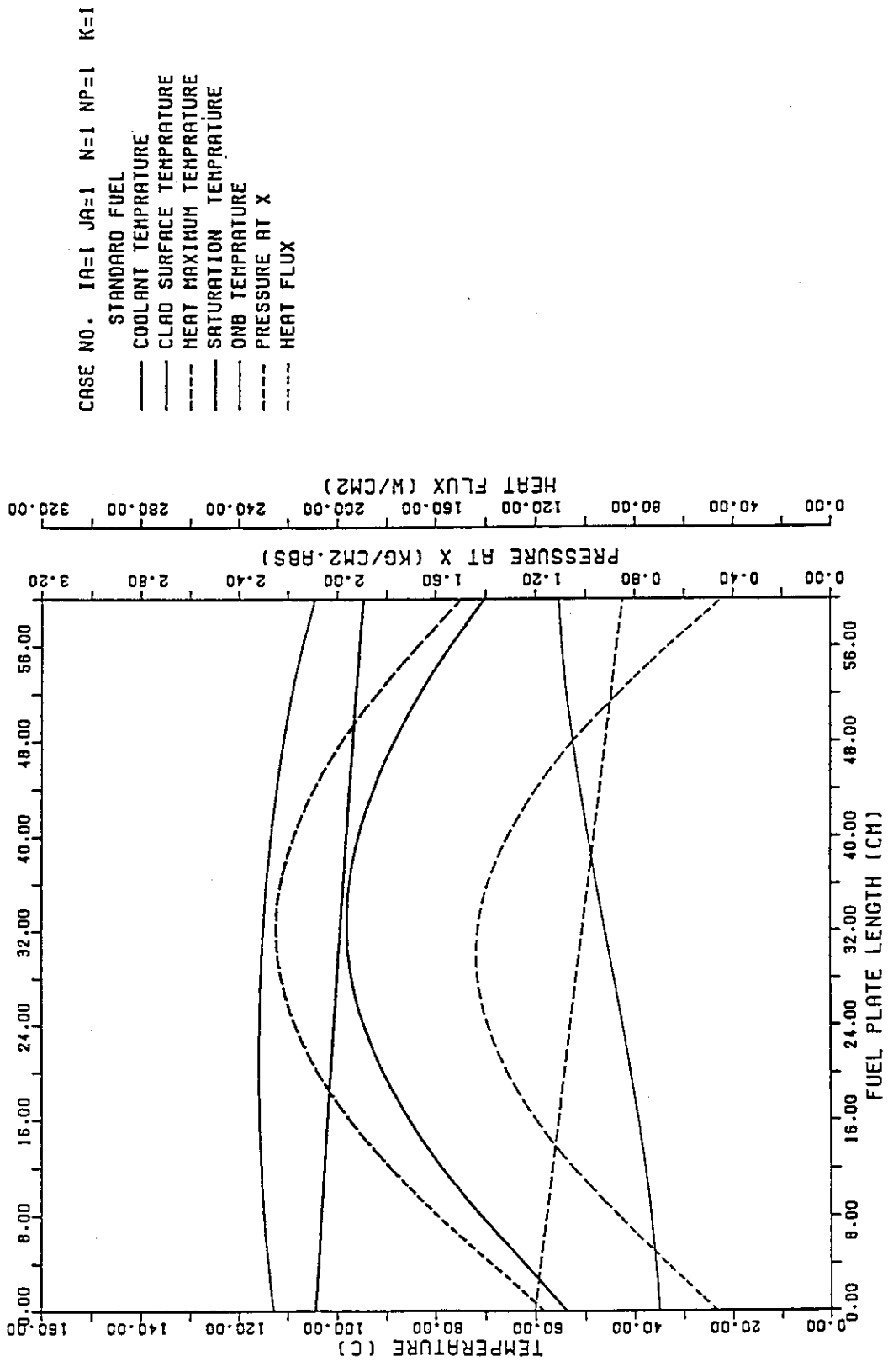


Figure 19 Calculation results of the temperature distributions at BOC core configuration under the nominal power condition for the hot channel.

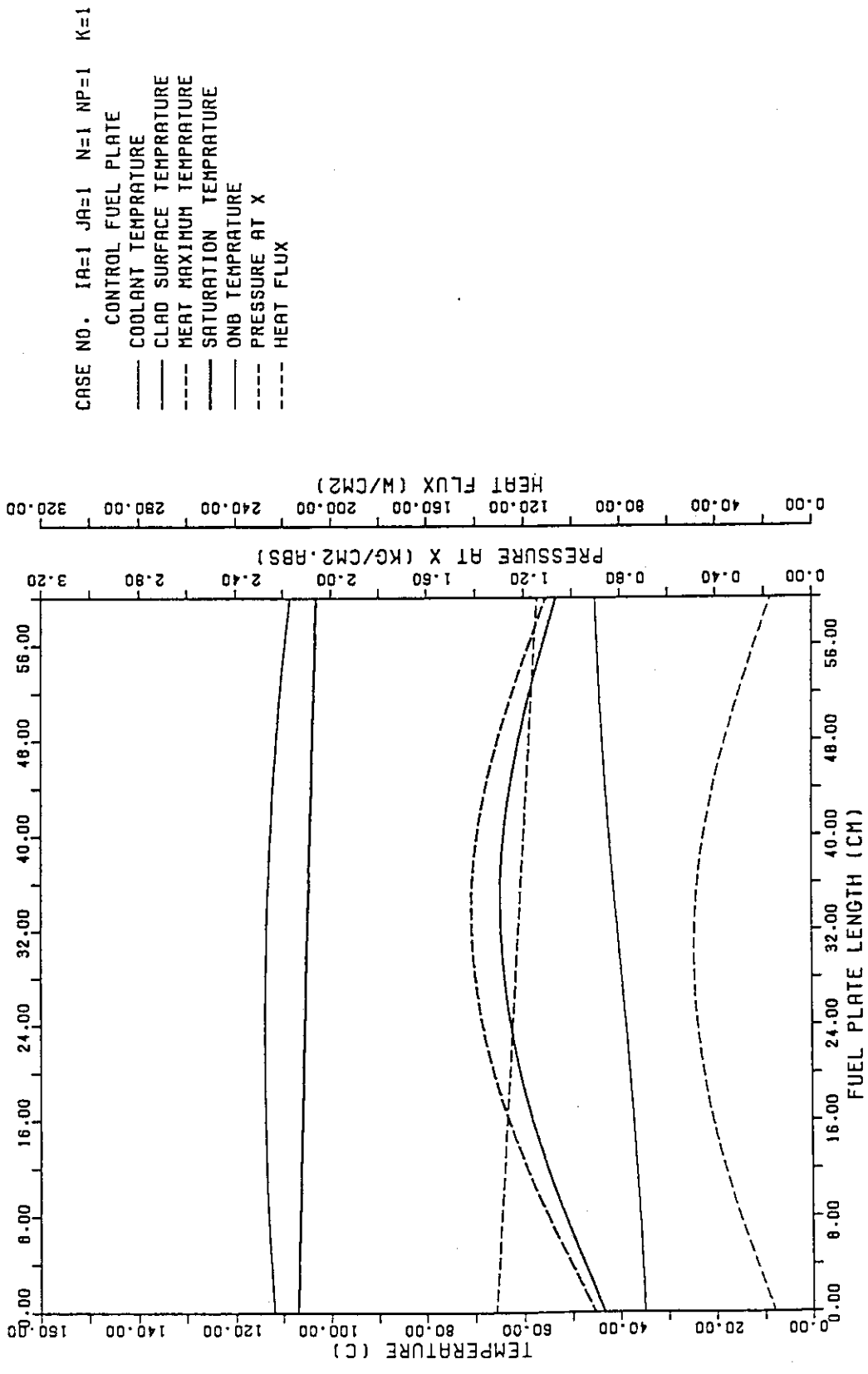


Figure 20 Calculation results of the temperature distributions at BOC core configuration under the nominal power condition for the control fuel element

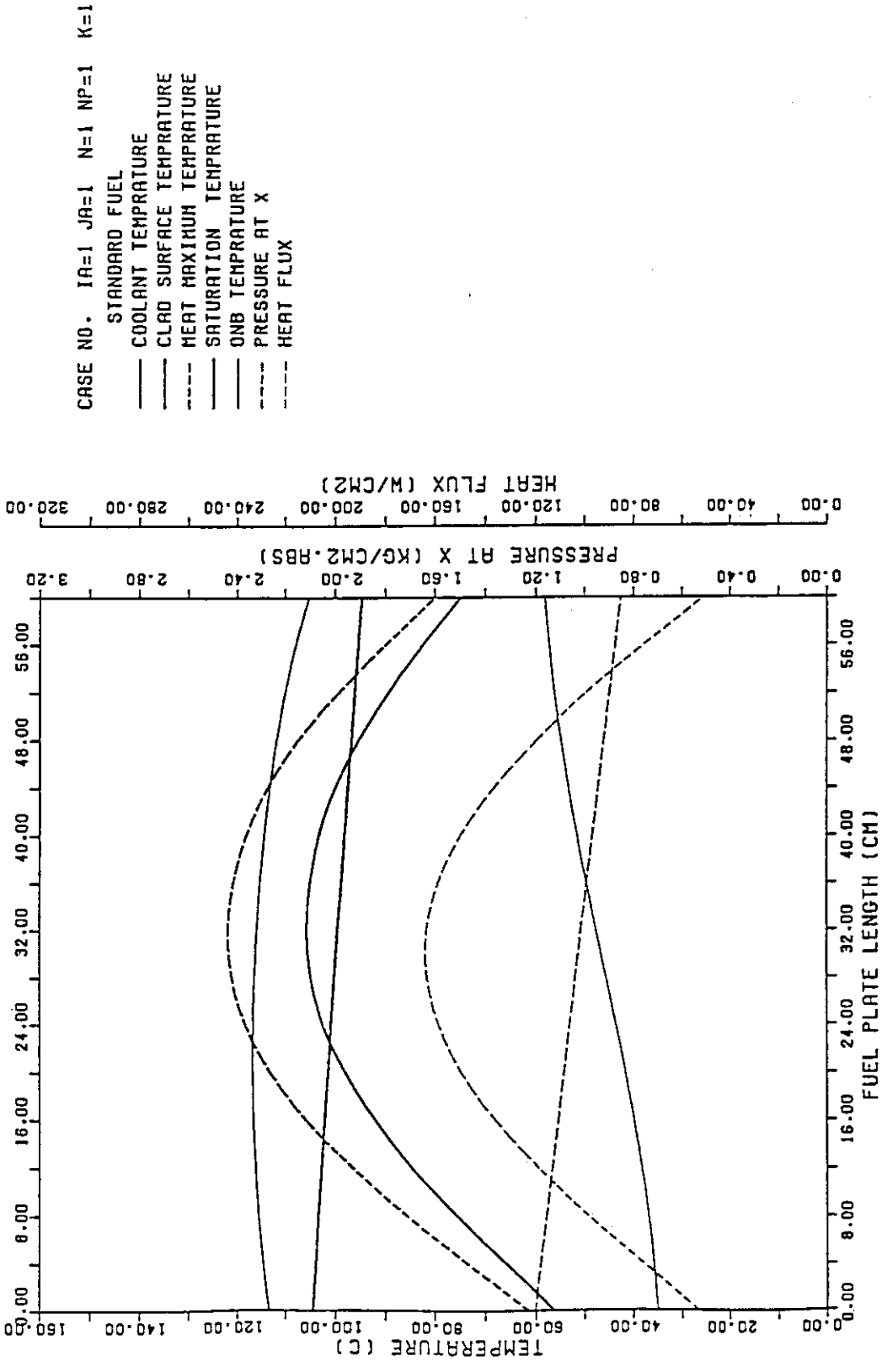


Figure 21 Calculation results of the temperature distributions at BOC core configuration under the over power condition for the hot channel

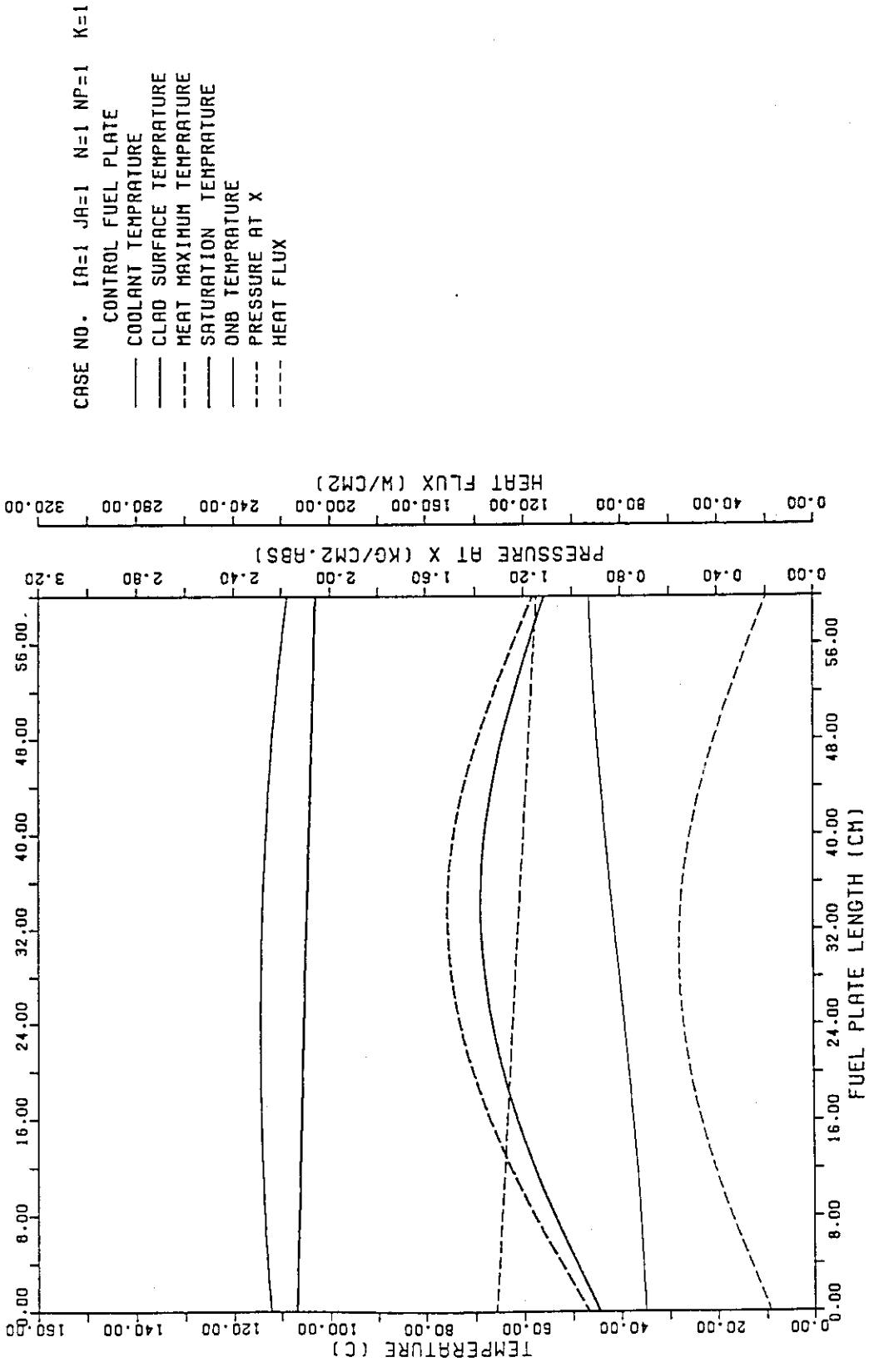


Figure 22 Calculation results of the temperature distributions at BOC core configuration under the over power condition for the control fuel element

6.2 Analysis results for natural convection cooling condition

MEX-15 will have a natural convection cooling mode as well as a forced convection cooling mode. The design criteria are the same as those for the forced convection cooling mode, neither to allow nucleate boiling anywhere in the core and no to have minimum DNBR less than 1.5.

It is proposed that two natural circulation flaps of 60 cm x 60 cm will be installed on the side wall of lower plenum for natural convection cooling as shown in Figure 4.

Thermal hydraulic calculations were carried out for BOC core under the natural convection cooling mode to investigate how much core power could be removed with the proposed natural circulation flaps. In the calculations, a radial peaking factor, a local peaking factor, axial power distribution factors, an uncertainty factor, and enginnering hot channel factors shown in Table 7 were used as well as forced convection cooling mode calculations. But, for flow rate calculations under natural convection, a radial peaking factor, a local peaking factor, an uncertainty factor, and enginnering hot channel factors were not considered, in other words, the flow rates under natural convection cooling mode were calculated as flow rates for an average channel to obtain conservative results.

In the thermal hydraulic calculations of BOC core under natural convection cooling, fuel plate surface temperature, ONB temperature, saturation temperature of the coolant, bulk coolant temperature and DNBR for the hot channel were analyzed as a function of the core power. A core inlet temperature of 35°C and a core inlet pressure of 1.51 kg/cm^2 abs. have been used in the calculations.

The main thermal hydraulic results calculated by COOLOD-N code are given in Table 14. In this table, it can be observed that the pressures at the fuel plate exit are the almost same. Because the coolant velocities under natural convection cooling are very low and the pressure loss across the core are also very small. Therefore, the local pressures in the core are almost equal to the head of water.

Figure 23 shows the results of total core power vs core flow rate. These results indicate that the core flow rate increases with an increase of the total core power. Because an increase of the total core power gives lower water density at the core exit and results in an increase of the driving force.

Figure 24 shows the calculations results of DNBR as a function of the total core power. The results show that an increase of total core power gives lower DNBR.

The temperature margin against ONB as a function of the total core power is shown in Figure 24. This figure shows that an increase of the total core power results in the decrease of ΔT_{ONB}.

It can be seen in Figure 25 the total core power up about 330 kW can be removed by the natural convection cooling mode under the condition that nucleate boiling is not permitted anywhere in the core. From the point of view of the safety margin of DNBR, there is an enough safety margin (about 5.9) even at the total core power of 330 kW.

From the above results, it can be concluded that the total power up to 300 kW, the reactor can be operated under natural convection cooling mode, with the two natural circulation flaps under the condition that the nucleate boiling is not allowed anywhere in the core. At the core power of 300 kW, the total core flow rate is 3.5 kg/s with an average core velocity of 3.8 cm/s and a MDNBR is 6.3 which is more higher than 1.5 for the safety margin against DNB.

Figure 26 and figure 27 show the typical temperature distributions under the natural convection cooling at the total core power of 300 kW and 1000 kW,respectively.

Table 14 Thermal-hydraulic analysis results of MEX-15 reactor under natural convection cooling condition

Parameters	Total core power (kW)						
	200	250	300	350	400	450	500
Core flow rate (kg/s)	2.721	3.120	3.469	3.797	4.108	4.339	4.688
Coolant velocity (cm/s)	2.93	3.42	3.81	4.17	4.52	4.85	5.17
Pressure at fuel plate exit (kg/cm^2)	1.433	1.433	1.433	1.433	1.433	1.434	1.434
Saturation temperature at fuel plate exit (°C)	109.4	109.4	109.4	109.4	109.4	109.4	109.4
Coolant temperature rise across the channel (°C)	50.9	55.5	59.9	63.8	67.4	70.7	73.8
Peak clad temperature (°C)	96.7	103.4	109.8	111.7	112.7	113.4	114.1
Peak heat flux (W/cm^2)	1.91	2.39	2.86	3.34	3.82	4.30	4.77
Temperature margin against ONB (ΔT_{ONB}) (°C)	14.2	7.8	1.6	-0.3	-1.3	-2.2	-2.8
Safety margin : DNBR	8.19	7.11	6.31	5.71	5.24	4.85	4.53

Parameters	Total core power (kW)						
	550	600	650	700	800	900	1000
Core flow rate (kg/s)	4.959	5.181	5.431	5.678	6.198	6.654	7.093
Coolant velocity (cm/s)	5.47	5.72	6.01	6.28	6.86	7.37	7.86
Pressure at fuel plate exit (kg/cm^2)	1.434	1.434	1.433	1.434	1.433	1.433	1.433
Saturation temperature at fuel plate exit (°C)	109.4	109.4	109.4	109.4	109.4	109.4	109.4
Coolant temperature rise across the channel (°C)	74.5	74.6	74.6	74.6	74.7	74.7	74.8
Peak clad temperature (°C)	114.7	115.2	115.6	116.0	116.7	117.4	118.0
Peak heat flux (W/cm^2)	5.25	5.73	6.21	6.68	7.64	8.59	9.55
Temperature margin against ONB (ΔT_{ONB}) (°C)	-3.5	-4.1	-4.5	-5.0	-5.6	-6.3	-6.8
Safety margin : DNBR	4.26	4.01	3.81	3.63	3.36	3.12	2.92

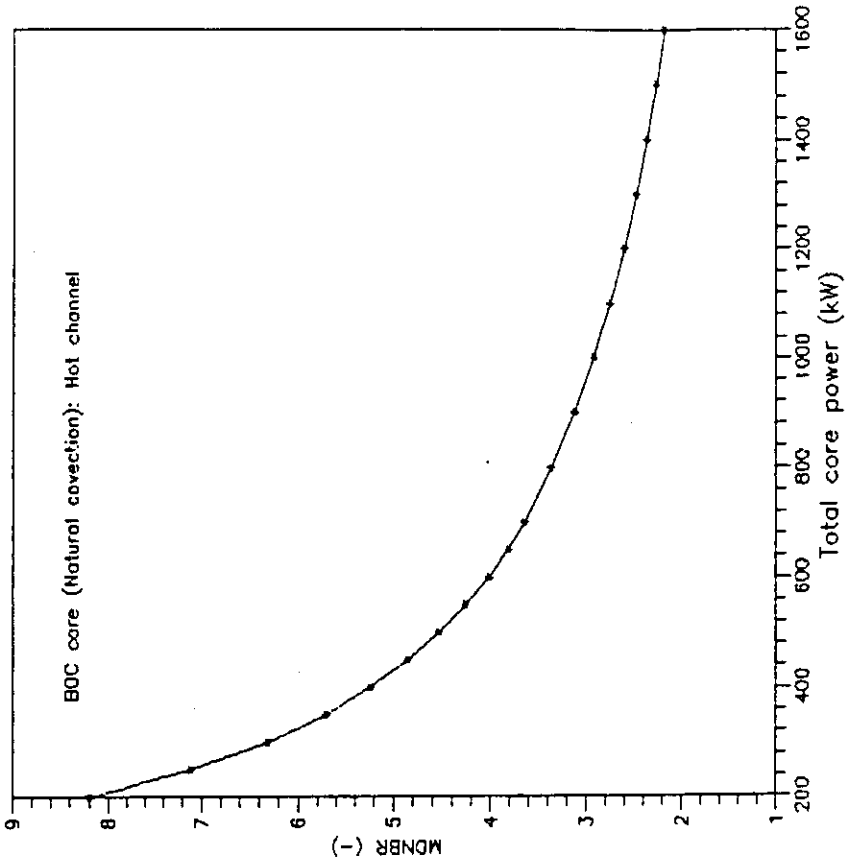


Figure 24 Calculation results of minimum DNBR vs. total core power under natural convection cooling mode.

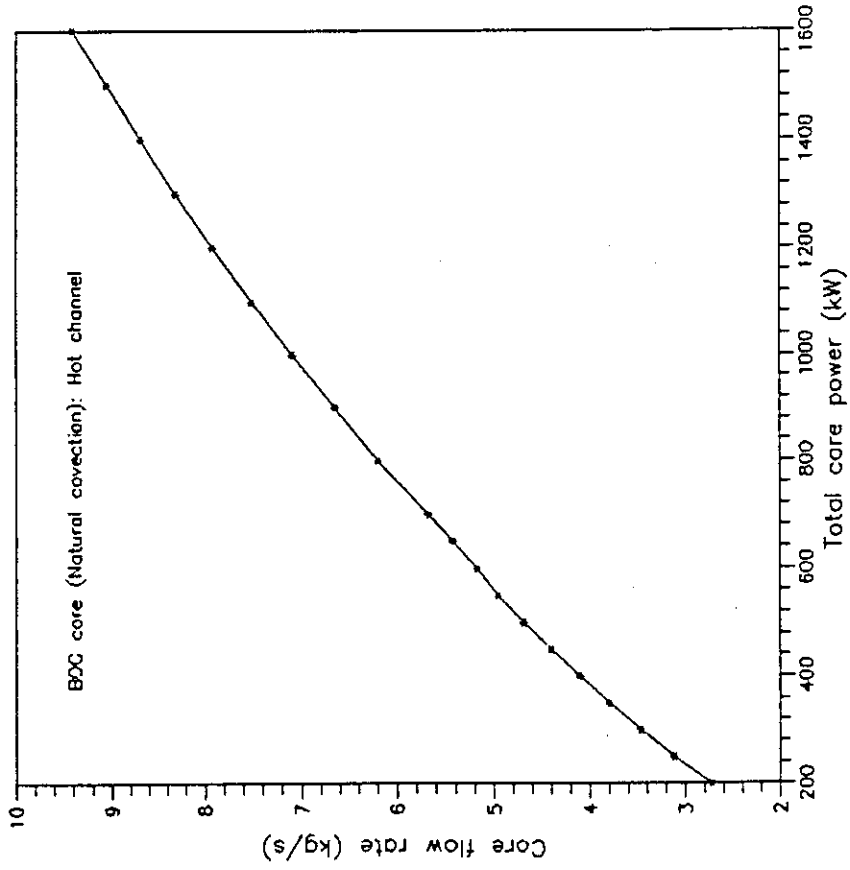


Figure 23 Calculation results of core flow vs. total core power under natural convection cooling mode.

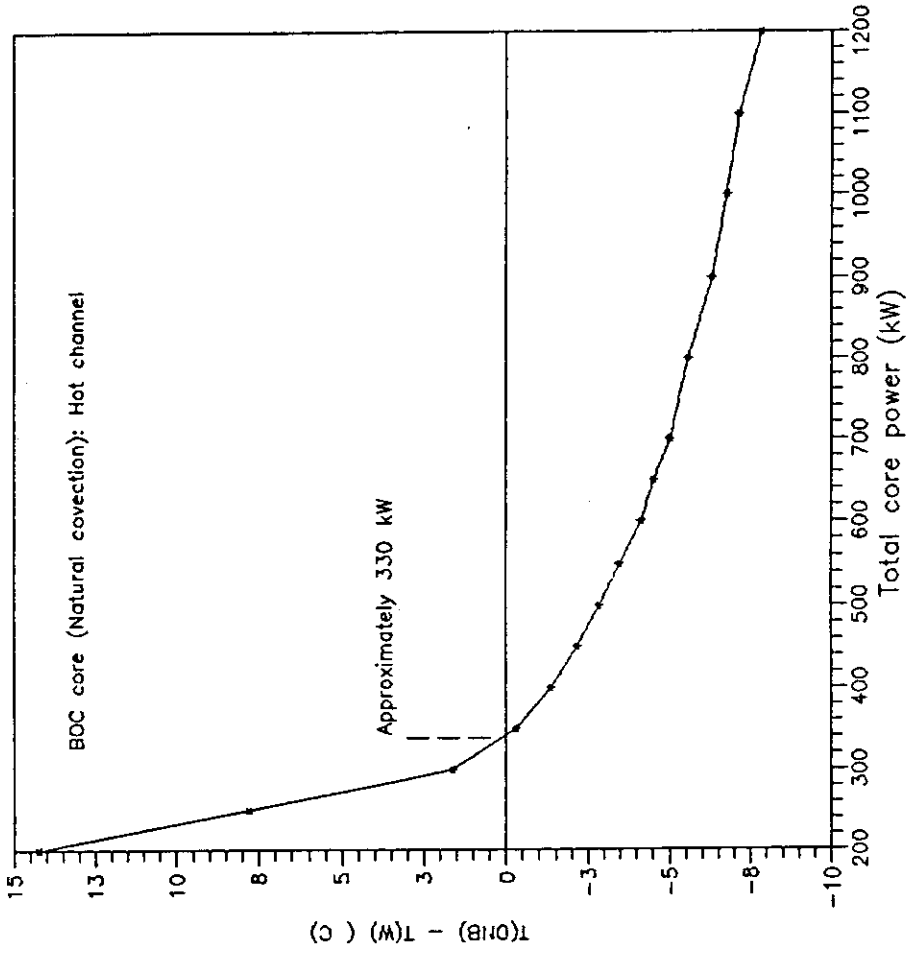


Figure 25 Calculation results of temperature margin against the ONB temperature vs. total core power under natural convection cooling mode.

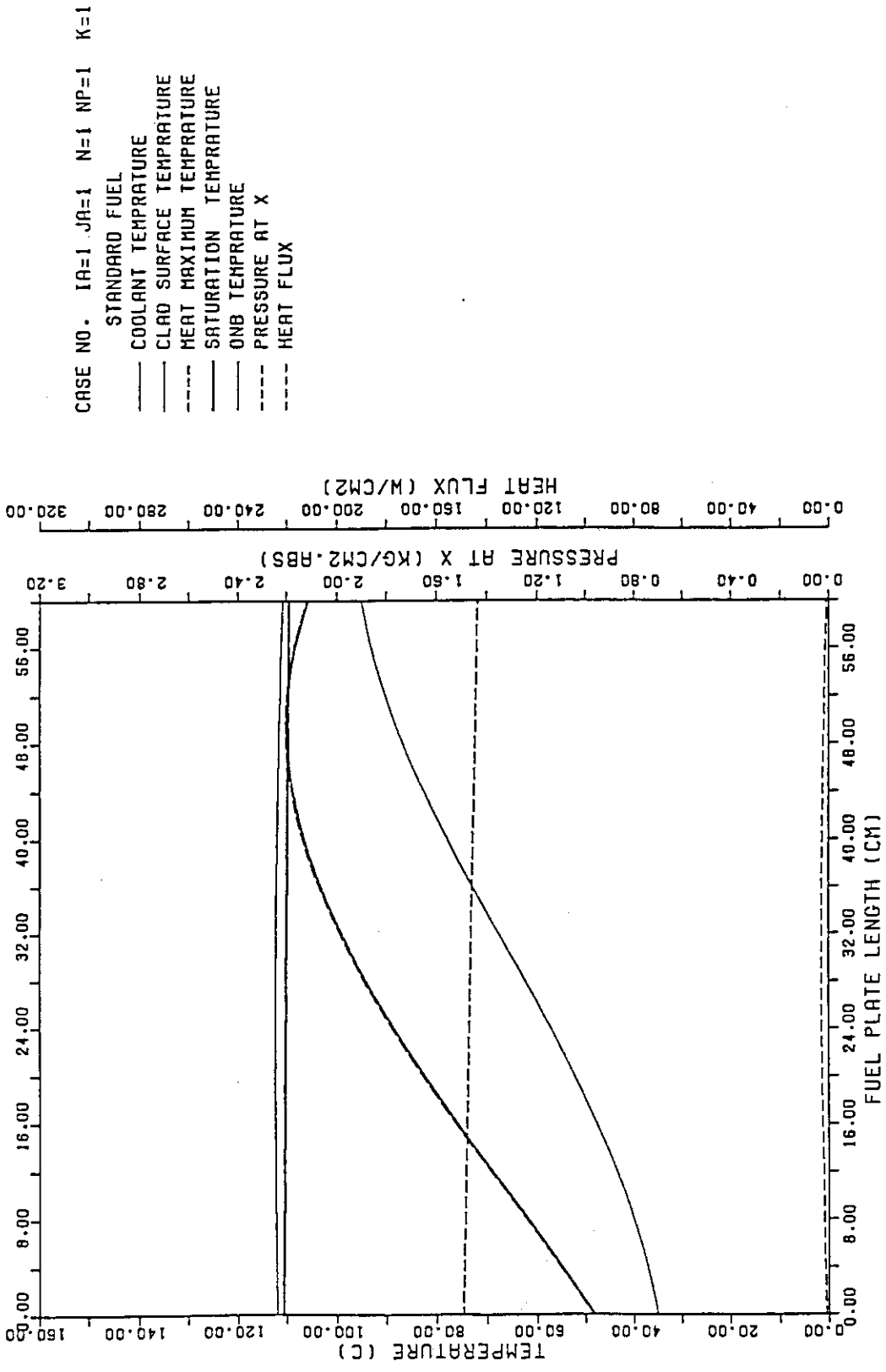


Figure 26 Calculation results of temperature distributions at BOC core configuration under the natural convection cooling mode. (Hot channel : Total core power = 300 kW)

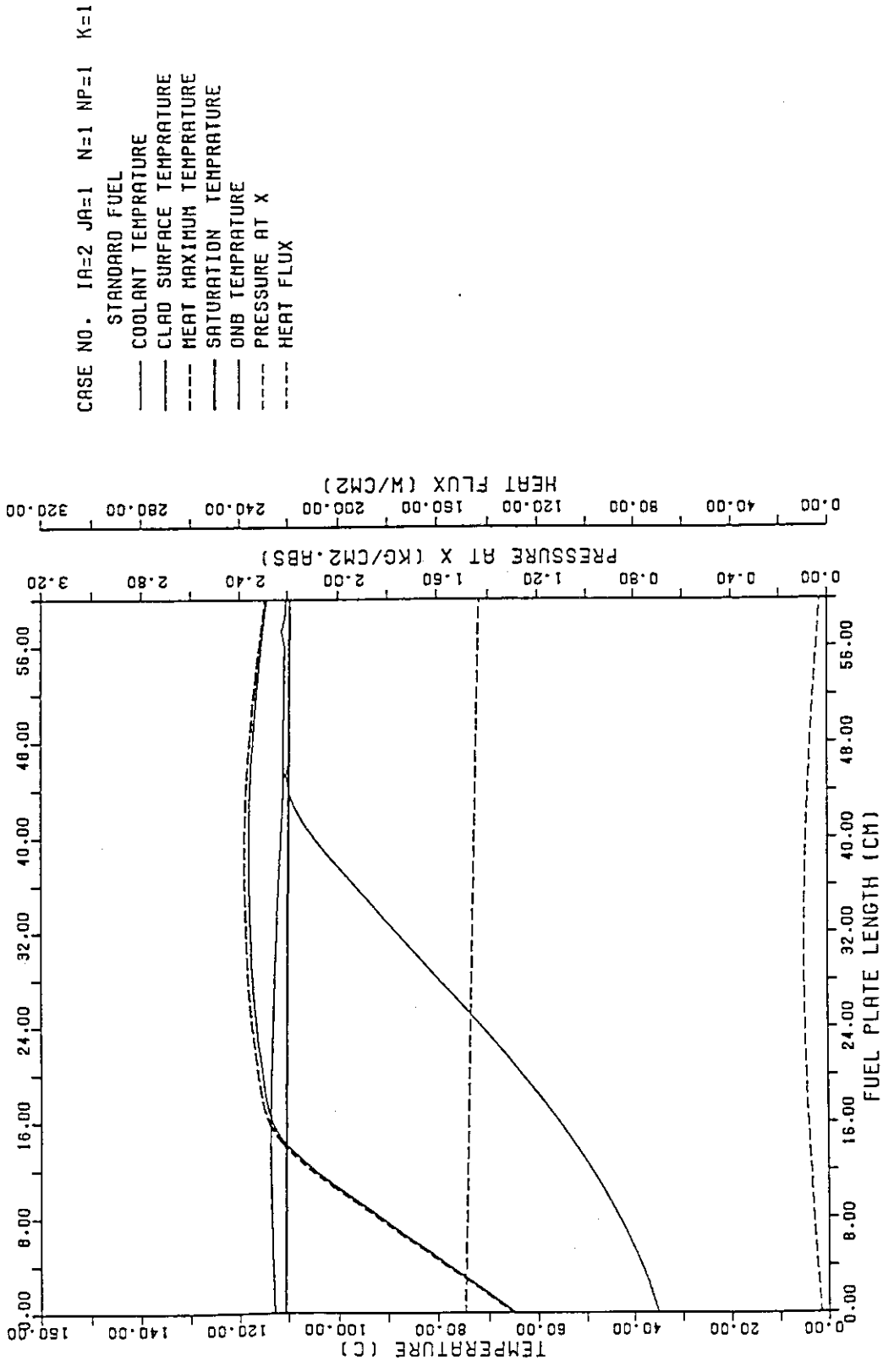


Figure 27 Calculation results of temperature distributions at BOC core configuration under the natural convection cooling mode. (Hot channel : Total core power = 1000 kW)

7. CONCLUSIONS

The core thermal-hydraulic conceptual design of the Multiple Purpose Research Reactor MEX-15 was performed for two cooling modes, forced convection cooling and natural convection cooling. The key criteria are first to avoid the nucleate boiling anywhere in the core and second to have enough safety margin to the DNB for normal operation conditions. The results of the thermal-hydraulic conceptual design and analysis show that the optimum coolant velocity in the standard fuel element is about 5.6 m/s with the minimum temperature margin against the ONB temperature of about 17 °C and the DNBR of 2.58 for the forced-convection cooling mode at a core power of 15 MW with the core inlet pressures of 1.43 kg/cm^2 and the core inlet coolant temperature of 35 °C.

It was also determined that the total core power up to about 300 kW can be removed by the natural convection cooling under the condition that nucleate boiling is not allowed anywhere in the core. The minimum temperature margin against ONB temperature and the minimum DNBR at 300 kW are 1.6 °C and 6.31, respectively.

The results obtained in this work establishes the preliminary technical specifications for the core thermal-hydraulic design of the Multiple Purpose Research Reactor MEX-15.

ACKNOWLEDGMENT

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APPENDIX A

The geometrical form and dimensions of the fuel element nozzle were chosen based on the pressure calculation results at nozzle and the pressure loss in it as a function of the coolant velocity.

The calculations were carried out for the forced convection cooling mode for four different pressures, 1.43, 1.53, 1.63 and 1.73 kg/cm^2 at the core inlet and three different core inlet temperatures, 20, 30 and 40 °C, respectively, under various flow rates through core ranging from 188.2 kg/s to 846.9 kg/s.

The geometrical form and dimensions proposed for the fuel element nozzle are given in the following table:

Geometrical form	Radio or Thickness (mm)	Width (mm)	Length (mm)
Cylindrical	47.8	-	149.5
Rectangular	50.0	47.83	149.5
Rectangular	60.0	57.49	149.5
Rectangular	65.0	62.70	149.5

The pressure calculation model used is shown in the Figure A1 and the flow area, equivalent hydraulic diameter, Cb factor and the resistance coefficient due to geometry change in the fuel element for each proposed geometrical form and dimensions for the nozzle of the fuel element are given in the following table:

Parameter	Cylindrical form	Rectangular form		
		1	2	3
Flow area (cm^2)	17.95	23.91	34.44	40.41
Equivalent hydraulic diameter (cm)	4.78	4.89	5.87	6.36
Factor Cb	64.0	96.0	96.0	96.0
Resistance coefficient due to geometry change	0.327	0.269	0.168	0.140

The pressure calculation results at each proposed nozzle for the pressure of 1.43 kg/cm^2 at the core inlet with three different core inlet temperatures as, are shown in the Figure A2 as a function of the coolant velocity.

Figure A3 shows the pressure loss at each proposed nozzle as a function the coolant velocity, for of the pressure of 1.43 kg/cm^2 at the core inlet with a core inlet temperature of 30 °C, respectively.

The pressure at nozzle will not be negative in a rang of the coolant velocities between 2 m/s and 9 m/s, and it was adopted as criteria for the selection of the dimensions of the flow area of the fuel element nozzle that the pressure loss at nozzle is as low as possible.

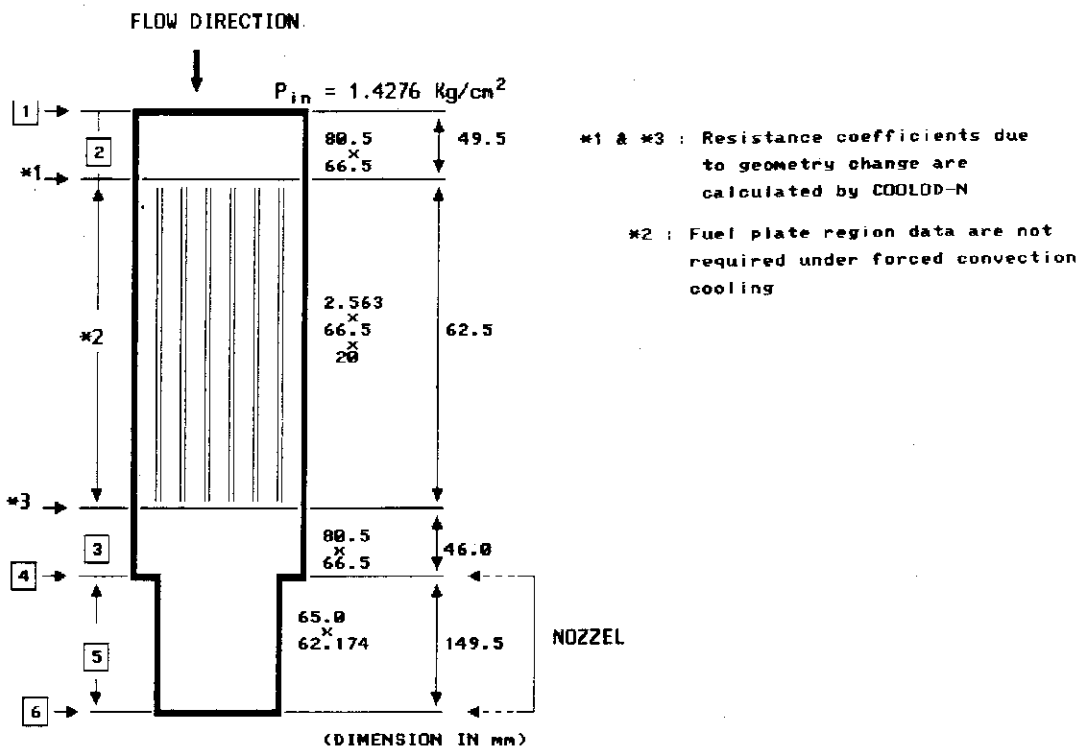

Figure A1 Pressure calculation model under forced convection cooling

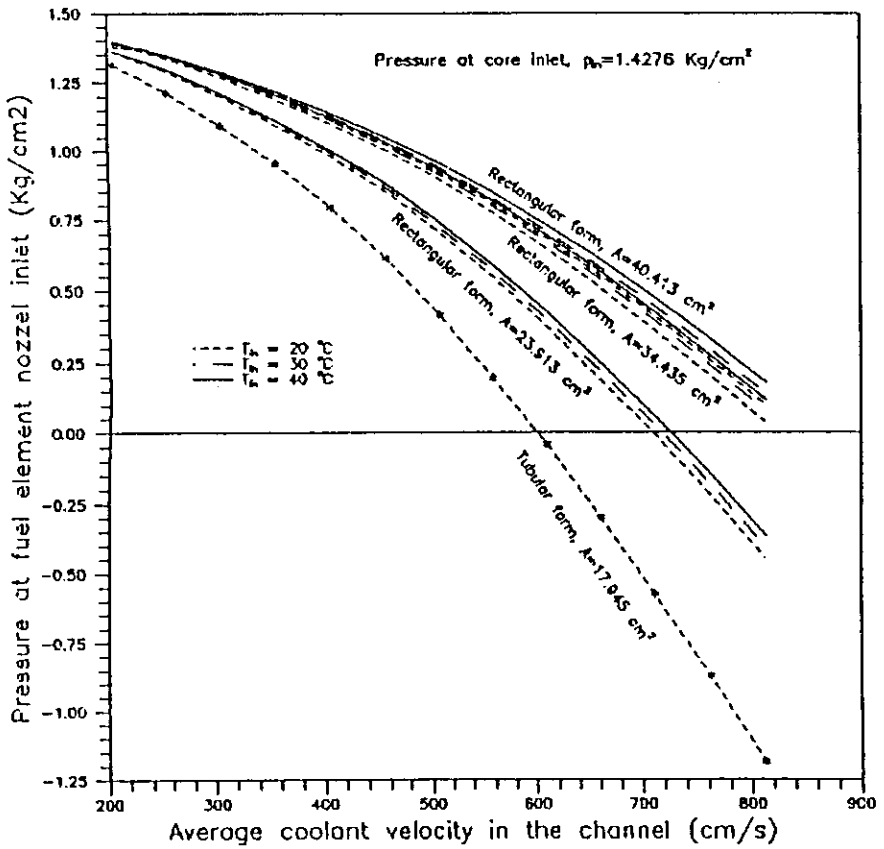

Figure A2 Calculation results of pressure for each nozzel as a function of coolant velocity for the standard fuel element at the core inlet pressure of 1.4276 kg/cm^2. (Hot channel)

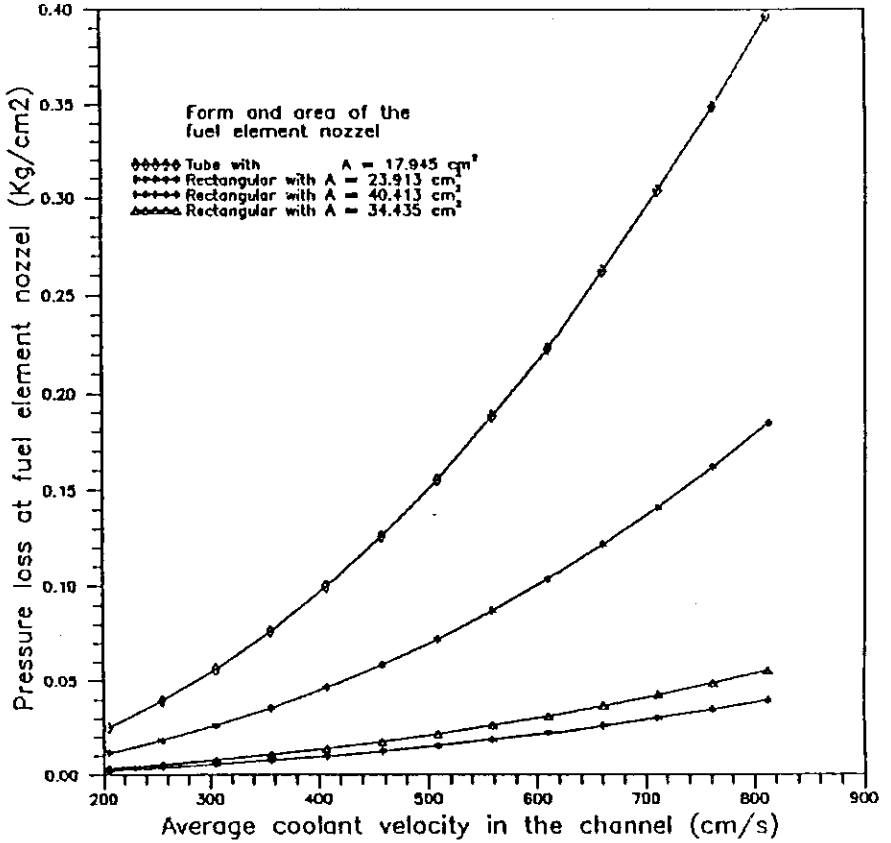

Figure A3 Calculation results of pressure loss at each nozzel as a function of coolant velocity for the standard fuel element at the core inlet pressure of 1.4276 kg/cm^2 with the core inlet temperature of 30°C. (Hot channel)