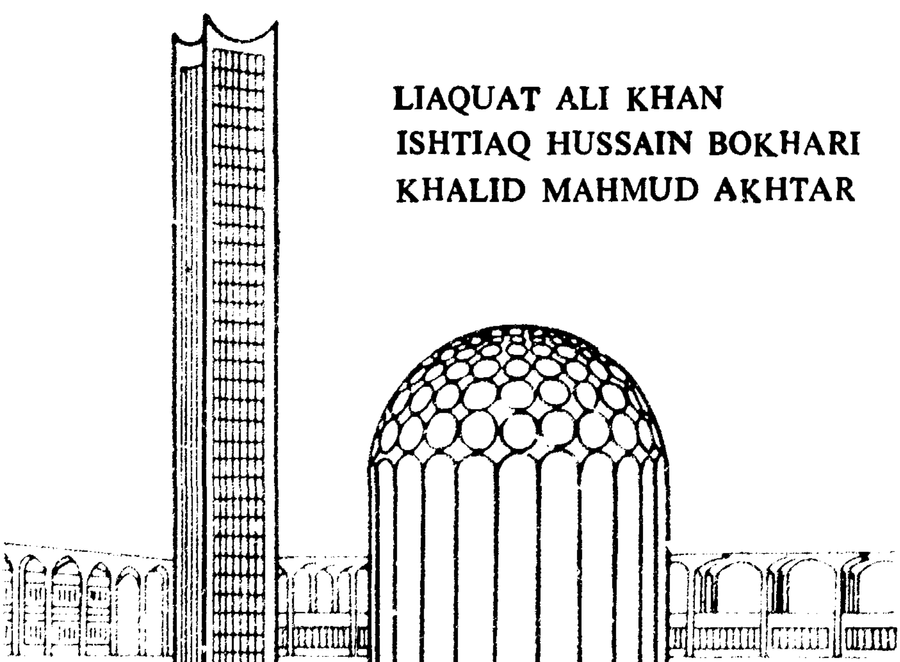
PINSTECH-121

NATURAL CONVECTION COOLING OF LEU CORES FOR PAKISTAN RESEARCH REACTOR-1



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ABSTRACT

The first high power and equilibrium LEU cores of PARR-1 have been analyzed to assess the maximum operating power based on natural convection cooling, need for forced cooling to remove the decay heat and to estimate safety margins that commensurate with the predetermined power limit. Computer code NATCON and standard correlations have been used for the analysis. The parameters studied include : coolant velocity, temperature distribution in the core, heat fluxes at onset of nucleate boiling, pulsed boiling and burnout.

CONTENTS

1. INTRODUCTION

The core of Pakistan Research Reactor-1 (PARR-1) is being converted from Highly Enriched Uranium (HEU) fuel to Low Enriched Uranium (LEU) fuel with power upgradation. The reactor is a swimming pool type research reactor using plate type fuel and cooled and moderated by demineralized light water. Heat generated by fission is removed by natural convection at low power levels (up to 100kW) and through forced circulation at higher power levels. During forced convection the coolant flows downward under gravity.

The conversion and upgradation process has lead to major changes in the fuel element design. It has, therefore, become essential to analyze natural convection cooling conditions to ensure the proper cooling of LEU core during low power operation of the reactor. The analysis is also needed to assess if natural convection cooling would be adequate to remove the fission product decay heat after reactor shutdown when the forced flow of coolant is stopped.

Computer code NATCON [1] and standard correlations have been used to compute coolant velocity, temperature distributions in the core, heat fluxes at onset of nucleate boiling, pulsed boiling and burnout and corresponding safety margins.

2. DESIGN PARAMETERS OF LEU CORES FOR PARR-1

A general description of design parameters of LEU fuel and core configurations for PARR-1 is provided in Table 1. Briefly, the LEU core for PARR-1 utilizes 19.75% enriched fuel in the form of U_3Si_2-Al. Each standard fuel element contains 290 g U^{235} in 23 plates and can be arranged in any configuration on a 6 X 9 grid plate. Each control fuel element contains about 164 g U^{235} distributed in 13 fuel plates and have a rectangular passage for the movement of oval shaped control rod.

Two core configurations have been selected for the analysis, one of these is the most compact core which will be assembled for first high power operation and the other equilibrium core.

2.1 First High Power Core

The first core to be assembled for high power operation is shown in Fig. 1. It is a 6x4 core and comprises of 17 standard and 5 control fuel elements with two vacant positions at one face. The core is reflected by water on all the sides except one side facing the thermal column.

2.2 Equilibrium Core

Configuration of the equilibrium core to be operated in the long run, is shown in Fig. 2. The core consists of 23 standard and 5 control fuel elements. There are two irradiation positions inside the core. The core is reflected by graphite on two opposite faces and by light water on the remaining sides.

2.3 Heat Removal System

Heat generated by nuclear reaction is removed by natural convection at low power levels (up to 100 kW) and through a forced circulation at higher power levels. The primary coolant water, during forced convection mode, flows by gravity from either the stall or open end of the pool (depending upon the core position in use) downward through the reactor core, grid plate and plenum into the hold-up tank. The pool outlet flow rate is regulated independently by means of butterfly valves. These valves are installed in a shielded valve pit and are operated manually. In the valve pit, after the control valve, the stall and open pool outlet lines are combined into one. An automatic flow control valve will be installed at this line to regulate flow and maintain pool level during reactor operation. For enhanced power level of 9 MW, coolant flow rate through the core will be adjusted to 900 m^3/h with the 'low flow trip' at 810 m^3/h (i.e. 90% of normal flow).

A hinged counterbalance safety flapper is attached to the side of plenum to allow transition from forced to natural convection. For high power operation, the flapper is closed initially and is counterbalanced with the flapper ballast weight and adjusting weights against the suction pressure of the coolant at high flow rates. The safety flapper drops open establishing natural convection if the coolant flow rate reduces to 200 m^3/h (i.e. 22% of normal flow). It has an interlock with the scram system for reactor power level above 100 kW (limit for natural convection).

3. COMPUTER CODE 'NATCON'

The code ' NATCON ' computes the steady state natural convection cooling conditions for a single rectangular channel cooled by light water. The driving force for natural circulation i.e. buoyancy is calculated based on the difference between the density of core water heated by core power and the density of pool water. The buoyancy force is balanced with frictional force. The steady state velocity is then obtained through an iterative procedure. Channel dimensions, hot channel factors, axial heat distribution, ambient pool temperature etc. are required as input. Light water properties are included in a subroutine. Output of the code includes heat flux, temperatures, pressures, flow rate etc.

NATCON may be run for a specific power or the program can increase the power, in steps, to a value at which the nucleate boiling is just reached.

4. STANDARD CORRELATIONS USED FOR THE ANALYSIS

Since the computer code NATCON does not support any choice of correlations for the determination of decay heat level after reactor shutdown, heat fluxes corresponding to pulsed boiling and burnout, these parameters have been calculated using the following correlations.

4.1 Fission Product Decay Heat

The decay heat generation rate, after reactor shutdown, has been calculated using standard fission product decay heat curve for uranium fueled reactors [2]. Time history of the decay power as a fraction of operating power of the reactor, after an infinite reactor operation time, is presented quantitatively in Table 2. For a finite operation time, this fraction can be obtained by the following relation:

$$P/P_o(t_o,t_s) = P/P_o(\infty,t_s) - P/P_o(\infty,t_o+t_s) \quad \text{.... (1)}$$

Where,

P = decay power after reactor shutdown (kW)

P_o = steady state operating power of the reactor (kW)

t_s = time after shutdown (s)

t_o = reactor operation time (s)

4.2 Heat Fluxes at Burnout and Pulsed Boiling

The problem discussed in this section is concerned with low heat flux and coolant velocity. If we start increasing the heat flux in a reactor having low coolant velocity (less than 1 m/s) in downward direction, one can observe the following phenomena [3]:

(i) Prior to boiling, flow will reverse its direction in the hot channel by reducing the coolant velocity to zero for a very short time. This phenomenon is called 'flow reversal', and is in principle similar to 'flow redistribution' except that it usually occurs in swimming pool type reactors without any previous boiling in the channel. During flow reversal, the heated slug of water which was supposed to escape from bottom of the channel will be reheated and there will be instantaneous rise in wall temperature;

(ii) Boiling will start in the hot channel and the oscillations in flow will be observed. These oscillations are generally accompanied by a periodic ejection of fluid from both ends of the channel (the channel being filled with vapors). This process is called 'pulsed boiling'. The wall temperatures during this boiling regime, remain moderate and oscillating;

(iii) Finally, when the heat flux exceeds a certain value, the pulsed boiling regime is accompanied by a permanent rise in the wall temperature which may lead to burnout.

The correlations related to these phenomena are given in the following subsections.

4.2.1 Heat Flux at Burnout

The heat flux corresponding to burnout can be calculated by [4,5]:

$$q_{BO} = D_h \; (0.023 \cdot \Delta T_{sub} + 4.56) \quad \pm 7\% \qquad \text{.... (2)}$$

Where,

q_{BO} = heat flux at burnout (W/cm^2)

D_h = hydraulic diameter of the channel (mm)

ΔT_{sub} = degree of subcooling at the channel inlet (^oC)

The above correlation is valid for the parameter ranges:

Pressure	: 1.17 to 1.73 bar
Hydraulic diameter (of aluminium tube)	: 6 to 10 mm
Heating length	: 916 mm
ΔT_{sub}	: 0 to 100 ^oC

The burnout heat flux is directly proportional to hydraulic diameter and inversely proportional to heating length.

4.2.2 Heat Flux at Pulsed Boiling

For small coolant velocities (less than 50 cm/s), the heat flux at pulsed boiling has been estimated [4,5] to be 1/4 to 1/2 of the heat flux at burnout.

5. METHODOLOGY

The reactor core has many flow channels. The individual channels do not generate the same heat and therefore under natural convection cooling they have different driving pressures and flow rates. For this reason two representative channels of the core have been selected. One assumes to have the hottest plate and associated flow channel and other being an average plate and associated flow channel. The results have then been generalized for the multichannel core.

The active core height has been divided in 20 axial regions and the axial power distribution has been represented by 21 axial mesh points having peak to average power ratio of 1.936 for the first high power core and 1.406 for the equilibrium core [6]. The axial power distribution, in both the cores, is shown in Figs. 3 and 4. Radial peaking factors of 1.622 and 2.098 [6], respectively, for first high power and equilibrium cores, multiplied the power distribution in the hot channel. Engineering hot channel factors have been used to further multiply the power in hot channel. These factors [7] include : (i) a factor of 1.2 for the coolant temperature rise due to manufacturing tolerances in the coolant channel spacing, (ii) a factor of 1.2 for the film temperature rise due to uncertainties in the heat transfer coefficient and inhomogenities in U^{235} distribution etc., and (iii) a factor of 1.1 for uncertainties in the calculated power distribution.

Thermal conductivities of clad and fuel have been calculated by using the following polynomials [8]:

$$k(LT\text{-}24\ Al) = 218.586 + 0.075215\ T - 1.607746\times10^{-4}\ T^2 \qquad \text{.... (3)}$$

$$k(U_3Si_2\text{-}Al) = 69.7032 + 0.0454835\ T - 8.76506\times10^{-5}\ T^2 \qquad \text{.... (4)}$$

Where,

k = thermal conductivity (W/m-^{o}K)

T = temperature (^{o}C)

These correlations are valid within temperature range of 0^oC to 550^oC.

All the calculations have been done with coolant inlet temperature of 38^oC and inlet pressure of 1.78 bar. The inlet pressure corresponds to static height of water from the channel bottom to a point 15 cm below the normal level of the pool (low level alarm set point).

6. RESULTS AND DISCUSSION

Results of the natural convection analysis, for LEU cores for PARR-1, are presented in Tables 3 to 5 and Figs. 5 to 10. The core parameters studied include: coolant velocity, temperature distribution in the cores, heat fluxes at onset of nucleate boiling, pulsed boiling and burnout etc.

Results of the analysis show that nucleate boiling will start when the reactor power approaches 451 kW and 622 kW, respectively, in first high power and equilibrium cores. These power levels correspond to peak heat flux of about 6.5 W/cm^2 in both the cores. It is important to note that these power levels are sensitive to coolant inlet temperature and height of water column above core top. For example, if we consider the case of first high power core, increasing coolant inlet temperature from 38°C to 60°C will reduce the power level corresponding to ONB from 451 kW to 335 kW. Similarly decrease in height of water column from 7.8 m to 5 m will reduce this power level from 451 kW to 416 kW.

The burnout heat flux at low velocities and similar conditions as in PARR-I has been calculated to be 24 W/cm^2. However, the pulsed boiling heat flux, at which the expulsion and re-entrance of water from both ends of the channel starts, is of the order of 6 W/cm^2 to 12 W/cm^2.

Another important factor limiting the reactor power in natural convection mode is the radiation level at the pool surface. According to reference 9, for power levels exceeding 200 kW the radiation fields are significantly high and may put an upper limit on operating power of the reactor.

From the above discussion it becomes clear that operating power level of the reactor, with natural convection mode, will have to be limited to some value below 200 kW. It is therefore recommended that the present limit of 100 kW may also be adopted for the converted core. Peak heat fluxes corresponding to this power level in first high power and equilibrium cores, will be 1.45 W/cm^2 and 1.04 W/cm^2, respectively.

While operating at 100 kW, the peak clad temperatures in both the cores will remain 50 °C below saturation temperature of water. The first high power core will have safety margins of 4.5, 4.1 and 16.5, respectively against onset of nucleate boiling, pulsed boiling and burnout. These margins for the equilibrium core will be 6.3, 5.7 and 23.0, respectively.

Now we take the case of post shutdown heat removal. The coolant flow during forced convection mode is in downward direction. After reactor shutdown the forced cooling is stopped and decay heat is removed by natural convection in upward direction. In a normal shutdown, forced circulation is continued after reactor shutdown, till the coolant outlet temperature drops down to 32 °C. It normally takes from 20 minutes to more than a hour depending upon the weather conditions. For a continuous operation of 5 days at 9 MW, decay power level of both the cores, at 10 minutes after reactor shutdown, has been calculated to be 160 kW.

This power level will drop down to 134 kW at 20 minutes. Both the values are much lower than those predicted for ONB. This shows that the natural convection cooling will be adequate to remove the decay heat in this case.

Considering a transient case by assuming that due to some false signal the automatic flow control valve at the coolant outlet line starts closing at its maximum speed while the reactor is operating at 9MW. The reactor scram will occur when the flow reduces to 90% of normal value after 1.65 s. Flow reversal will take place due to opening of safety flapper when flow reduces to 22% of its normal value at about 8.7 s. In transition from forced to natural convection a period of very low flow and flow inversion will occur during which the heated slug of water which was supposed to escape from the bottom of the channel will be reheated and there will be a sharp rise in temperatures. Computer code PARET [10] has been used to study such a situation. Results indicate that clad surface will reach a maximum temperature of 130 ^{o}C causing nucleate boiling in the hot channel about 10 s after the transient i.e. 8.35 s after reactor shutdown. At this instant the core power will be about 430 kW which is slightly lower than the power at ONB. This shows that nucleate boiling will persist until the slug of water being reheated escapes from top of the channel. After this the clad temperature will drop down and natural convection will remove the decay heat adequately.

7. CONCLUSIONS

Based on the analysis, following conclusions could be drawn :

(i) Safe operating power level for the natural convection cooling mode is 100 kW;

(ii) Nucleate boiling will commence in the first high power and equilibrium cores if the power level approaches 451 kW and 622 kW, respectively;

(iii) The first high power and equilibrium cores will enter into burnout crisis if the power level exceeds 1.6 MW and 2.3 MW, respectively;

(iv) Natural convection cooling would be adequate to remove the fission product decay heat after reactor shutdown when the forced cooling is stopped.

8. REFERENCES

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TABLE 1

Design Parameters of LEU Cores for PARR-1

Reactor Type	Swimming Pool
Reactor Power (MW)	9
Fuel	U_3Si_2-Al
U^{235} Enrichment (%)	19.75
Grid	6x9
Lattice Pitch (mm)	81x77.11
No. of Fuel Elements in :	
- First High Power Core	
. Standard Fuel Elements	17
. Control Fuel Elements	5
- Equilibrium Core	
. Standard Fuel Elements	23
. Control Fuel Elements	5
Fuel Element Dimensions (mm)	79.63x75.92
No. of Fuel Plates in :	
- Standard Fuel Element	23
- Control Fuel Element	13
No. of Dummy Plates in :	
- Standard Fuel Element	Nil
- Control Fuel Element	4
Shape of Fuel Plates	Straight
Thickness of Fuel Plate (mm) :	
- Inner Plates	1.27
- Outer Plates	1.5
Width of Fuel Plate	6.692
Length of Fuel Plate (mm)	625.0
Fuel Meat Dimensions (mm)	62.75x0.51x600
Thickness of Clad (mm) :	
- Inner Plates	0.38
- Outer Plates	0.495
Thickness of Side Plates (mm)	4.5
Shape of Control Rods	Oval
Absorber Material	Ag(80.5%) In(14.6%) Cd(4.9%)
Control Absorber Dimensions (mm)	64.96x3.1x600
Loading of U^{235} (g) in :	
- Standard Fuel Element	290
- Control Fuel Element	164
Water Channel Thickness (mm)	2.10
Coolant Inlet Temperature (^oC)	38
Coolant Pressure at (bar) :	
- Channel Top	1.712
- Channel Bottom	1.780
Max. Coolant Flow Rate (m^3/hr)	900

TABLE 2

Tabular Data for Standard Decay Heat Curve With Infinite Reactor Operation Time

Time After Shutdown t_s Seconds	Relative Power P/P_o	Time After Shutdown t_s Seconds	Relative Power P/P_o
1×10^{-1}	0.0675	6×10^{4}	0.00566
1×10^{0}	0.0625	8	0.00505
2	0.0590	1×10^{5}	0.00475
4	0.0552	2	0.00400
6	0.0533	4	0.00339
8	0.0512	6	0.00310
1×10^{1}	0.0500	8	0.00282
2	0.0450	1×10^{6}	0.00267
4	0.0396	2	0.00215
6	0.0365	4	0.00166
8	0.0346	6	0.00143
1×10^{2}	0.0331	8	0.00130
2	0.0275	1×10^{7}	0.00117
4	0.0235	2	0.00089
6	0.0211	4	0.00068
8	0.0196	6	0.00062
1×10^{3}	0.0185	8	0.00057
2	0.0157	1×10^{8}	0.000550
4	0.0128	2	0.000485
6	0.0112	4	0.000415
8	0.0105	6	0.000360
1×10^{4}	0.00965	8	0.000303
2	0.00795	1×10^{9}	0.000267
4	0.00625		

TABLE 3

Effects of Reactor Power on Peak Temperatures and Coolant Velocities in First High Power Core Cooled by Natural Convection

Reactor Power (kW)	Peak Temperatures (^{o}C)		Average Coolant Velocity (cm)	
	Clad Surface	Coolant Exit	Hot Channel	Average Channel
10	45.8	45.7	1.26	0.77
100	65.5	59.9	4.42	2.61
200	82.2	67.4	6.59	3.85
300	97.5	72.9	8.33	4.83
400	111.8	77.3	9.89	5.70
451*	119.1	79.3	10.61	6.11

* power level corresponding to ONB

TABLE 4

Effects of Reactor Power on Peak Temperatures and Coolant Velocities in Equilibrium Core Cooled by Natural Convection

Reactor Power (kW)	Peak Temperatures (^{o}C)		Average Coolant Velocity (cm)	
	Clad Surface	Coolant Exit	Hot Channel	Average Channel
10	46.6	46.3	1.15	0.62
100	64.6	61.7	4.03	2.07
200	76.3	69.9	5.99	3.04
300	86.6	75.9	7.58	3.80
400	96.5	80.7	8.98	4.48
500	106.7	84.9	10.23	5.09
600	116.8	88.7	11.37	5.65
622*	119.1	89.4	11.62	5.76

* power level corresponding to ONB

TABLE 5

Steady State Thermal Hydraulic Analysis of PARR-1
With Natural Convection Cooling at 100 kW

	First High Power Core	Equilibrium Core
Operating Power (kW)	100	100
Power Peaking Factors:		
- Axial	1.936	1.406
- Radial	1.622	2.098
- Engineering	1.584	1.584
- Total	4.974	4.672
Average Heat Flux (W/cm^2)	0.29	0.22
Peak Heat Flux (W/cm^2)	1.45	1.04
Total Flow Rate (m^3/h)	6.00	6.19
Average Coolant Velocity (cm/s):		
- Average Channel	2.61	2.07
- Hot Channel	4.42	4.03
Saturation Temperature at Channel Exit (°C)	115.5	115.5
Steady State Temperatures (°C):		
- Coolant Temperature Rise Across:		
. Average Channel	14.4	14.0
. Hot Channel	21.9	23.7
- Peak Clad Temperature		
. Average Channel	53.4	52.7
. Hot Channel	65.5	64.6
Onset of Nucleate Boiling:		
- Average Heat Flux (W/cm^2)	1.31	1.39
- Peak Temperatures (°C):		
. Clad Surface	119.1	119.1
. Coolant Exit	79.3	89.4
Pulsed Boiling Heat Flux (W/cm^2)	6-12	6-12
Burnout Heat Flux (W/cm^2)	24.0	24.0
Safety Margins:		
- Margin to ONB	4.5	6.3
- Margin to Pulsed Boiling	4.1-8.2	5.7-11.5
- Margin to Burnout	16.5	23.0

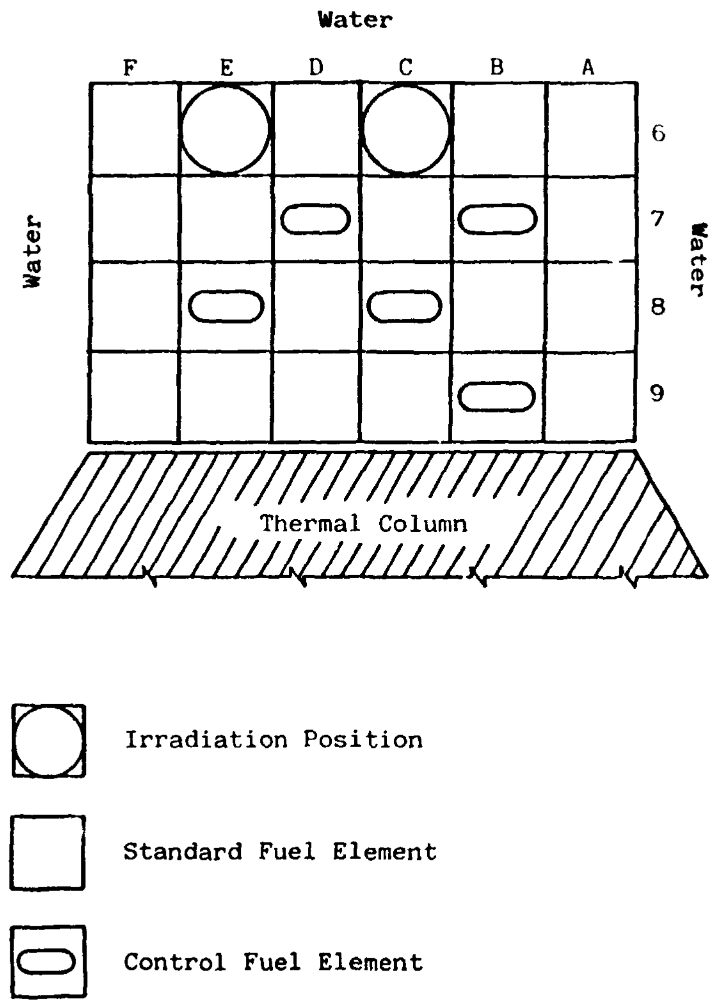

Fig. 1 Configuration of the First High Power Core for PARR-1

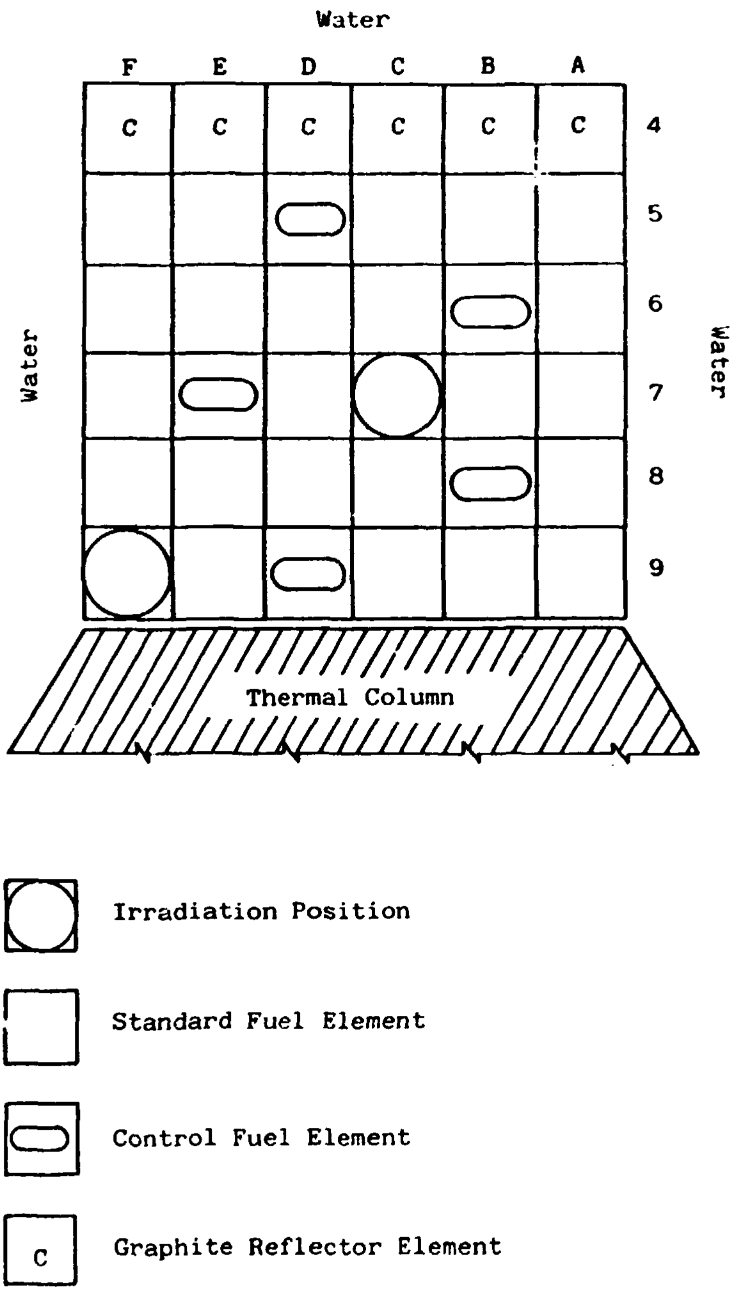

Fig. 2 Configuration of the Equilibrium LEU Core for PARR-1

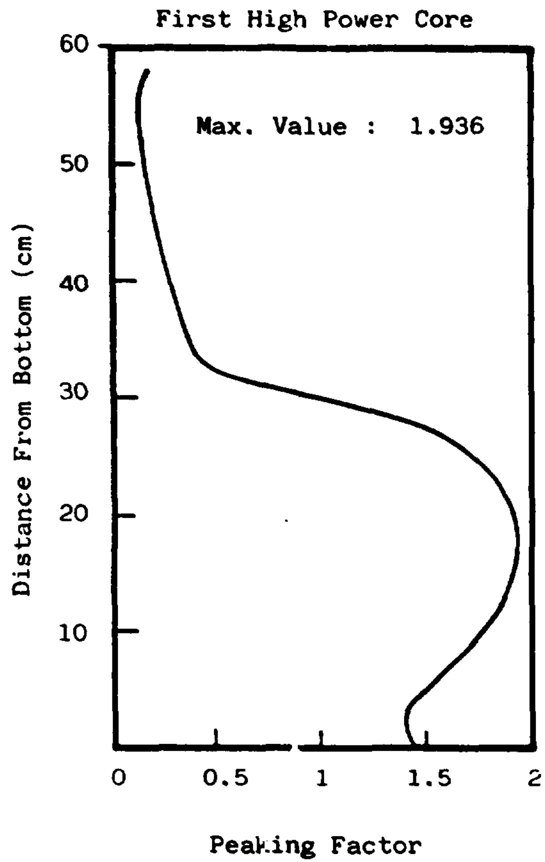

Fig. 3 Axial Power Distribution in the Hot Channel.

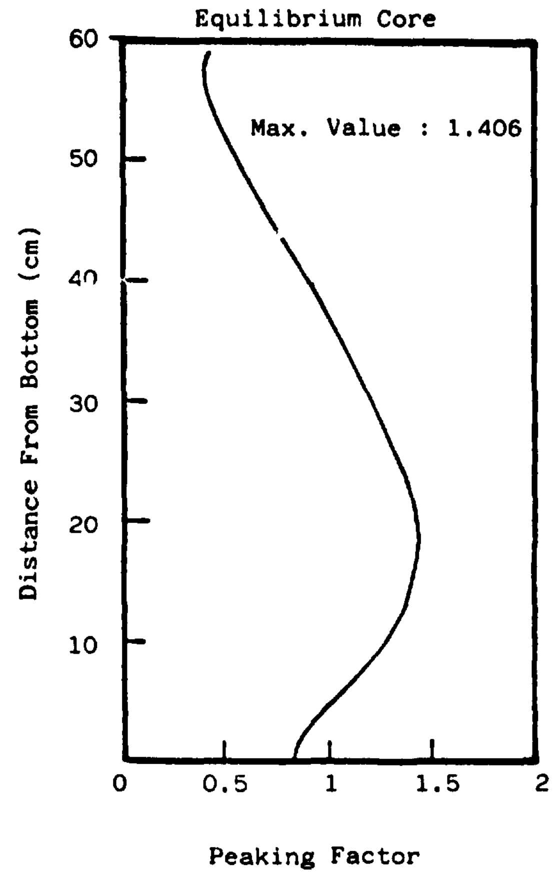

Fig. 4 Axial Power Distribution in the Hot Channel.

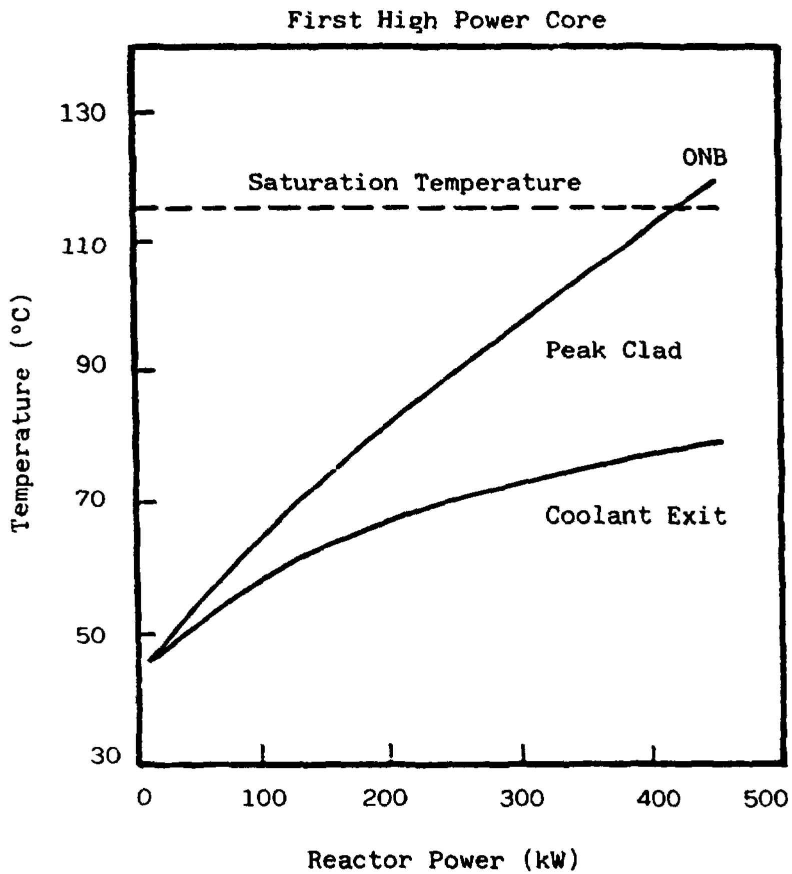

Fig. 5 Peak Temperatures as a Function of Reactor Power.

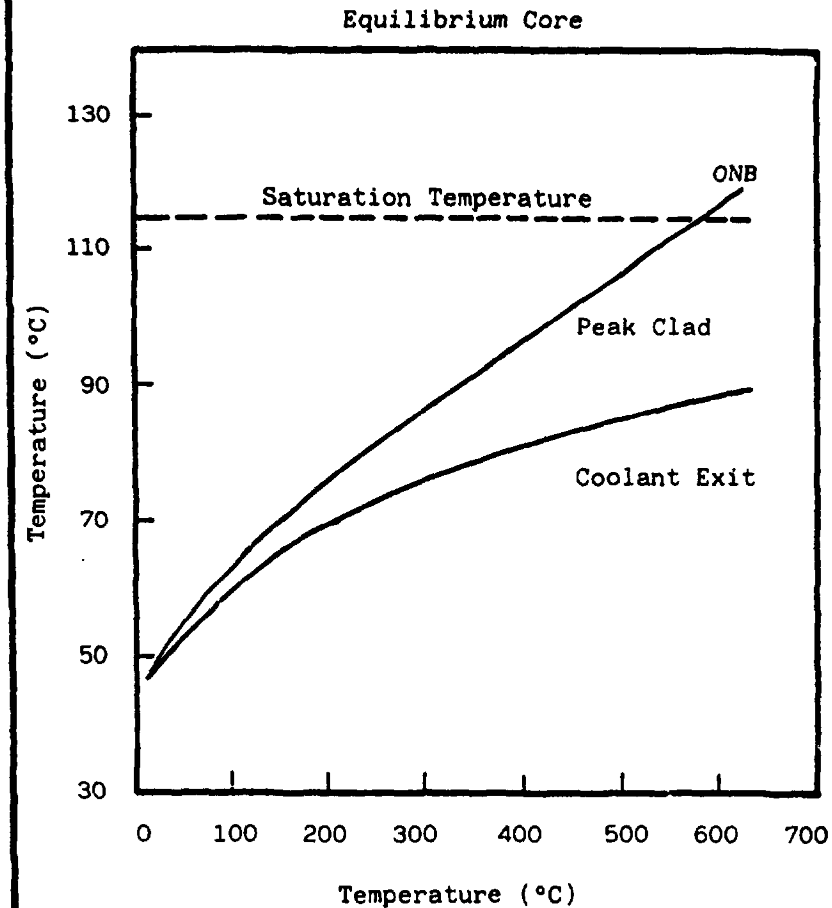

Fig. 6 Peak Temperatures as a Function of Reactor Power.

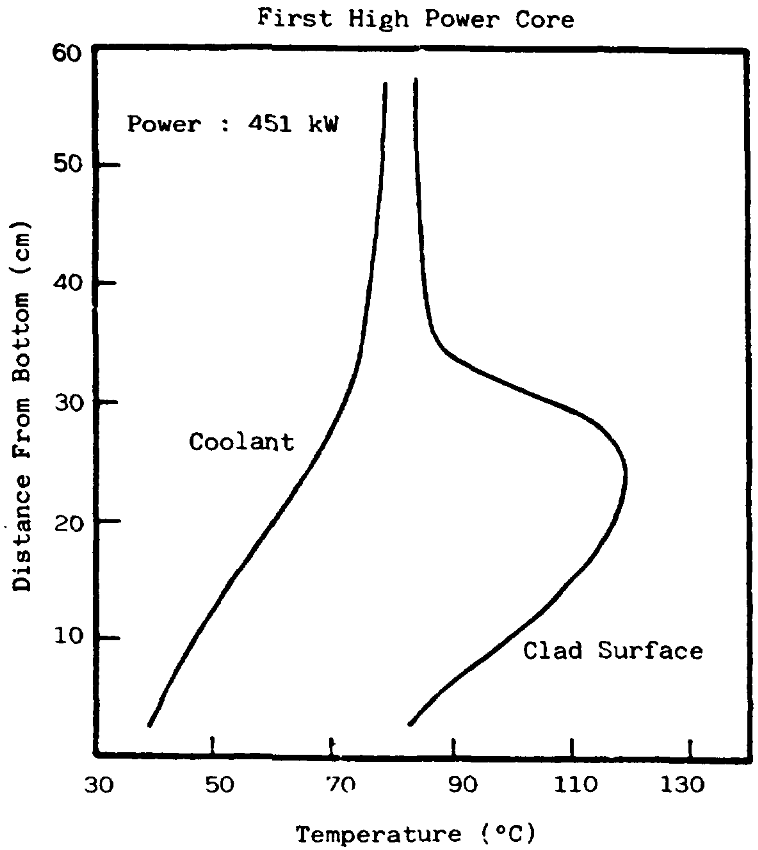

Fig. 7 Axial Temperature Distribution in the Hot Channel at ONB.

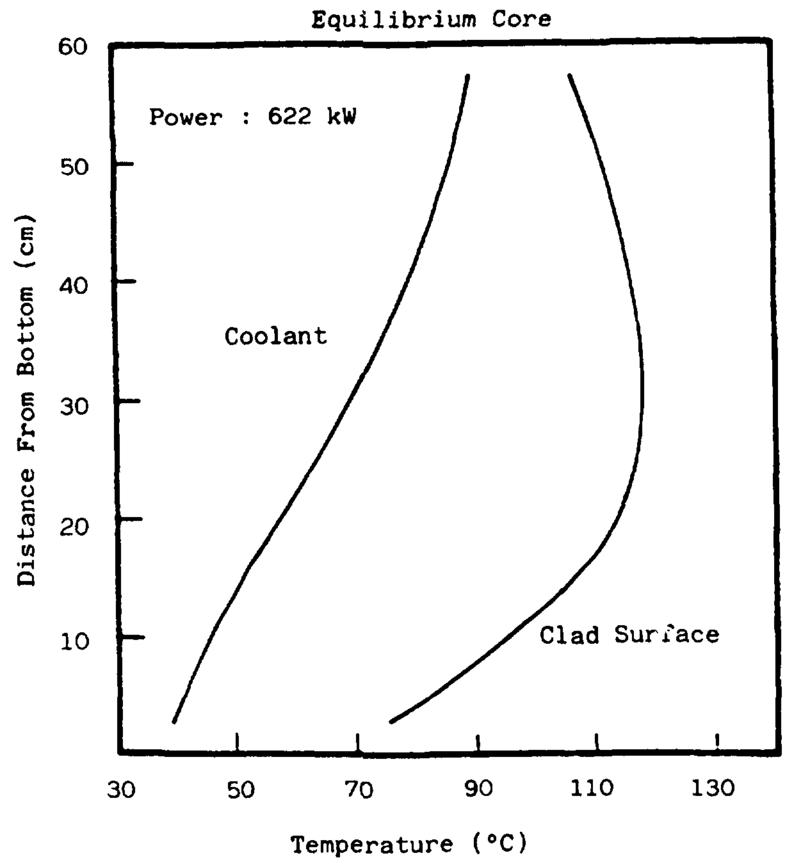

Fig. 8 Axial Temperature Distribution in the Hot Channel at ONB.

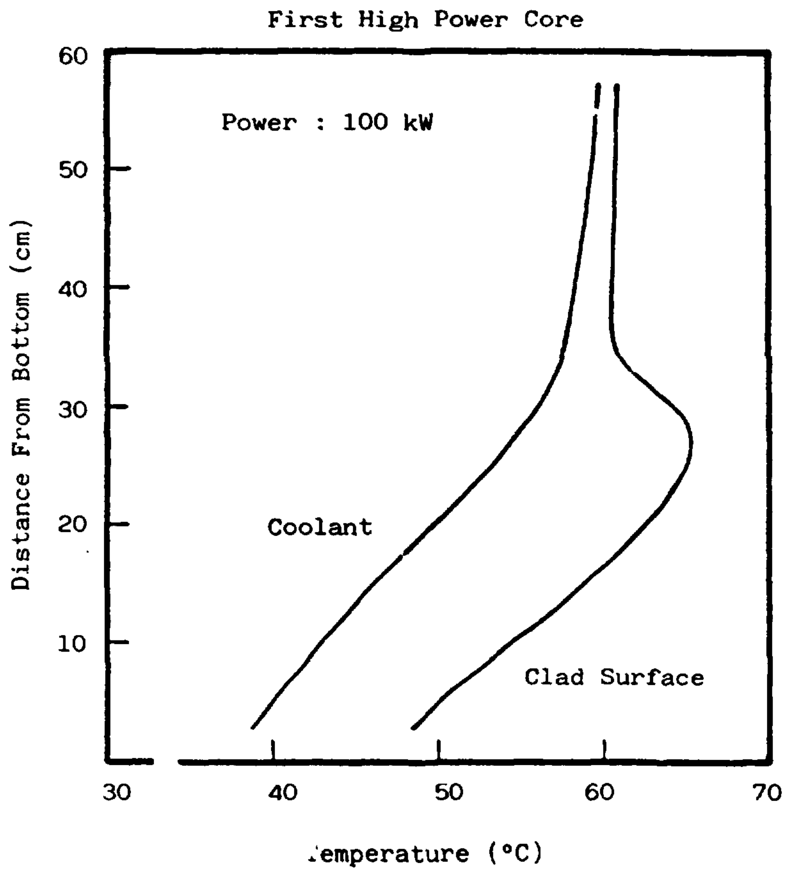

Fig. 9 Axial Temperature Distribution in the Hot Channel at Steady State Power.

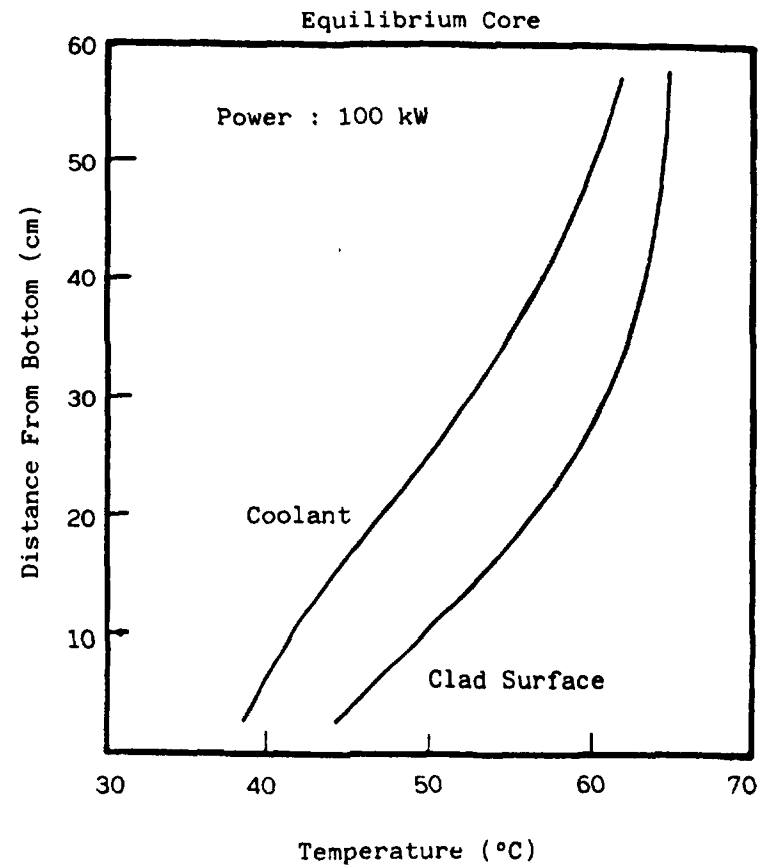

Fig. 10 Axial Temperature Distribution in the Hot Channel at Steady State Power.