HEAT-TRANSFER ANALYSIS OF THE EXISTING HEU AND PROPOSED LEU CORES OF PAKISTAN RESEARCH REACTOR

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L. A. Khan and R. Nabbi

Abstract

In connection with conversion of Pakistan Research Reactor from the use of Highly Enriched Uranium (HEU) fuel to the use of Low Enriched Uranium (LEU) fuel, steady-state thermal hydraulic analysis of both existing HEU and proposed LEU cores has been carried out. Keeping in mind the possibility of power upgrading, the performance of proposed LEU core, under 10 MW operating conditions, has also been evaluated. Computer code HEATHYD has been used for this purpose. In order to verify the reliability of the code, IAEA benchmark 2 MW reactor was analyzed.

The cooling parameters evaluated include; coolant velocity, critical velocity, pressure drop, temperature distribution in the core, heat fluxes at onset of nucleate boiling, flow instability and burnout and corresponding safety margins.

From the results of the study it can be concluded that the conversion of the core to LEU fuel will result in higher safety margins, as compared to existing HEU core, mainly because the increased number of fuel plates in the proposed design will reduce the average heat flux significantly. Anyhow upgrading of the reactor power to 10 MW will need the flow rate to be adjusted between 850 to 900 m³/hr, to achieve reasonable safety margins, at least, comparable with the existing HEU core.
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<tr>
<td>$T_{\text{out}}$</td>
<td>Water Temperature at Core Outlet</td>
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</tr>
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<td>$W$</td>
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<td>$W_h$</td>
<td>Effective Fuel Plate Width for Heat Transfer</td>
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</tr>
<tr>
<td>$W_P$</td>
<td>Total Plate Width or Chord of Curved Plate</td>
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<tr>
<td>$z$</td>
<td>Axial Location along the Channel</td>
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<td>Unit</td>
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<tr>
<td>λ</td>
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<td>KJ/Kg</td>
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<tr>
<td>μ</td>
<td>Dynamic Viscosity of Water</td>
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</tr>
<tr>
<td>ρ</td>
<td>Density of Water</td>
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<tr>
<td>ν</td>
<td>Poisson's Ratio</td>
<td>(dimensionless)</td>
</tr>
<tr>
<td>ν</td>
<td>Kinematic Viscosity of Water</td>
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1. INTRODUCTION

Pakistan Research Reactor (PARR-1) is a 5 MW, swimming pool type research reactor using 93% enriched uranium fuel. Demineralized light water is used as coolant and moderator. One side of the reactor core is reflected by two rows of graphite where as remaining sides are surrounded by light water. The reactor is operating safely and providing services to its users for the last 20 years.

Core of PARR-1 is to be converted from the use of Highly Enriched Uranium (HEU) fuel to the use of Low Enriched Uranium (LEU) fuel. In this connection steady state Thermal-Hydraulic analysis of existing HEU and proposed LEU cores has been carried out and being presented in this report. For this purpose computer code HEATHYD has been used. In order to verify the reliability of the computer code, IAEA benchmark problem /1/ was analyzed using mathematical models suggested by different organisations. Computer code was then applied to the same problem to compare the results.

After the verification, thermal-hydraulic performance of both existing HEU core (PARR-1) and proposed LEU core (PARR-2) have been evaluated using above mentioned computer code. Mathematical models were applied, in cases, where the code was not applicable.

Conversion of PARR-1 from the use of HEU fuel to the use of LEU fuel can be availed as an opportunity to upgrade the reactor power to 10 MW. One of the attractive design features of PARR-1, helpful for power upgrading, is that the flow through the core can be increased to 180% of present flow rate without any changes in the embedded piping of primary cooling circuit. Anyhow additional pumps, heat exchangers, demineralizers etc. will be needed for this purpose.

Keeping in mind the possibility of power upgrading, thermal-hydraulic analysis of proposed LEU fuel, under 10 MW operating conditions (PARR-3) has also been carried out. For the purpose of comparison with 5 MW cases, maximum possible flow rate of 180% was assumed. Anyhow performance of PARR-3 under various flow rates, ranging from 121% to 180% of the existing flow, has also been studied. Purpose of this study was to determine the optimum working flow rate for a power level of 10 MW.
2. DESCRIPTION OF DESIGN PARAMETERS OF EXISTING HEU AND PROPOSED LEU CORES OF PAKISTAN RESEARCH REACTOR

A general design description of existing HEU core of PARR-1 and proposed LEU core of PARR-2 (5 MW) and PARR-3 (10 MW) is provided in Table(4).

Briefly, the HEU core of PARR-1 utilizes 93% enriched uranium fuel in the form of UAl -Al. The core comprise of 24 standard MTR elements and 6 control elements arranged in a 6 x 5 pattern on a 6 x 9 grid. Each standard element contains 16 fuel plates plus 2 aluminium dummy plates, where as each control element contains 8 fuel plates plus 1 aluminium dummy plate and has a central gap providing passage for oval shape control rods. Fuel meat dimensions for each fuel plate are 61.0 x 0.51 x 600 mm. The coolant flow rate through the core is 536 m³/hr.

The proposed LEU core of the reactor (PARR-2) will utilize 19.75% enriched fuel in the form of $\text{U}_3\text{O}_8$ - Al. With an operating power of 5 MW the core will contain 25 standard MTR elements and 5 control elements. Each standard element will contain 23 fuel plates (with no dummy plates) and each control element will contain 17 fuel plates plus 4 aluminium dummy plates forming a passage for fork type control absorbers. Fuel meat dimensions for the proposed fuel are 6.275 x 0.51 x 600 mm. Coolant flow rate through the core will be same as in existing HEU core i.e 536 m³/hr.

The proposed 10 MW reactor (PARR-3) will have the same core configuration and fuel element design as PARR-2. To provide adequate cooling the flow rate through the core will be increased.

3. THEORY AND METHODS FOR THERMAL-HYDRAULIC ANALYSIS

Steady state operating conditions of a nuclear reactor depend on heat flux, coolant velocity, inlet water temperature and pressure. So the design of a plate type fuel element requires basic thermal-hydraulic informations such as, the heat transfer regime at which Onset of Nucleate Boiling (ONB) will occur, the Pressure Drop and Flow Rate through the fuel element, the Departure from Nucleate Boiling (DNB), the conditions for Flow Instability and the Critical Velocity beyond which fuel plates will collapse. This section outlines the approaches and correlations used to obtain these informations.

3.1 Properties of Water as a Function of Temperature

In a reactor core, temperature of water at any point depends upon the axial and radial location of the point within the core. So the physical properties of water, such as, density, specific heat, thermal conductivity and viscosity keep on changing from point to point.
The important physical properties of water as a function of its temperature are given by the following polynomials /2,3/:

\[ \rho = 1000 \left( 1.0029 - 1.5838 \times 10^{-6} T - 2.847 \times 10^{-6} T^2 \right) \]  

(1)

\[ C_p = 4.167 + 0.05 \exp(-0.0734 T) + 0.0031 \exp(0.0268 T) \]  

(2)

\[ k = 0.654 + 9.22 \times 10^{-4} (T-60) - 6.2 \times 10^{-6} (T-60)^2 \]  

(3)

For water temperature above 100°C

\[ \mathcal{J}' = 2.91 \times 10^{-3} - 2.85 \times 10^{-5} (T-100) + 3.1 \times 10^{-7} (T-100)^2 - 2.8 \times 10^{-9} (T-100)^3 + 9.6 \times 10^{-12} (T-100)^4 \]  

(4)

For water temperature less than 100°C

\[ \mathcal{J} = \mathcal{J}' - \left[ 0.85 T'^{1.01} - T'^3/25 + T'^6/2000 \right] \times 10^{-5} \]  

(5)

where

\[ \mathcal{J} = \text{kinematic viscosity of water below 100°C}, \text{ and} \]

\[ T' = 10 - T/10 \]

3.2 Coolant Average Velocity

The volumetric flow rate through an element is related to the average velocity of the coolant by:

\[ Q = 0.36 N_f U W \tau_w \]  

(6)

3.3 Pressure Drop Across Fuel Element (\( \Delta P_f \))

The total pressure drop across a fuel element consists of the losses in upper and lower end boxes of the element and the friction pressure drop across the fuel plates. These losses are in addition to the losses due to change in height between core inlet and outlet.

3.3.1 Friction Pressure Losses (\( \Delta P_f' \))

Most of the pressure drop in research reactors comes from the friction losses in the fuel channels. Two different approaches to calculate friction losses are being outlined, the second one gives about 1% higher value.
(a) Most plate type research reactors were designed for subcooled core flow under normal operation. So the friction pressure loss can be calculated from the following standard formula /4/:

\[
\Delta P_f = 4 f \left( \frac{L_c}{D_e} \right) \left( \frac{\rho U^2}{2} \right) \times 10^{-5}
\]  

(7)

For turbulent flow in smooth channels, the friction factor can be expressed as /5/:

\[
f = 0.0791 \left/ R_e^{0.25} \right. \quad \text{for} \quad 5000 < R_e < 51094
\]

(8)

\[
f = 0.0460 \left/ R_e^{0.2} \right. \quad \text{for} \quad 51094 < R_e
\]

(9)

(b) For a channel with constant geometry (hydraulic diameter) the friction losses for the whole channel are given by:

\[
\Delta P_f = A \left( \frac{L_c}{D_e} \right) \left( \frac{\rho U^2}{2} \right) \times 10^{-5}
\]

(10)

where

\[
A \quad \text{is Darcy number and can be calculated as follow:}
\]

- **Without Heating**

  For a smooth wall and for Reynolds number between 3000 and 300,000

  \[
  A = A_0 = 0.0056 + 0.5 \left\{ \frac{1}{R_e} \right\} -0.32
  \]

  (KOO-formula)

  (11)

  For Reynolds number less than 2000

  \[
  A_0 = \frac{K'}{R_e}
  \]

  where \( K' = 96 \) for rectangular channels

  (12)

  For Reynolds number between 2000 and 3000 an intermediate value is adopted.

- **With Heating**

  In a channel where one wall is heated or cooled, the friction factor is changed due to the change in viscosity near the wall.

  \[
  A = A_0 R
  \]

(13)
R results from experimental measurements /6/: 

\[ R = 1 - 0.5 \ (1+Y)^b \ \log_{10}(1+Y) + 0.04 \quad (14) \]

where

\[ Y = \left( \frac{\mu}{\mu_w} - 1 \right) \left( \frac{1}{\mu_w} \right)^{0.17} \quad (15) \]

\[ b = 0.17 - 2.0 \times 10^{-6} \ \left( \frac{a}{R_e} + 1800 \right) / R_e \quad (16) \]

where

- $\mu$ is dynamic viscosity at water temperature
- $\mu_w$ is dynamic viscosity at wall temperature.

Equation (14) is valid for circular and rectangular channels. An approximation is given by /7/: 

\[ R = 1 - (0.0047 - 0.000033 \ T) \ (T - T_w) \quad (17) \]

where wall and fluid temperatures are in degree $C^\circ$.

3.3.2 Entrance And Exit Pressure Losses ($\Delta P_{en}, \Delta P_{ex}$).

These losses come from the energy loss due to the channel and from the variation in dynamic pressure which is positive at the entrance and negative at the exit. Such losses depend on the flow rate, geometries and dimensions of the end boxes.

For a given discharge, assuming constant coolant density, the velocities in the end boxes can be related by:

\[ \frac{U_0}{U} = \frac{A_c}{A_0} \quad (18) \]

The entrance and exit pressure losses can be calculated from the following standard formula /8/: 

\[ \Delta P_{en} = K_{en} \left( \frac{\rho U^2}{2} \right) \times 10^{-5} \quad (19) \]

\[ \Delta P_{ex} = K_{ex} \left( \frac{\rho U^2}{2} \right) \times 10^{-5} \quad (20) \]

where

\[ K_{en} = \left( \frac{1}{\beta - 1} \right)^2 + 0.05 \quad , \quad \beta = 0.63 + 0.37 \left( \frac{A_c}{A_0} \right)^2 \quad (21) \]
\[ K_{ex} = (1 - A_c/A_0)^2 + 0.05 \] (22)

### 3.3.3 Pressure Losses Due to Change in Height

This pressure drop comes from change in height between core inlet and outlet and is given by the product of specific weight of the fluid and change in height. For downward flow its contribution is negative.

### 3.3.4 Dynamic Pressure Losses

It is a reversible process i.e positive at entrance and negative at exit. Dynamic pressure drop can be calculated by:

\[ \Delta P_d = (\rho U^2/2) \times 10^{-5} \] (23)

### 3.4 Saturation Temperature of Water

Saturation temperature of water as a function of its pressure, at any point 'z', along the length of the channel is given by /2,3/:

\[ T_{sat,z} = -167 \ln(1.05 - \ln(226 P_z)/167/0.065) \] (24)

for \( P_z < 4.3 \) bar

\[ = -165 \ln[ -\ln(P_z/422.23)/11.12 ] \] (25)

for \( P_z > 4.3 \) bar

### 3.5 Critical Velocity

When a high speed flow passes through a gradually narrowing passage, the pressure head of the stream is converted into velocity head, which creates a suction force on the wall. If the wall is movable or flexible, the passage will be reduced automatically to zero and the flow will be stopped. As soon as the flow becomes stagnant, the stream pressure increases to its maximum value, pushing the wall back, thus providing a wide passage again. These suction and pushing forces can act periodically and thus vibrate the core structure. This mechanism appears to govern the vibration of parallel fuel plates. These hydraulic vibrations can result in large deflection of the plates, causing local overheating and possibly a complete blockage of coolant flow.

Parallel fuel plate vibrations has been analyzed by Miller /9/, Scavuzzo /10/, and Remick /11/. Miller has derived a formula for critical velocity...
based on the interaction between the changes in cross-sectional areas, coolant velocities and pressure in two adjacent channels. According to Miller the critical velocity above which significant vibration will occur for fuel plate assemblies with long edges attached to side plates is given by:

For flat plate assemblies

\[ V_{\text{crit}} = \left[ 15 \times 10^5 \frac{E (t_p^3 - t_m^3)}{t_w} \frac{\rho}{\nu} \frac{W^4 (1 - \nu^2)}{1/2} \right]^{1/2} \]  \( (26) \)

For curved plate assemblies

\[ V_{\text{crit}} = [2 \times 10^5 \frac{E (t_p^3 - t_m^3)}{t_w} / 3 \frac{\rho}{\nu} \frac{W^4 (1 - \nu^2)}{1/2}]^{1/2} \times \]
\[ \left[ \beta_1 \sin^5 \alpha / (1/6 - \sin 2\alpha / 8\alpha + \cos 2\alpha / 12) \right]^{1/2} \]  \( (27) \)

where

\[ \beta_1 = \frac{P_1 R_1}{X_1} \]

\( P_1 \) = Pressure difference tending to bend or stretch fuel plates.

\( R_1 \) = Initial mean radius of curvature of curved plate.

\( X_1 \) = Longitudinal distance from edge support.

\( 2\alpha \) = Curved plate arc between the supports.

The ratio of critical velocity for curved plate to that for the flat plate obtained by Eq. (27) and (26), respectively, is plotted in Fig. (1).

Miller's results are frequently applied, and Wambganss /12/, who extended the work, concluded that the critical flow velocities could be from 0.63 to 0.85 times those derived by Miller.

For design purposes, ref. /13/ recommends that the coolant velocity be limited to 2/3 of the critical velocity given by Miller.

3.6 Onset of Nucleate Boiling (ONB)

Onset of Nucleate Boiling is taken as a limit for single-phase cooling and is not a limiting criterion in the design of a fuel element. It is the heat transfer regime which 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 over \( T_{sat} \) by a quantity \( T_s - T_{sat} \). So under ONB conditions, the clad surface temperature over which nucleate boiling will occur for a given local coolant pressure and surface heat flux can be expressed by the correlation developed by Bergles and Rohsenow /14/: 

\[
T_s = T_{sat} + \frac{5}{9} \left( 9.23 \frac{q}{P_z} \right) 1.156 \left( P_z^{0.0234} / 2.16 \right) \tag{28a}
\]

Rearranging above equation, the heat flux at ONB is given by:

\[
q_{ONB} = P_z^{1.156} / 9.23 \left[ 1.8 (T_s - T_{sat}) \right] (0.463 P_z^{0.0234})^{-1} \tag{28b}
\]

This correlation is applicable down to the low pressure characteristic of research and test reactors.

The local clad surface temperature can be calculated from the coolant temperature and local heat flux as follow:

\[
T_s = T_{in} + 20 \frac{W_h}{\mu} \int_0^Z q \, dz / \left( W \, G \, t \, C_\tau \, p \right) + q/h \tag{29}
\]

The second term on the right hand side of Eq.(29) is the coolant temperature rise from the channel entrance to the axial location 'Z'. The third term is the film temperature difference between the clad surface and the coolant.

By equating Eq.(28a) and (29), an expression that relates heat flux, water channel thickness and mass flux is obtained. This relationship enables calculation of maximum allowable surface heat flux without local boiling for a given channel thickness and flow conditions.

The actual axial location at which ONB will occur depends upon the axial heat flux distribution, the coolant velocity and the pressure drop along the channel.

For simplicity, heat flux at ONB can be calculated conservatively by using the worst combination of parameters i.e. ONB occurs at the channel exit with peak heat flux, lowest pressure and saturation temperature, and highest coolant temperature rise. With these assumptions, the resulting expression for Eq.(28a) and (29) becomes:

\[
T_{sat} + \frac{5}{9} \left( 9.23 \frac{q}{r_a} \frac{f_a}{f_r} \right) P_z^{1.156} \left( P_z^{0.0234} / 2.16 \right) =
T_{in} + 20 \frac{f_a}{r_a} \frac{q_H}{W_r} / \left( G \, t \, C \, W_f \, f_a \, q_a / h \right) \tag{30}
\]
The heat transfer coefficient 'h' can be derived by using the Boelter correlation /15/:

\[ h = 0.023 \left( \frac{k}{100} \frac{D_e}{D_e} \right) R_e^{0.8} P_r^{0.3} \]  
(31)

where

\[ R_e = \frac{G D_e}{100 \mu} \quad \text{(Reynolds number)} \]  
(32)

\[ P_r = \left( \frac{\mu C_p}{k} \right) \times 10^3 \quad \text{(Prandtl number)} \]  
(33)

With conservative assumptions made, the heat flux at ONB calculated from Eq.(30) will be about 15% lower than that calculated from Eqs.(28) and (29) using an iterative procedure for the exact ONB location.

3.6.1 Comparison of ONB Correlations With Experiment:

During Boiling Experiment /16/ performed in the Oak Ridge Research Reactor the coolant temperature at two axial locations near the channel exit was measured. The thermal-hydraulic conditions in this case were modelled and the coolant and clad surface temperatures were calculated using computer code COBRA-3C/ERTR /17/. The calculated coolant temperature agreed with the measured value. Then corresponding to this coolant temperature, peak clad surface temperature of 124.7 °C was calculated using COBRA-3C/ERTR. The combination of Eqs.(28) and (29) predict that ONB will occur when the clad surface temperature is higher than 121.3 °C.

The power level at which boiling commences under test thermal-hydraulic conditions calculated by Eq.(28) and (29) is about 5-6% lower than that measured during Boiling Experiment.

3.7 Flow Instability

The term 'Flow Instability' refers to flow oscillations of constant or variable amplitude that are analogous to vibrations in a 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. Ruddic /18/, found that the critical heat flux will be reduced by 40% when the flow is oscillating. This adverse affect was also found by Lowdermilk /19/. Thus for plate type fuel design, the critical heat flux that leads to the Onset of Flow Instability may be more limiting that that of stable burnout.

The most common instabilites encountered in heated channels with forced convection are the flow excursion (or Ledinegg instability) and density wave oscillation (or Dynamic instability).

Flow instability can not occur in single phase flow except through flow induced vibrations or deformations. In two phase flow, the pressure of saturated water vapour in the form of bubbles provide a new mechanism which affect the flowrate-pressure drop relationship in a complex manner. Consider a channel in which flow starts decreasing. As a result pressure drop will decrease. A stage will be reached at which power generated in the channel is high enough to form the bubbles, first small bubbles and than of larger size. The larger bubbles eventually restrict the channel cross-section, and force the liquid phase to accelerate in order to maintain the same mass flux down the channel. This acceleration, in turn, leads to an increased pressure drop which restricts flow further, enhancing the flow blockage.

For most research and test reactors, the steady state operating system pressure is low and the inlet coolant temperature is much lower than the saturation temperature. So it can be concluded that flow excursion will occur for a given flow rate at high enough heat flux, and density wave oscillation will not occur under normal operating conditions.

3.7.1 Streaming Stability Safety Factor

Whittle and Forgan /20/ measured the mass flow, exit temperature and pressure drop corresponding to minima in the pressure drop - vs- flow rate curve for subcooled water flowing (upward and downward) in narrow heated channels (width 2.54 cm, thickness 0.14 to 0.32 cm, and length 40 to 61 cm) under the following conditions:

\[ 1.2 \leq P_{\text{exit}} \leq 1.7 \text{ bar} \]

\[ 83 \leq \frac{L_H}{D_H} \leq 190 \]

where

\[ L_H = \text{Heated length of channel} \]
\[ D_H = \text{Heated equivalent diameter of the channel.} \]
\[ = 4 \left( \frac{\text{channel flow area}}{\text{channel heated perimeter}} \right) = 4 \frac{t_w W}{(t_w + W_h)} \quad (34) \]

Based on these measurements the following correlation was proposed:

\[ R = \frac{(T_{\text{out}} - T_{\text{in}})}{(T_{\text{sat}} - T_{\text{in}})} \quad (35) \]

where

\[ R = \frac{1}{1 + \eta \frac{D_H}{L_H}} \quad (36) \]

\( \eta \) is bubble detachment parameter.

Flow instability, for different geometries, has been studied by Whittle and Forgan /20,21/, Croft /22/, Maulbetsch and Griffith /23/, Waters /24/ and Grenoble /25,26,27,28,29/.

Values of \( \eta \) suggested by Bowring /30/, Costa /31/, and Levy /32/ are 12-35, 30, and 37 respectively. A value of \( \eta = 25 \) was determined as best fit for Whittle and Forgan data.

For different values of \( \eta \), \( R \) is plotted as a function of \( L_H/D_H \) in Fig. (2).

Streaming Stability Safety Factor 'S_f' is defined as:

\[ S_f = R \frac{(T_{\text{sat}} - T_{\text{in}})}{(T_{\text{out}} - T_{\text{in}})} \quad (37) \]

and gives the margin to onset of flow instability.

\subsection*{3.7.2 Average Heat Flux at Onset of Flow Instability}

Power corresponding to flow instability can be calculated from coolant temperature rise i.e. \( R(T_{\text{sat}} - T_{\text{in}}) \), specific heat, and flow rate. Then considering heat transfer area, one can determine the corresponding heat flux. Balancing the units, the average heat flux at Onset of Flow Instability can be expressed in terms of velocity, channel geometry, temperature, and fluid properties:

\[ \bar{q}_{\text{OFI}} = 0.05 \left[ R \frac{C_p}{W} \frac{t_w}{W_H} U \frac{(T_{\text{sat}} - T_{\text{in}})}{L_H} \right] \quad (38) \]

In order to clarify use of Eq. (38), following facts must be kept in mind:

- The effect of channel entrance loss, which is a strong stabilizing factor for the system, is not included in the correlation. Thus the
system could be more stable than the correlation predicts;

-Since pressure drop characteristics are not required, the accuracy of the prediction does not depend on two phase correlations (subcooled void fraction, pressure drop, and heat transfer coefficient). All two phase effects are included in parameter 'η';

-The phenomenon is sensitive to system pressure through the saturation temperature, $T_{sat}$

3.7.3 Effect of Axial Heat Distribution On Flow Instability

Flow instability is intimately related to pressure drop. The pressure drop depends on the local water quality, which follows from the axial heat distribution. Consequently some dependence on the axial heat flux distribution may be expected.

The influence of axial heat flux distribution on the onset of flow excursion was investigated experimentally by Forgan/20/, Croft /22/, Water/24/, and Courland et al /28/. The axial heat flux distribution tested include uniform, chopped cosine, and ramp at the channel exit. It was reported that the possible effects are small.

3.7.4 Effects of Various Parameters on Flow Instability

The effects of various flow instability parameters are /33/:

- High heat input aggravates flow instability because of the resulting high rate of vapourization;

- Increased inlet subcooling, within a critical value, reduces stability. Beyond that value, a further increase in subcooling may reverse the trend and stabilize the flow;

- Increased resistance at the exit strongly decreases the stability because it increases the exit pressure drop when a large void is generated in the channel;

- Increased resistance at the inlet strongly increases the stability. When the flow is slowed by a disturbance, a large pressure head will be retained at the inlet and will be available for forcing the fluid through the channel thereby stabilizing the flow;

- Increased system pressure increases the stability in proportion to the increase in the reduced pressure, primarily because of the increase in saturation temperature;

12
Increased pressure at core head in a long vertical channel increases the stability of upward boiling flow and decreases the stability of downward boiling flow. This occurs because the buoyancy of the vapour aids upward flow and opposes downward flow. With the later, there is a tendency for the vapour to agglomerate and remain in the channel.

3.8 Departure From Nucleate Boiling (DNB)

For reactor design purpose, acceptable data on burnout heat flux are needed since Departure from Nucleate Boiling is potentially a limiting design constraint. Data for DNB correlations applicable to low pressure plate-type research and test reactors having rectangular channels are very limited. Most of the DNB correlations are round tube. A round tube DNB correlation (Labuntsov) and a narrow channel correlation (Mirshak), applicable in low pressure range and recommended by most of the authors are being presented with their range of applicability.

3.8.1 The Labuntsov Correlation

The Labuntsov correlation is based on experimental data from several sources. These data covers a wide range of velocity and pressure, but all have positive subcooling at the channel exit. Labuntsov observed that the burnout heat fluxes are determined by the pressure, coolant velocity, and the magnitude of subcooling at the exit and that these fluxes are virtually independent of the length, diameter, and configuration of the operating channel. The effect of the channel dimensions becomes pronounced only for diameters that are less than 2 mm. So it can be said that Labuntsov correlation is independent of channel geometry. According to Labuntsov critical heat flux is given by:

\[ q_c = 145.4 \cdot 0(p) \left[ 1 + 2.5 \cdot \frac{U^2}{\Theta(p)} \right]^{1/4} \left( 1 + 15.1 \cdot \frac{\Delta T_{\text{sub}}}{\chi \cdot P} \right)^{1/2} \]  \hspace{1cm} (39)

where

\[ \Theta(p) = 0.99531 \cdot P^{1/3} \cdot (1 - \frac{P}{P_c})^{4/5} \]  \hspace{1cm} (40)

\[ \Delta T_{\text{sub}} = T_{\text{sat}} - T_{\text{in}} - \Delta T_c \]  \hspace{1cm} (41)

The above relation is valid with in the parameter ranges given below:

- **Steam quality**: negative-0
- **Velocity**: 0.7 to 45 m/sec.
- **Pressure**: 1.0 to 200 bar absolute
- **Subcooling\(\Delta T_{\text{sub}}\)**: 0 to 240°C
- **\(q_c\)**: 116 to 5234 W/cm²
3.8.2 The Mirshak Correlation [35]:

The Mirshak correlation is based on data from annular channels (with heated tube diameter of 1.27 cm and 2.03 cm) and rectangular channels (with channel width of 6.4 cm, heated strip width of 5.08 cm, channel thickness from 0.3 to 0.58 cm). For both test sections, only one side of the channel was heated. All data correlated have positive subcooling at the channel exit. According to Mirshak critical heat flux is given by:

\[
q_c = 151 (1 +0.1198 U) (1 + 0.00914 \Delta T_{sub}) (1 + 0.19 P)
\]

where

\[
\Delta T_{sub} = T_{sat} - T_{in} - T_c
\]

The above correlation is valid within the parameter ranges given below:

Steam quality: negative
Velocity: 1.5 to 13.7 m/sec
Subcooling(\(\Delta T_{sub}\)): 5 to 75°C
Pressure: 1.72 to 5.86 bar absolute
Equivalent Diameter: 0.53 to 0.17 cm
\(q_c\): 284 to 1022 \(\text{w/cm}^2\)

3.8.3 Water Subcooling

In both the Labuntsov and Mirshak correlations, the burnout heat flux depends on the water subcooling and vice versa. For this reason the determination of the critical heat flux requires an iterative procedure. From an energy balance for a rectangular channel, the water subcooling can be expressed as a function of channel heat flux, channel geometry, coolant velocity and coolant properties:

\[
\Delta T_{sub} = T_{sat} - T_{in} - \Delta T_c
\]

\[
= T_{sat} - T_{in} - 20 H_{co} \frac{W}{\rho C_p} t_w f U
\]

(43)

By substituting this expression into the Labuntsov and Mirshak correlations, the burnout heat fluxes can be derived as a function of coolant velocity for a given inlet water temperature, system pressure and channel geometry.
3.8.4 Extrapolation of the Labuntsov and Mirshak Correlations With Zero Subcooling at Low Velocities

For very low velocities, the lower limit of heat flux can be estimated from the pool boiling case. The pool boiling heat flux, as given by Rohsenow and Griffith correlation /36/ is as follow:

\[ q_{c, \text{pool boiling}} = 1.21 \times 10^{-3} \rho_0 \lambda \left( \frac{\rho_1 - \rho_\lambda}{\rho_0} \right)^{0.6} \]  \hspace{1cm} (44)

Where \( \rho_1 \) and \( \rho_\lambda \) are liquid and steam densities respectively, and \( \lambda \) is the heat of vaporization.

For the low coolant velocities at which subcooling is negative, the burnout heat fluxes can be reasonably estimated by using Labuntsov and Mirshak correlations with zero subcooling i.e \( \Delta T_{\sub} = 0 \). In such a case the lower limit of burnout heat fluxes agrees well with that of pool boiling /1/.

3.8.5 General Remarks

One thing which must be kept in mind is that all DNB correlations are based on data from uniformly heated channels. In reactor core, however, the heat flux varies along the length of the channel. One result of nonuniform heating is that burnout does not always occur at the channel exit as it does with uniform heating, provided instabilities are avoided. For lack of a better alternative, one can conservatively assume that the burnout heat flux predicted by the uniform profile correlations is equal to the peak (maximum) heat flux in a channel with a non-uniform profile. For correlations which depend on water subcooling (i.e., lower burnout heat flux for lower subcooling), one can further conservatively assume that DNB occurs at the channel exit, where the water subcooling is the lowest.

DNB is a complex phenomenon even for a simple channel geometry. The burnout heat flux depends on a number of variables, such as pressure, channel geometry, coolant velocity, and coolant inlet or exit conditions. The experimental data from which the DNB correlations are developed are scattered in the space of DNB dependent variables. For example, some data are in positive subcooling region while others are in positive steam quality region; some data are limited to only one system pressure and/or one inlet water temperature. So far, a complete data set, which is applicable to plate type fuel channels and covers the low pressure range and exit conditions from positive subcooling to positive steam quality, has not been found. Consequently, engineering judgements and cautions are required in using the above DNB correlations to estimate the burnout heat flux in plate type fuel channels, especially when the applied conditions are outside the range of nominal applicability of the correlations.
A specific single correlation can not be recommended as 'Best' for all research reactors. Instead, applicable correlation must be considered to see how it performs at conditions of interest. Improved accuracy ultimately depends upon specific tests in the geometry of interest, in mockups, or through in-plant tests.

4. COMPUTER CODE 'HEATHYD'

The computer code HEATHYD was used for steady state thermal-hydraulic analysis. This code has been developed for Thermal-Hydraulic analysis of MTR type fuel element.

The major items of input to the code include: fuel element geometry, power generated and flow through the element, axial and radial peaking factors, coolant properties, static pressure etc.

Code output includes: coolant velocity, pressure drop, saturation pressure and temperature, clad surface and coolant temperature, local heat flux, margin to onset of nucleate boiling and burnout, coolant temperature at burnout and streaming stability safety factor.

5. THERMAL HYDRAULIC ANALYSIS OF IAEA 2 MW REACTOR

A general description of the design parameters of the 2 MW reactor is provided in Table(1). Briefly, the core is assumed to contain 21 standard MTR elements and four control elements. Each standard element contains 19 fuel plates with 0.51 mm thick meat. Each control element contains 15 fuel plates also with 0.51 mm thick meat.

This reactor has been analyzed using Mathematical Models provided in section 3 and the computer code HEATHYD (section 4). The purpose of benchmark calculations was to check how closely the results obtained are when the calculations are run for identical cases.

In addition to Table(1), Table(2) shows the input parameters used in the calculations. The results of benchmark 2 MW reactor, obtained by two different approaches, are summarized in Table(3).

5.1 Conclusions

The results obtained for IAEA 2 MW reactor (see Table.3), by two different approaches, show that there is very good agreement between the computer code and mathematical models. As a result of this study, it is concluded that the computer code HEATHYD is reliable and can be applied, with fairly good degree of confidence, to evaluate the thermal-hydraulic performance of existing HEU and proposed LEU cores of Pakistan Research Reactor.
6. THERMAL HYDRAULIC ANALYSIS OF PAKISTAN RESEARCH REACTOR

A general design description of existing HEU core of PARR-1 and proposed LEU core of PARR-2 (5 MW) and PARR-3 (10 MW) is provided in Table(4). All the three cases have been analyzed using computer code HEATHYD. In addition mathematical models were applied to determine critical velocity and heat flux at onset of flow instability.

A flow rate of 536 m³/hr was considered for a power level of 5 MW, whereas the performance of the proposed core, under 10 MW operating conditions, was evaluated under various flow rates ranging from 650 m³/hr to 965 m³/hr. The purpose of this study was to determine optimum working flow rate for 10 MW power level.

In addition to Table(4), Table(5), (6) and (7) show the input parameters used in the calculations. Axial flux distribution was assumed to follow a cosine shape with an extrapolation distance of 8 cm. For the hottest channel in the core, the radial peaking factor was assumed to be 1.66.

6.1 Calculation Methods And Conclusions

The results of the thermal hydraulic analysis are summarized in Table (8) and (9). Where as Fig.(3) through (28) give their graphical representation. The methods of calculation and interesting points of the study are outlined in the following sub-sections.

6.1.1 Coolant Flow Rate and Velocity

Volumetric flow rate through the fuel element is required as input to the computer code HEATHYD. For a power level of 5 MW, the flow rate through the core was taken to be 536 m³/hr which is exactly the same as in case of existing HEU core. Anyhow for a power level of 10 MW, various flow rates ranging from 650 m³/hr to 965 m³/hr were considered. In all the three cases 10% bypass flow was assumed. It was further assumed that the flow rate through standard and control fuel elements is the same.

From the results obtained it is concluded that for the same flow rate, coolant velocity through proposed fuel will be higher because of its smaller flow area. Anyhow keeping the flow rate and plate thickness constant, reduction in number of fuel plates will result in reduction of coolant velocity.
6.1.2 Pressure Drop

For a fixed flow rate and fuel geometry, pressure drop was calculated using the computer code. The entrance pressure loss coefficient, which is required as input to the code, was calculated using Eq. (21). As shown in Fig. (4), a thicker water channel requires a lower pressure drop for the same coolant velocity.

6.1.3 Critical Velocity

Critical velocities for existing and proposed fuel were determined using Eq. (27) and (26) respectively. The values obtained were reduced by a factor of 2/3 as recommended by reference /13/. Critical velocity for proposed fuel is lower because of its small water channel thickness and slightly larger plate width.

For a fixed plate thickness, if the number of plates per element is reduced, a higher critical velocity will be achieved because of increased water channel thickness. On the other hand, for a fixed water channel thickness, reduction in number of plates per element corresponds to thicker, more rigid plates, and therefore, a higher critical velocity.

6.1.4 Axial Temperature Distribution

Coolant and clad surface temperatures, along the channel length, were determined using HEATHYD. It was observed that, for the same power and flow rate, proposed LEU fuel has lower surface heat flux and clad surface temperature but almost the same coolant temperature rise across the channel. So it can be concluded that, for the same power level (average heat flux x heat transfer area), the design with fewer plates will have higher average heat flux and higher clad surface temperature. But for a fixed power and flow rate coolant temperature rise along the channel will remain unchanged.

6.1.5 Heat Flux at Onset of Nucleate Boiling

The axial distribution of heat flux at ONB was calculated using HEATHYD. In these calculations pressure and saturation temperature of the coolant was assumed to be constant along channel length.

The actual axial location at which ONB will occur depends upon axial heat flux distribution, the coolant velocity and pressure drop along the channel. The location of ONB and corresponding heat flux and clad surface temperature were obtained by iterative procedure, using computer code and Eq. (28).
Heat flux at ONB can be increased by increasing the coolant velocity. But for same coolant velocity, a thicker water channel allows higher heat flux at ONB.

6.1.6 Heat Flux at Onset of Flow Instability

The limiting heat flux at onset of flow instability was calculated using Forgan correlation (Eq. (36) and (38) with $\eta = 25$). For a given system pressure and inlet coolant temperature, this heat flux is proportional to the coolant velocity and channel thickness i.e. for the same water channel thickness a design with higher coolant velocity or for the same coolant velocity, a design with a thicker water channel will have higher heat flux at onset of flow instability.

6.1.7 Burnout Heat Flux

Burnout heat fluxes have been calculated using the code. Both Labuntsov and Mirshak correlations (see section 3.8) were used. Velocities in case of PARR-1 and PARR-2, and pressure in all the three cases is outside the range of applicability of Mirshak correlation. In addition to this, for all the three cases under study, the exit subcooling is negative which is outside the range of applicability of both the Labuntsov and Mirshak correlations. Under such circumstances the burnout heat fluxes were estimated using these correlations extrapolated with zero subcooling.

In general, for the same water channel thickness, burnout heat flux increases as the coolant velocity increases.

7. DISCUSSION

The results of the study show that proposed LEU core of Pakistan Research Reactor will have higher safety margins as compared to existing HEU core. Insipite the fact that LEU core will have lower heat fluxes at Onset of Nucleate Boiling and Flow Instability mainly due to reduction in water channel thickness, still the safety margins will be higher than HEU core because of lower average heat flux in the proposed fuel.

The Burnout heat flux obtained for LEU fuel is higher because of slight increase in coolant velocity with almost negligible change in coolant pressure. As described in section 6.1.7, most of the parameters are outside the range of applicability of the correlations used for this purpose. So more detailed analysis is needed to obtain the final results.

The existing flow rate will provide adequate cooling for proposed core under 5 MW operating conditions. Anyhow extreme care is to be taken to ensure the proper cooling of end fuel plates.
Upgrading of the reactor to 10 MW will require the flow rate to be adjusted between 850 m³/hr to 900 m³/hr in order to achieve reasonable safety margins, at least comparable, with the existing HEU core. Heat fluxes at Onset of Nucleate Boiling, Flow Instability and Burnout, within the recommended flow range, will be significantly higher than corresponding 5 MW cases, but due to increase in average heat flux, the safety margins will be comparable.
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| Description of Design Parameters of IAEA 2 MW Reactor Used in Benchmark Calculations |
|---|---|
| **Reactor Type** | Pool Type MTR |
| **Steady State Power Level (MW)** | 2 |
| **Fuel** | UAl -Al |
| **Fuel Enrichment** | 93% |
| **Fuel Element Dimensions (mm)** | 76 × 80 × 600 |
| **Number of Fuel Elements in the Core** | 25 |
| (a) **Standard Fuel Elements** | 21 |
| (b) **Control Fuel Elements** | 4 |
| **Number of Fuel Plates in:** | |
| (a) **Standard Fuel Element** | 19 |
| (b) **Control Fuel Element** | 15 |
| **Shape of the Plates** | Straight |
| **Thickness of the Plates (mm)** | |
| (a) **Inner Plates** | 1.27 |
| (b) **Outer Plates** | 1.50 |
| **Total Width of the Plate (mm)** | 66.4 |
| **Fuel Meat Dimensions (mm)** | 0.51 × 63 × 600 |
| **Thickness of the Clad (mm)** | |
| (a) **Inner Plates** | 0.38 |
| (b) **Outer Plates** | 0.495 |
| **Water Channel Thickness (mm)** | 2.916 |
| **Coolant Flow Rate (m³/hr)** | 311 |
| **Coolant Inlet Temperature(C°)** | 38 |
TABLE-2

Input Values Used in the Thermal-Hydraulic Analysis of IAEA 2 MW Reactor

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value Used</th>
<th>Parameter</th>
<th>Value Used</th>
</tr>
</thead>
<tbody>
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<td>ν</td>
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<td>ρ (Kg/m^3)</td>
<td>990.2</td>
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</tr>
<tr>
<td>λ (KJ/Kg)</td>
<td>2203.2</td>
<td>K_{en}</td>
<td>0.1010</td>
</tr>
<tr>
<td>E (bar)</td>
<td>0.7306 x10^6</td>
<td>K_{ex}</td>
<td>0.1349</td>
</tr>
</tbody>
</table>

* For the purpose of comparison of results, these values have been taken from reference /1/.
<table>
<thead>
<tr>
<th>Design Parameter</th>
<th>Results Obtained Using Mathematical Models</th>
<th>Results Obtained Using Computer Code</th>
</tr>
</thead>
<tbody>
<tr>
<td>Volumetric Flow per Element (m³/hr)</td>
<td>12.45</td>
<td>12.45</td>
</tr>
<tr>
<td>Coolant Velocity (m/s)</td>
<td>0.94</td>
<td>0.94</td>
</tr>
<tr>
<td>Critical Velocity (m/s)</td>
<td>12.59</td>
<td>Not available</td>
</tr>
<tr>
<td>Pressure Drop (bar)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Entrance</td>
<td>$4.42 \times 10^{-4}$</td>
<td>$4.41 \times 10^{-4}$</td>
</tr>
<tr>
<td>(b) Friction</td>
<td>$1.61 \times 10^{-2}$</td>
<td>$1.54 \times 10^{-2}$</td>
</tr>
<tr>
<td>(c) Exit</td>
<td>$5.90 \times 10^{-4}$</td>
<td>Not available</td>
</tr>
<tr>
<td>(d) Dynamic</td>
<td>$4.37 \times 10^{-3}$</td>
<td>$4.37 \times 10^{-3}$</td>
</tr>
<tr>
<td>Pressure at Channel Exit (bar)</td>
<td>1.961</td>
<td>1.961</td>
</tr>
<tr>
<td>Saturation Temperature at Channel Exit (C⁰)</td>
<td>119.60</td>
<td>119.60</td>
</tr>
<tr>
<td>Coolant Temperature Rise-Across the Core (C⁰)</td>
<td>5.78</td>
<td>5.75</td>
</tr>
<tr>
<td>Average Heat Flux (watt/cm²)</td>
<td>5.76</td>
<td>5.75</td>
</tr>
<tr>
<td>Peak Heat Flux (watt/cm²)</td>
<td>18.20</td>
<td>18.20</td>
</tr>
<tr>
<td>Average Heat Flux at ONB (watt/cm²)</td>
<td>11.44</td>
<td>12.05</td>
</tr>
<tr>
<td>Limiting Heat Flux at Onset of Flow Instability (watt/cm²)</td>
<td>102.67</td>
<td>100.23</td>
</tr>
<tr>
<td>Burnout Heat Flux (watt/cm²)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Labuntsov</td>
<td>231.2</td>
<td>223.5</td>
</tr>
<tr>
<td>(b) Mirshak</td>
<td>230.6</td>
<td>230.6</td>
</tr>
<tr>
<td>Pool Boiling Peak Heat Flux (watt/cm²)</td>
<td>169.7</td>
<td>Not available</td>
</tr>
<tr>
<td>Safety Margins:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Margin to ONB</td>
<td>1.986</td>
<td>1.984</td>
</tr>
<tr>
<td>(b) Margin to Onset of Flow Instability</td>
<td>5.64</td>
<td>5.57</td>
</tr>
<tr>
<td>(c) Margin to DNB</td>
<td></td>
<td></td>
</tr>
<tr>
<td>-Labuntsov</td>
<td>12.70</td>
<td>12.28</td>
</tr>
<tr>
<td>-Mirshak</td>
<td>12.67</td>
<td>12.67</td>
</tr>
</tbody>
</table>
TABLE-4

Description of Design Parameters of Existing HEU and Proposed LEU Cores of Pakistan Research Reactor

<table>
<thead>
<tr>
<th>Design Parameter</th>
<th>Existing HEU Core (PARR-1)</th>
<th>Proposed LEU Core PARR-2 (PARR-3)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Type</td>
<td>Pool Type MTR</td>
<td>Pool Type MTR</td>
</tr>
<tr>
<td>Steady State Power Level (MW)</td>
<td>5</td>
<td>5 (10)</td>
</tr>
<tr>
<td>Fuel</td>
<td>$^{x}_{\text{UA1-Al}}$</td>
<td>$^{3.8}_{\text{U,0-Al}}$</td>
</tr>
<tr>
<td>Fuel enrichment</td>
<td>93%</td>
<td>19.75%</td>
</tr>
<tr>
<td>Fuel Element Dimensions (mm)</td>
<td>$79.6 \times 75.9$</td>
<td>$80.6 \times 76.1$</td>
</tr>
<tr>
<td>Number of fuel Elements in the Core</td>
<td>30</td>
<td>30</td>
</tr>
<tr>
<td>(a) Standard Fuel Elements</td>
<td>24</td>
<td>25</td>
</tr>
<tr>
<td>(b) Control Fuel Elements</td>
<td>6</td>
<td>5</td>
</tr>
<tr>
<td>Number of Fuel Plates in:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Standard Fuel Element</td>
<td>16</td>
<td>23</td>
</tr>
<tr>
<td>(b) Control Fuel Element</td>
<td>8</td>
<td>17</td>
</tr>
<tr>
<td>Number of Dummy Plates in:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Standard Fuel Element</td>
<td>2</td>
<td>Nil</td>
</tr>
<tr>
<td>(b) Control Fuel Element</td>
<td>1</td>
<td>4</td>
</tr>
<tr>
<td>Shape of the Plates</td>
<td>Curved</td>
<td>Straight</td>
</tr>
<tr>
<td>Thickness of the Plates (mm)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Inner Plates</td>
<td>1.27</td>
<td>1.27</td>
</tr>
<tr>
<td>(b) Outer Plates</td>
<td>1.27</td>
<td>1.50</td>
</tr>
<tr>
<td>Total Width of the Plate (mm)</td>
<td>66.48</td>
<td>67.1</td>
</tr>
<tr>
<td>Fuel Meat Dimensions (mm)</td>
<td>$0.51 \times 61 \times 600$</td>
<td>$0.51 \times 62.75 \times 600$</td>
</tr>
<tr>
<td>Thickness of the Clad (mm)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Inner Plates</td>
<td>0.38</td>
<td>0.38</td>
</tr>
<tr>
<td>(b) Outer Plates</td>
<td>0.38</td>
<td>0.458</td>
</tr>
<tr>
<td>Water Channel Thickness (mm)</td>
<td>3.12</td>
<td>2.23</td>
</tr>
<tr>
<td>Coolant Flow Rate (lt/hr)</td>
<td>536</td>
<td>536 (965 for 10 MW)</td>
</tr>
<tr>
<td>Coolant Inlet Temperature (°C)</td>
<td>32</td>
<td>32</td>
</tr>
</tbody>
</table>
**TABLE-5**

Input Values Used in Thermal-Hydraulic Analysis of PARR-1

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value Used</th>
<th>Parameter</th>
<th>Value Used</th>
</tr>
</thead>
<tbody>
<tr>
<td>P (bar abs)</td>
<td>1.603</td>
<td>ν</td>
<td>0.33</td>
</tr>
<tr>
<td>T&lt;sub&gt;sat&lt;/sub&gt; (°C)</td>
<td>113.4</td>
<td>ρ (Kg/m³)</td>
<td>993.5</td>
</tr>
<tr>
<td>T&lt;sub&gt;in&lt;/sub&gt; (°C)</td>
<td>32</td>
<td>μ (Pascal-sec)</td>
<td>6.0507 x10⁻⁴</td>
</tr>
<tr>
<td>f&lt;sub&gt;a&lt;/sub&gt;</td>
<td>Cosine</td>
<td>k (w/mK)</td>
<td>0.628</td>
</tr>
<tr>
<td>f&lt;sub&gt;r&lt;/sub&gt;</td>
<td>1.66</td>
<td>C&lt;sub&gt;p&lt;/sub&gt; (KJ/Kg-K)</td>
<td>4.177</td>
</tr>
<tr>
<td>λ (KJ/Kg)</td>
<td>2217.2</td>
<td>K&lt;sub&gt;en&lt;/sub&gt;</td>
<td>0.1043</td>
</tr>
<tr>
<td>E (bar)</td>
<td>0.7306 x10⁶</td>
<td>K&lt;sub&gt;ex&lt;/sub&gt;</td>
<td>0.1404</td>
</tr>
</tbody>
</table>

* Water properties are evaluated at mean coolant temperature. For 5 MW these are evaluated at 36 °C.*
### TABLE-6

**Input Values Used in the Thermal-Hydraulic Analysis of PARR-2**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value Used</th>
<th>Parameter</th>
<th>Value Used</th>
</tr>
</thead>
<tbody>
<tr>
<td>P (bar abs)</td>
<td>1.586</td>
<td>ν</td>
<td>0.33</td>
</tr>
<tr>
<td>T&lt;sub&gt;sat&lt;/sub&gt; (°C)</td>
<td>113.1</td>
<td>ρ (kg/m&lt;sup&gt;3&lt;/sup&gt;)</td>
<td>993.5</td>
</tr>
<tr>
<td>T&lt;sub&gt;in&lt;/sub&gt; (°C)</td>
<td>32</td>
<td>μ (Pascale-sec)</td>
<td>6.0507 x10&lt;sup&gt;-4&lt;/sup&gt;</td>
</tr>
<tr>
<td>f&lt;sub&gt;a&lt;/sub&gt;</td>
<td>Cosine</td>
<td>k (w/mK)</td>
<td>0.628</td>
</tr>
<tr>
<td>f&lt;sub&gt;r&lt;/sub&gt;</td>
<td>1.66</td>
<td>C&lt;sub&gt;p&lt;/sub&gt; (KJ/Kg-K)</td>
<td>4.177</td>
</tr>
<tr>
<td>λ (KJ/Kg)</td>
<td>2217.2</td>
<td>K&lt;sub&gt;en&lt;/sub&gt;</td>
<td>0.1356</td>
</tr>
<tr>
<td>E (bar)</td>
<td>0.7306 x10&lt;sup&gt;6&lt;/sup&gt;</td>
<td>K&lt;sub&gt;ex&lt;/sub&gt;</td>
<td>0.1920</td>
</tr>
</tbody>
</table>

*Water properties are evaluated at mean coolant temperature. For 5 MW these are evaluated at 36 °C*
### TABLE-7

**Input Values Used in the Thermal-Hydraulic Analysis of PARR-3**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value Used</th>
<th>Parameter</th>
<th>Value Used</th>
</tr>
</thead>
<tbody>
<tr>
<td>$P$ (bar abs)</td>
<td>1.485</td>
<td>$\nu$</td>
<td>0.33</td>
</tr>
<tr>
<td>$T_{sat}$ ($^\circ C$)</td>
<td>111.1</td>
<td>$\rho$ (Kg/m$^3$)</td>
<td>992.0</td>
</tr>
<tr>
<td>$T_{in}$ ($^\circ C$)</td>
<td>32</td>
<td>$\mu$ (Pascal-sec)</td>
<td>$6.0507 \times 10^{-4}$</td>
</tr>
<tr>
<td>$f_a$</td>
<td>Cosine</td>
<td>$k$ ($\omega$/mK)</td>
<td>0.633</td>
</tr>
<tr>
<td>$f_r$</td>
<td>1.66</td>
<td>$\lambda$ (KJ/Kg)</td>
<td>4.178</td>
</tr>
<tr>
<td>$\lambda$ (KJ/Kg)</td>
<td>2222.7</td>
<td>$K_{en}$</td>
<td>0.1356</td>
</tr>
<tr>
<td>$E$ (bar)</td>
<td>$0.7306 \times 10^6$</td>
<td>$K_{ex}$</td>
<td>0.1920</td>
</tr>
</tbody>
</table>

* Water properties are evaluated at mean coolant temperature. For 10 MW these are evaluated at 40 $^\circ C$. 
### TABLE-8

Thermal Hydraulic Analysis of HEU and LEU Cores of Pakistan Research Reactor

<table>
<thead>
<tr>
<th>Core Parameter</th>
<th>Existing HEU Core</th>
<th>Proposed LEU Core</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>5 MW (PARR-1)</td>
<td>5 MW (PARR-2)</td>
</tr>
<tr>
<td></td>
<td>10 MW (PARR-3)</td>
<td></td>
</tr>
<tr>
<td>Volumetric Flow per Element (m/ hr)</td>
<td>16.08</td>
<td>16.08</td>
</tr>
<tr>
<td>Coolant Velocity (m/s)</td>
<td>1.27</td>
<td>1.36</td>
</tr>
<tr>
<td>Critical Velocity (m/s)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Inner Plates</td>
<td>13.0</td>
<td>10.77</td>
</tr>
<tr>
<td>(b) Outer Plates</td>
<td>14.01</td>
<td>14.01</td>
</tr>
<tr>
<td>Pressure Drop (bar)</td>
<td>8.333E-04</td>
<td>1.255E-03</td>
</tr>
<tr>
<td></td>
<td>2.454E-02</td>
<td>4.184E-02</td>
</tr>
<tr>
<td>(b) Friction</td>
<td>1.119E-03</td>
<td>1.756E-03</td>
</tr>
<tr>
<td>(c) Exit</td>
<td>7.980E-03</td>
<td>9.147E-03</td>
</tr>
<tr>
<td>(d) Dynamic</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure at Channel Exit (bar)</td>
<td>1.603</td>
<td>1.586</td>
</tr>
<tr>
<td>Saturation Temperature at Channel Exit (°C)</td>
<td>113.4</td>
<td>113.05</td>
</tr>
<tr>
<td>Coolant Temperature Rise Across:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Element Shearing Average Power</td>
<td>9.96</td>
<td>9.47</td>
</tr>
<tr>
<td>(b) Element Shearing Peak Power</td>
<td>16.39</td>
<td>15.63</td>
</tr>
<tr>
<td>(c) Core (including bypass flow)</td>
<td>8.08</td>
<td>8.04</td>
</tr>
<tr>
<td>Peak Clad Temperature (°C)</td>
<td>86.18</td>
<td>11.55</td>
</tr>
<tr>
<td>Average Heat Flux (w/cm²)</td>
<td>15.8</td>
<td>10.16</td>
</tr>
<tr>
<td>Peak Heat Flux (w/cm²)</td>
<td>34.37</td>
<td>22.11</td>
</tr>
<tr>
<td>Onset of Nucleate Boiling:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Peak Heat Flux at ONB (w/cm²)</td>
<td>61.88</td>
<td>55.81</td>
</tr>
<tr>
<td>(b) Clad Temperature at ONB (°C)</td>
<td>122.3</td>
<td>121.8</td>
</tr>
<tr>
<td>(c) Location of ONB (cm, down mid plane)</td>
<td>3</td>
<td>6</td>
</tr>
<tr>
<td>Limiting Heat Flux at Onset of Flow Instability (w/cm²)</td>
<td>125.43</td>
<td>99.9</td>
</tr>
<tr>
<td>Burnout Heat Flux (w/cm²)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Labuntsov</td>
<td>237</td>
<td>243</td>
</tr>
<tr>
<td>(b) Mirshak</td>
<td>227</td>
<td>228</td>
</tr>
<tr>
<td>Safety Margins:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(a) Margin to ONB</td>
<td>1.8</td>
<td>2.6</td>
</tr>
<tr>
<td>(b) Margin to Onset of Flow Instability</td>
<td>3.65</td>
<td>4.52</td>
</tr>
<tr>
<td>(c) Margin to DNB</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- Labuntsov</td>
<td>6.9</td>
<td>10.99</td>
</tr>
<tr>
<td>- Mirshak</td>
<td>6.6</td>
<td>10.31</td>
</tr>
</tbody>
</table>

* Read as 8.333x10
<table>
<thead>
<tr>
<th>Volumetric Flow Through Core (m³/hr)</th>
<th>650</th>
<th>700</th>
<th>750</th>
<th>800</th>
<th>850</th>
<th>900</th>
<th>950</th>
<th>1000</th>
</tr>
</thead>
<tbody>
<tr>
<td>Volumetric Flow per Element (m³/hr)</td>
<td>1.647</td>
<td>2.25</td>
<td>2.85</td>
<td>3.45</td>
<td>4.05</td>
<td>4.65</td>
<td>5.25</td>
<td>5.85</td>
</tr>
<tr>
<td>Pressure Drop (bar)</td>
<td>1.774</td>
<td>2.20</td>
<td>2.70</td>
<td>3.20</td>
<td>3.70</td>
<td>4.20</td>
<td>4.70</td>
<td>5.20</td>
</tr>
<tr>
<td>Friction Coefficient</td>
<td>3.90E-2</td>
<td>5.80E-2</td>
<td>7.70E-2</td>
<td>9.60E-2</td>
<td>1.15E-1</td>
<td>1.34E-1</td>
<td>1.53E-1</td>
<td>1.72E-1</td>
</tr>
<tr>
<td>Gravitational Force</td>
<td>5.80E-2</td>
<td>7.70E-2</td>
<td>9.60E-2</td>
<td>1.15E-1</td>
<td>1.34E-1</td>
<td>1.53E-1</td>
<td>1.72E-1</td>
<td>1.91E-1</td>
</tr>
<tr>
<td>Pressure Distribution (bar)</td>
<td>1.65</td>
<td>2.20</td>
<td>2.75</td>
<td>3.30</td>
<td>3.85</td>
<td>4.40</td>
<td>4.95</td>
<td>5.50</td>
</tr>
<tr>
<td>Saturation Temperature (°C)</td>
<td>112.67</td>
<td>113.22</td>
<td>113.77</td>
<td>114.32</td>
<td>114.87</td>
<td>115.42</td>
<td>115.97</td>
<td>116.52</td>
</tr>
<tr>
<td>Coolant Temperature Rise (°C)</td>
<td>33.3</td>
<td>37.1</td>
<td>40.9</td>
<td>44.7</td>
<td>48.5</td>
<td>52.3</td>
<td>56.1</td>
<td>59.9</td>
</tr>
<tr>
<td>Across Core: Average Power (W/cm²)</td>
<td>18.5</td>
<td>25.0</td>
<td>31.5</td>
<td>38.0</td>
<td>44.5</td>
<td>51.0</td>
<td>57.5</td>
<td>64.0</td>
</tr>
<tr>
<td>(a) Element Shear Stress (N/m²)</td>
<td>12.3</td>
<td>17.9</td>
<td>23.5</td>
<td>29.1</td>
<td>34.7</td>
<td>40.3</td>
<td>45.9</td>
<td>51.5</td>
</tr>
<tr>
<td>(b) Element Shear Stress (N/m²)</td>
<td>8.5</td>
<td>12.5</td>
<td>16.5</td>
<td>20.5</td>
<td>24.5</td>
<td>28.5</td>
<td>32.5</td>
<td>36.5</td>
</tr>
<tr>
<td>(c) Maximum Shear Stress (N/m²)</td>
<td>6.5</td>
<td>9.5</td>
<td>12.5</td>
<td>15.5</td>
<td>18.5</td>
<td>21.5</td>
<td>24.5</td>
<td>27.5</td>
</tr>
<tr>
<td>Average Heat Flux (W/cm²)</td>
<td>26.4</td>
<td>29.6</td>
<td>32.8</td>
<td>36.0</td>
<td>39.2</td>
<td>42.4</td>
<td>45.6</td>
<td>48.8</td>
</tr>
<tr>
<td>Peak Heat Flux (W/cm²)</td>
<td>23.4</td>
<td>26.6</td>
<td>29.8</td>
<td>33.0</td>
<td>36.2</td>
<td>39.4</td>
<td>42.6</td>
<td>45.8</td>
</tr>
<tr>
<td>Onset of Nuclear Boiling</td>
<td>20.3</td>
<td>23.5</td>
<td>26.7</td>
<td>29.9</td>
<td>33.1</td>
<td>36.3</td>
<td>39.5</td>
<td>42.7</td>
</tr>
<tr>
<td>(a) Location of ONB (cm)</td>
<td>4.5</td>
<td>7.0</td>
<td>9.5</td>
<td>12.0</td>
<td>14.5</td>
<td>17.0</td>
<td>19.5</td>
<td>22.0</td>
</tr>
<tr>
<td>Maximum Cladding Surface Temperature (°C)</td>
<td>228.3</td>
<td>239.4</td>
<td>250.5</td>
<td>261.6</td>
<td>272.7</td>
<td>283.8</td>
<td>294.9</td>
<td>306.0</td>
</tr>
<tr>
<td>Limiting Heat Flux at First Onset of Nucleate Boiling</td>
<td>20.3</td>
<td>23.5</td>
<td>26.7</td>
<td>29.9</td>
<td>33.1</td>
<td>36.3</td>
<td>39.5</td>
<td>42.7</td>
</tr>
<tr>
<td>(a) Location of ONB (cm)</td>
<td>4.5</td>
<td>7.0</td>
<td>9.5</td>
<td>12.0</td>
<td>14.5</td>
<td>17.0</td>
<td>19.5</td>
<td>22.0</td>
</tr>
<tr>
<td>Burnout Heat Flux at Mid Plane</td>
<td>26.4</td>
<td>29.6</td>
<td>32.8</td>
<td>36.0</td>
<td>39.2</td>
<td>42.4</td>
<td>45.6</td>
<td>48.8</td>
</tr>
<tr>
<td>Safety Margin: Margin to ONB</td>
<td>1.46</td>
<td>1.66</td>
<td>1.86</td>
<td>2.06</td>
<td>2.26</td>
<td>2.46</td>
<td>2.66</td>
<td>2.86</td>
</tr>
<tr>
<td>(b) Margin to ONB</td>
<td>1.56</td>
<td>1.76</td>
<td>1.96</td>
<td>2.16</td>
<td>2.36</td>
<td>2.56</td>
<td>2.76</td>
<td>2.96</td>
</tr>
<tr>
<td>(c) Margin to ONB (cm)</td>
<td>4.5</td>
<td>7.0</td>
<td>9.5</td>
<td>12.0</td>
<td>14.5</td>
<td>17.0</td>
<td>19.5</td>
<td>22.0</td>
</tr>
<tr>
<td>Margin to DNB</td>
<td>1.46</td>
<td>1.66</td>
<td>1.86</td>
<td>2.06</td>
<td>2.26</td>
<td>2.46</td>
<td>2.66</td>
<td>2.86</td>
</tr>
<tr>
<td>(b) Margin to DNB</td>
<td>1.56</td>
<td>1.76</td>
<td>1.96</td>
<td>2.16</td>
<td>2.36</td>
<td>2.56</td>
<td>2.76</td>
<td>2.96</td>
</tr>
<tr>
<td>(c) Margin to DNB (cm)</td>
<td>4.5</td>
<td>7.0</td>
<td>9.5</td>
<td>12.0</td>
<td>14.5</td>
<td>17.0</td>
<td>19.5</td>
<td>22.0</td>
</tr>
<tr>
<td>* Road as 3,90×10⁴</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Figure 1(a) Critical velocity vs $\alpha$ for curved-plate assembly relative to that of flat-plate assembly. Edges hinged.

Figure 1(b) Critical velocity vs $W_p/t_p$ for curved-plate assembly relative to that of flat-plate assembly. Edges hinged.
\[ R = \frac{(T_{out} - T_{in})}{(T_{sat} - T_{in})} \]

Figure 2. Correlation for Flow Instability and Bubble Detachment

Flow Instability

- \( \Delta \) Tube, Whittle and Forgan 20,21
- \( V \) Channel, Maltbatsch 22
- \( x \) Channel, Waters 24
- \( + \) Channel, Grenoble 25,26,27,28,29

Bubble Detachment

- \( n = 12 - 35 \) Boetting 30
- \( n = 30 \) Costa 31
- \( n = 32 \) Levy 32
Figure 3: Volumetric Flow Through An Element as a function of velocity.
FIG(4) PRESSURE DROP ACROSS CHANNEL AS A FUNCTION OF VELOCITY

Coolant Velocity (m/sec)

Pressure Drop (bar)

PARR-1

PARR-2
FIG(5) COOLANT PRESSURE ALONG THE CHANNEL LENGTH
FIG(7) AXIAL PEAKING FACTOR USED IN THE ANALYSIS (COSINE SHAPE)
FIG(10) TEMPERATURE PROFILE IN PEAK AND AVERAGE CHANNELS OF PARR-3
FIG (1.4) DISTRIBUTION OF LOCAL AND ‘ONB’ HEAT FLUXES IN PARR-1
FIG. 15: DISTRIBUTION OF LOCAL AND NON-HEAT FLUXES IN PARR-2
FIG(16) DISTRIBUTION OF LOCAL AND 'ONB' HEAT FLUXES IN PARR-3

Heat Flux (W/cm²)

Distance (cm)

16

32

48

56

0 40 80 120

D

ONB D
FIG. (17) DISTRIBUTION OF LOCAL AND CRITICAL HEAT FLUXES IN PARR-1

Distance (cm)

Heat Flux (W/cm²)

Mishask Correlation

Lamberts Correlation

Extrapolated With Zero Subcooling
FIG. 19) DISTRIBUTION OF LOCAL AND CRITICAL HEAT FLUXES IN PARR-3

- Mirskyh Correlation
- Lappntsov Correlation

Extrapolated with zero subcooling
FIG(20)  COOLANT VELOCITY AS A FUNCTION OF FLOW IN PARR-3
FIG(21) PEAK TEMPERATURE OF CLAD AS A FUNCTION OF FLOW IN PARR-3
FIG (22) COOLANT TEMP. AT EXIT OF HOTTEST CHANNEL AS A FUNCTION OF FLOW RATE (m³/h)
FIG(23) HEAT FLUX AT ONB AS A FUNCTION OF FLOW IN PARR-3
FIG (24) HEAT FLUX AT OFI AS A FUNCTION OF FLOW IN PARR-3