American International Journal of Research in Science, Technology, Engineering & Mathematics

Available online at http://www.iasir.net

ISSN (Print): 2328-3491, ISSN (Online): 2328-3580, ISSN (CD-ROM): 2328-3629

AIJRSTEM is a refereed, indexed, peer-reviewed, multidisciplinary and open access journal published by International Association of Scientific Innovation and Research (IASIR), USA
(An Association Unifying the Sciences, Engineering, and Applied Research)

Initial Thermal Hydraulic Design of a 1000MWe Water Reactor with an Improved Thermal Efficiency

C. E. Okon[*1], D. E. Oku[*2]
[*1]School of Physics & Astronomy, University of Manchester,
Manchester, UNITED KINGDOM.
[*2]Department of Physics, University of Calabar-NIGERIA,

Abstract: *This design considers the thermal hydraulic processes involved in the transfer of power from the core to the secondary system of a PWR nuclear plant. In practice a net plant efficiency of around 32% - 36% would be typical for a water-cooled reactor. I choose to operate with a much higher temperature (330°C) to see its effect on the reactor performance. This result to a higher thermal efficiency (about 45%) and core power (about 2218MWth). Safety margins and conditions have been considered and all thermodynamical parameters and heat transfer processes involved has been calculated accurately. Analytical approach of the general heat conduction equation in cylindrical coordinate is employed to solve the temperature distribution and heat flux and the calculated temperature distribution follows the expected decay pattern. The axial and radial profile is shown.*

I. Introduction

The quest for nuclear technology lies on its importance in the generation of electricity. Although many reactors were built primarily to produce plutonium for nuclear weapons and for scientific research, they continue to be developed as power generators. They became economically viable in the 1960's and today fission reactors are a major source of energy in many countries[6].

Most modern reactor designs capitalized on the good qualities of light water (H_2O) as a moderator. Its excellent slowing-down properties and relatively low diffusion and slowing-down lengths (compared with carbon) enables the size of the core to be reduced considerably to one which is, typically, about 3m in height and diameter. However, enriched fuel is used to offset the reduction in thermal utilization factor *f*, due to greater tendency of water to absorb neutrons. In order to maintain its effectiveness a coolant, water must be prevented from boiling as far as reasonably possible and it is subjected to a very high pressure (about 150-160bar) to raise its boiling point. It is circulated through external heat exchangers, where steam for turbines is generated. Such a reactor using water as both moderator and coolant became my choice of reactor design. It uses enriched uranium dioxide fuel and zircaloy cladding. This reactor is cheaper to build and operate. Material specification for my chosen design can be found in Appendix.

II. Aim & Objective

The aim of this design is aim at improving upon the thermal efficiency of a Pressurized Water Reactor (PWR). The reason for my choice is because PWR is the only commercial nuclear power plant currently in operation in Africa, located in South Africa. Also the first commercial nuclear power plant expected to be built in my country "Nigeria" is likely to be PWR. This reactor is easy to build and manage, less expensive and there is much availability of the cooling source in my country. Besides all this attractive features, there is need to improve upon its thermal efficiency, because high thermal efficiency implies a lower cooling-water requirement.

III. Methodology

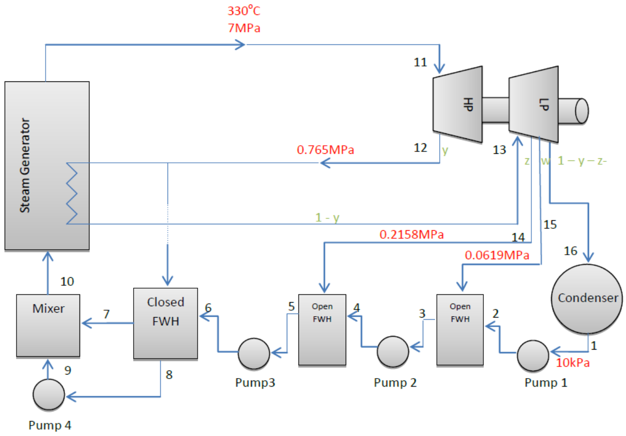
A. Power Conversion

This design considers a steam power plant with two turbine operating on an ideal reheat-regenerative Rankine cycle with two open feedwater heater, one close feedwater heater and one reheater. It is assumed that the heat enters the turbine at 7MPa and 330°C and it is condensed in the condenser at 0.01MPa. The figure below shows the layout of the power conversion system.

In order to calculate the thermal efficiency and the core power, certain assumptions were made

- 300°C and 330°C inlet and outlet temperatures respectively. By reviewing various articles and publications majority of PWR power plant operates at about 280°C and 330°C. The reason for my choice is to examine the effect on the reactor efficiency and performance.

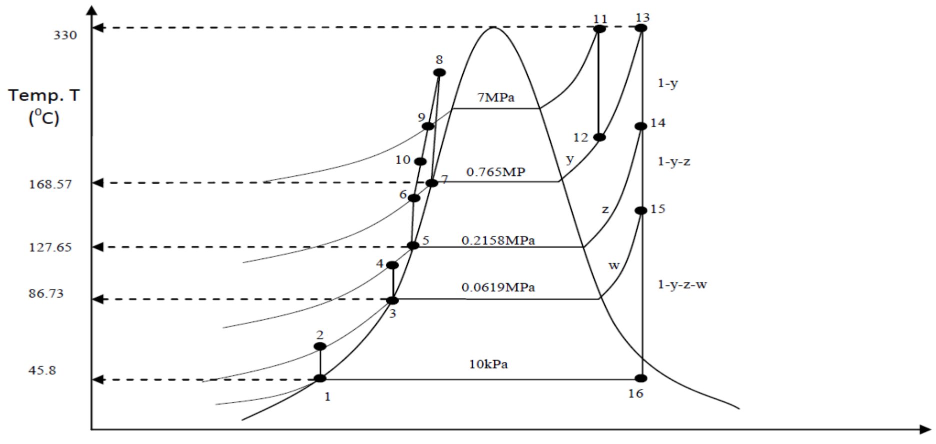
- Condenser pressure is chosen to be 0.01MPa because any value lower than this will result to air leakage into the condenser and moisture content of steam in the turbine [2].
- The steam exiting the low turbine is assumed to have a quality of about 90% to avoid excessive wear of the turbine blades [2]



Also, the following thermodynamic analysis was carried out in order to ascertain the thermal efficiency and the thermal power output. The figure below shows the ideal circle for the layout above.

- The enthalpy at various steps is calculated using the steam table.
- The pressure at both turbines is calculated. The closed FWH and reheat loops are splitted after the high pressure (HP) turbine rather than bleeding off steam from the high pressure turbine. This enables a simpler and less costly turbine.
- The two Open Feedwater Heater pressure is gotten by finding the bleeding saturated temperature using this relation T5 = T1 + 2/3(T7 – T1) and T3 = T1 + 1/3(T7-T1) and relating this temperature to pressure via the steam tables.
- The operating pressure of the HP turbine and the Close Feedwater Heater is found by assuming that entropy S13 ≈ S16 and T13 ≈ T11
- Between point 6 and 10, the diagram account for temperature change due to mixing. At this point the energy balance for the mixing is introduced in order to get the enthalpy h9. Hence T9<T8

Therefore, the efficiency from the thermodynamic analysis is found to be 45%. This percentage value has contributed a significant improvement on the current PWR operating all over the world offering between 32% - 36% thermal efficiency.



B. *Nominal and Hot Channel Description*

During Core Thermal Design, there are some safety factors that need to be considered in order to guide against the temperature of the fuel exceeding its theoretical set limit. Those safety factors are called the hot channel factors. In practice, however, local deviations in behavior from strict adherence to that predicted by theoretical correlations take place. These local deviations (aside from those due to the fuel acting as a sink and the moderator as the source of thermal neutrons are caused by factors, such as the presence of partially inserted control rods depressing thermal neutrons in their vicinity with consequence large peaking elsewhere, non-homogeneity in the moderator (e.g., when in solid form or when boiling takes place) or fuel and the presence of other structural homogeneities [7].

Also, the presence of a good, infinite reflector causes peaking of the thermal neutron flux at the ends of the reactor core and consequent high heat generation and high fuel temperatures there. The safety factor for flux distribution usually takes into account the above deviation as well as the deviation of the overall flux distribution from the core average.

C. Power Peaking Factor (PPF)

In this design, the most realistic PPF was found by looking at historical data cited from 1999 World Nuclear Industry Handbook. The average data of 2.25 for six different reactors is used [1].

D. Material Specification of the Design

The material specifications for fuel, cladding and coolant is the same as those use for PWR reactors currently in operation although some values varies, they still operate within the safe margin and initial conditions. The system pressure is 155bar, Core Inlet Temperature is 300^{o}C, Core Outlet Temperature is 330^{o}C and the bulk coolant temperature is 315^{o}C. Other parameters for this design can be found in the Appendix.

D.1 Fuel Centerline Temperature

The theoretical temperature limit for this design is 2000^{o}C. The design specification is carried out in such a way that it cannot exceed this temperature margin. All initial and hot-spot factors are being considered in this design so that this safe margin is not exceeded. If the centerline temperature exceeds this set limit, the fuel will melt as a result of boiling.

D.2 Rod Length

The maximum rod length that could be use for this design is 6.6337m (if the length is greater than this set value, the surface heat flux will increase beyond the critical point resulting to boiling crisis). The minimum rod length is 1.70m (any value less than this will result to greater heat generation in the core above the theoretical set value of 2000^{o}C) which could also cause the failure of the cladding material.

My chosen rod length for this design is 3.5m. The main reason why I choose 3.5m wasn't because most of the conventional reactors in operation were using this value, rather it is because I got an exact (not approximate) value for the total fuel length (189,189m). Using the rod with length 4m could also give an approximate value (216,216) but this could be more expensive to build considering the total numbers of rods involve. So low cost must also be put into consideration for any design proposed.

IV. Result & Discussions

A. Mass Flow Rate

Now using the exit and inlet temperatures, 300^{o}C and 330^{o}C respectively, the calculated core power of Q=2200MWth and the specific heat c_P*=6296.8 [J/kgK]* we find:

$$\dot{m} = \frac{Q_{Th}}{C_p \Delta T} \cong 11742.47kg/s \cong 3261.8tons/hr$$

If the mass rate of flow of the coolant and the coefficient of heat transfer are the same in all fuel channels, irrespective of their position in the core, the maximum fuel temperature of the fuel elements, $T_{m,}$ vary accordingly. The fuel elements near the center of the core may have temperatures in excess of the safe limit, while those near the core boundaries operate much below this limiting temperature. For this design, my calculated and desired value for the coolant channel flow rate is 0.21720kg/s.

B. Temperature Profile

An analytical approach of the general heat conduction differential equation in cylindrical coordinate is employed to solve the temperature distribution and heat flux for the fuel element. In steady state operation of a reactor, the temperature distribution is determined by thermal balance between heat generated in the core and heat transferred to the coolant, since the neutron flux level is limitless. Provided that there is an adequate heat removal system and initial conditions are acknowledge, the temperature in the core will not exceed the specific safety limits, and damage to fuel pins and other reactor materials would be prevented. Calculated values for the temperature profiles for this design can be found in Table 1C of the appendix.

Table 1A shows the DNBR profile for each node. While Table 1B shows the surface temperature for each node by taking into account the Power Peaking Factor (PPF) at each axial step. It can be seen from the table that the maximum clad surface temperature is ≈ 426^{o}C using this equation

$$T_c = T_{in} + \frac{q_c''' A H_e}{\dot{m} C_p \pi}\left(sin\frac{\pi z}{H_e} + sin\frac{\pi H}{2H_e}\right) + \frac{q_c''' A}{2\pi r h} cos\frac{\pi z}{H_e}$$

Plots showing the Axial temperature profile, Radial temperature profile, DNBR profile and Power Peaking Factor can be seen in Fig. 1A"1", 1A"2", Fig. 1B and Fig. 1C in the Appendix.

C. Thermal Limits/Limiting Powers

C.1 Variation Of Mass Flow Rate

➢ Effect on CHF and Fuel Rod Surface Temperature

In a nuclear reactor system the critical heat flux is the heat flux at which a boiling crisis occurs that causes an abrupt rise of the fuel rod surface temperature and, subsequently, a failure of the cladding material [8].

It is well known that CHF is dependent on the geometrical conditions as well as the thermal-hydraulic conditions.
Examples of operating conditions for water cooled reactors can be found from the table below.

Table A: Examples of Operating Conditions of Different Reactors

	PWR	BWR	PHWR	WER
Pressure P (MPa)	≈15.7	≈7.2	≈10.5	≈15.7
Avg. Mass Flux G (Mg/m^2s)	≈4.0	≈3.0	≈5.0	≈4.0
Avg. Outlet Steam Quality X, (-)	≈-0.15	≈0.15	≈0.0	≈-0.15
Fuel Rod Diameter d (mm)	≈9.5	≈12.3	≈13.1	≈0.1
Pitch of Diameter Ratio, (-)	≈1.3	≈1.3	≈1.15	≈1.4

Table A, gives just one example from each group. It should be kept in mind that for the same reactor group, e.g. PWR, the operating conditions may differ from one design to another and vary over a wide range except for the operating pressure.
The CHF was calculated using the W-3 correlation developed by Tong [8]. According to my results in Table 3A and 3B (variation of mass flow rate) in the Appendix, it is seen that the CHF and T_{max} is affected by channel flow rate. As you decrease the flow rate, there is a corresponding decrease in the HTC and G, which in turns causes an increase in CHF and T_{max}. As the mass flow rate is decreased further below 0.0028tons/hr, the CHF remains massively high. This effect causes a rise in the fuel centerline temperature above the theoretical limiting value of 2000^{o}C. At this point, boiling crisis occur which could result to a failure of the cladding material. The graph of **Mdot Vs CHF, Mdot Vs T_{max}, CHF Vs T_{max}** are plotted to explain this phenomenon, those graphs can be found in the Fig.3B 1, 2&3 of the Appendix.

➢ Effect On Mass Flux G (Kg/M^2s)

For different designs of PWR's the operating pressures only ranges from 15.0MPa to 16.0MPa. The mass flux changes more significantly. In a conventional PWR the mass flux is about 4Mg/m^2s, whereas in a highly conversional PWR a much higher mass flux (6Mg/m^2s) is required (Oldekop et al. 1982).
By increasing the mass flow rate above 0.34tons/hr, it is observed that the mass flux exceed its maximum operating value of 4Mg/m^2s(Limit) as stated above for a conventional PWR. Further increase above 0.57786tons/hr will result to an increasing value of mass flux beyond the limiting value (6Mg/m^2s) for the validity of the W-3 Correlation equation. The graph of **Mdot Vs HTC, Mdot Vs G,** are plotted to explain this phenomenon. See Fig.3A"2" and Fig. 3A"5" of Appendix.

➢ Effect on DNBR

Further increase above 12.593tons/hr could results to the melting of the core because the MDNBR limit of 1.3 would have been exceeded. The graph of **Mdot Vs MDNBR** is plotted to explain this phenomenon. See Fig.3A"6"of Appendix.
The design of water cooled reactor requires a sufficient safety margin with regard to the critical heat flux, mass flux and DNBR in order to avoid boiling crisis and high void fraction in sub-channels.

D. Variation of the Inlet Temperature

The enthalpy of the coolant changes directly with the inlet temperature. To investigate further how inlet temperature affects the reactor design, it was necessary to vary the inlet temperature from 290oC to 299oC. In carrying out this analysis, there was no significant effect on factors which would have been considered to depend on the inlet conditions. Those factors are; the fuel centerline temperature, the mass flow rate and the heat transfer coefficient. Rather it was observed that the quality has an influence on the variation. As the inlet temperature is decreased below 300^{o}C the quality also increases significantly. Different plots to show its effect are shown in Fig. 4a, 4b, 4c, 4d, and 4e of the Appendix.

E. Variation of Rod Length

In order to give justification to my desire rod length of **3.5m,** I did some calculations of different parameters for different rod lengths to see the most realistic rod length value. From my calculated analysis, the following conclusions were drawn;

❖ *LONG ROD:* increase in rod length result to increase in quality with a corresponding increase in the total number of rods and the heat generated in the core is minimized. Departure from nuclear boiling (DNB) occurs when the rod is increased above the maximum set value.

❖ *SHORT ROD:* a shorter rod length result to a lesser number of rods. But at very shorter value below 2m, the centerline temperature increases above the theoretical set value of 2000^{o}C.

Therefore, the most realistic rod length for this design is 3.5m with quality of 10%.
Different plots for the above explanation can be seen in Fig. 2A, 2B, 2C, and 2D. Table 2 of the Appendix shows the calculated parameters for different length values.

F. Coolant Systems

F.1 Circuit Pressure Drop (Assume a homogeneous model)

Pumping-power requirements are determine by the pressure drop in the cooling system and the rate of flow of coolant. In homogenous model, the two phases are assumed to flow as a single phase possessing mean flow properties and a suitable single-phase friction factor is developed to represent the two phase flow [4]. Using the data for Pressure Drop Calculation in Appendix 1, it is observed that the inlet enthalpy is less than the saturation enthalpy and thus at the inlet we are dealing with a single phase flow, therefore, the exit enthalpy is gotten from the thermodynamics equation below. Assume power of 64000W.

$$h_{out} = h_{in} + \frac{Q}{\dot{m}}$$

$$h_{out} = 1.63 * 10^6 J/kg$$

From the above result, it is seen that the exit enthalpy is greater than the saturation enthalpy and thus somewhere in the pipe the flow changes from single phase to two phase. We know that the single flow part (or the region where $h<h_f$) is some fraction of the total enthalpy, multiplying this fraction by the total length of the pipe will tell us the length of the pipe in single phase flow L_{sp}.

$$L_{SP} = \left(\frac{h_f - h_{in}}{h_{out} - h_{in}}\right) * L$$

$$L_{SP} = 3.47m$$

The pressure drop required to overcome friction for the turbulent flow of coolant in the channel between the fuel rods is given by [4]. ;

$$\Delta P_{SP} = f_{SP} * \frac{L_{SP}}{D} * \frac{G^2}{2\rho_{SP}} \Psi$$

Where, ΔP_{SP} is the Single Phase Pressure drop, Ψ (=1.3) is the correction for fluid flow in rod bundles [1] and the friction factor for single phase flow is given as; $f_{SP} = 0.184 Re^{-0.2}$

$$\therefore \Delta P_{SP} = 1.42E4\ Pa \quad \approx 0.0142\ MPa \qquad \ldots\ldots\ldots\ldots SINGLE\ PHASE\ PRESSURE\ DROP$$

For two phase calculation, the pressure drop equation is expressed as;

$$\Delta P_{TP} = \varphi_{TP}^2 * \Delta P_{SP} * \left(\frac{L - L_{SP}}{L_{SP}}\right)$$

where;

$$\varphi_{TP}^2 = \frac{1}{V_f}\left(\frac{V_{exit} + V_f}{2}\right) = 1.01$$

is the friction multiplier factor and the exit specific volume is given as $v_{exit} = v_f + x_{exit} * v_g$

$$\therefore \Delta P_{TP} = 1.13E2\ Pa \quad \approx 0.113kPa \qquad \ldots\ldots\ldots\ldots TWO\ PHASE\ PRESSURE\ DROP$$

Hence, from my analysis the Total Pressure Drop "ΔP" across the core is given as;

$$\Delta P = \Delta P_{SP} + \Delta P_{TP} = 1.44E4\ Pa \quad \approx 0.0144MPa$$

F.2 Pumping Power

Many reactor parameters depend upon temperature. Reactor temperature, however, is usually a function of the operating power of the reactor, and changes in power level may lead to changes in the criticality of the system [1].

The pumping work required by the circulating coolant to overcome pressure losses through a complete loop (reactor, piping, heat exchangers, etc) with allowance for pump efficiency is gotten by

$$Q_p = \frac{\Delta P * \dot{m}}{\eta \rho_{avg}} = 3.57E + 5W$$

Where η is the efficiency (assumed 80%), $\dot{m}\ \&\ \rho_{avg} = 590kg/m^3$ is the core mass-flow rate and average flow density of the coolant respectively.

With no allowances for efficiency, the pumping power required to circulate the coolant through the core is *2.86E+5W.*

G. Heat Removal (Steam Generator)

A typical 1000MWe plant steam generator, with an overall height of about 20.7m is considered in this design. The heated primary system coolant from the reactor vessel passes through the inverted U-shaped tubes and saturated steam at 7MPa. The following parameters is used to calculate the overall heat transfer coefficient, mean temperature difference and number of tubes used.

Hot Leg Temperature T_{hot} = 330°C

Cold Leg Temperature T_{cold} = 300°C

Pinch Point DT_{sat} = 20°C

Sat. Temp from Sec. T_{sat}. @ 70bar = 285.8°C

Kt = 30W/m.K
Tube inner diameter D1 = 0.019m
Tube outer diameter D2 = 0.0222m
Linear Heat Flux q'' = 2.64E4W/m
Core Power Q = 2.22E9W
Therefore, the overall heat transfer coefficient is gotten from the heat transfer equation;

$$Q = UAdT$$

$$U = \left(\frac{1}{h} + \frac{a}{k_t}\right)^{-1} = 4489W/m\text{^}2.K$$

Where

$$h_s = \frac{q''}{\Delta T_{sat}}$$

$$A = \frac{Q}{UdT} = 16936m^2$$

Number of tubes used according to AP1000 design equation is given as

$$N = \frac{A}{L\pi D_2} = 5305$$

Steam generators in Nuclear Plants are made up of between 3000-16000 tubes.

H. Safety Strategy

The fundamental process which underlies the operation of a reactor is, of course, the fission chain reaction, and the central problem of the reactor designer is to provide a system in which a self-sustained chain reaction can occur with complete safety. At the same time, the reactor must be capable of fulfilling the function for which it is designed, i.e., the production of power, isotopes, etc. A reactor will become supercritical or subcritical if its properties are changed in such a way that its multiplication factor becomes different from unity. For instance, if a coolant channel becomes clogged, some of the fuel rods, denied proper cooling, may increase in temperature to a point where they melt. In this design, the following safety strategies where considered;

- In super critical region, the heat transfer property of water decline (reduces). Therefore, operating in or near this region should be avoided in normal and accident scenarios.
- Since my operating pressure is 155bar and the critical saturation pressure 221bar, there is a safety margin of 66bar. Similarly, my operating temperature is 330^{o}C with the corresponding saturation temperature of 345^{o}C and the critical saturation temperature is 374.15^{o}C, there is safety margin of 29.19oC.
- The core average exit coolant temperature should not be lower than that desired for good thermal efficiency of the plant. That is, the exit temperature of the coolant should be equal in all channels. Also, during the adjustment of the coolant mass flow rate to a desired value, the flow rate should be made proportional to the heat generated in the adjacent fuel element (the inlet temperature should be assumed uniform in this process)

V. Conclusion

Nuclear reactors are potentially dangerous devices, and for this reason they must be designed with care. The accidental release of the accumulated fission products from even a small power reactor can lead to a disaster of major proportions. Even if the fission products are prevented from escaping to the surrounding community by a suitable containment vessel, repairs of the reactor can be a lengthy and expensive undertaking.

References

[1] Nuclear Engineering International, 1999 World Nuclear Industry Handbook, Nick Fielder, 1999.
[2] Y. A. Cengel and M. A. Boles, Thermodynamics an Engineering Approach, McGraw Hill, 1994.
[3] M. El-Wakil, Nuclear Power Engineering, McGraw-Hill, 1962.
[4] Glassione and Sesonske, Nuclear Reactor Engineering, Van Nostrand Reinhold Company, 1981.
[5] J. Lamarsh and A. Baratta, Introduction to Nuclear Engineering, Prentice Hall Inc., 2001.
[6] J. Lilley, Nuclear Physics Principles and Applications, John Wiley and Sons, Ltd, 2001.
[7] N. Todreas and M. Kazimi, Nuclear Systems 1 Thermal Hydraulic Fundamentals, Thermisphere Publishing Corporation, 1990.
[8] R. H. S. Winterton, Thermal Design of Nuclear Reactors, Wheaton & Co. Ltd., Exeter, 1981.

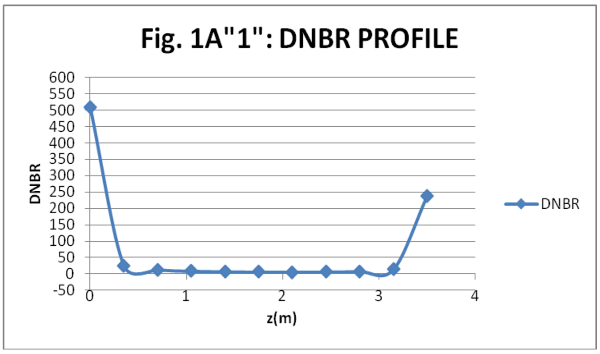


Fig. 1A"1": DNBR PROFILE
DNBR
z(m)
DNBR

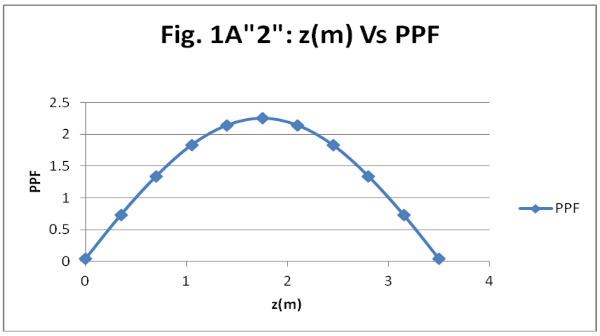


Fig. 1A"2": z(m) Vs PPF
PPF
z(m)
PPF

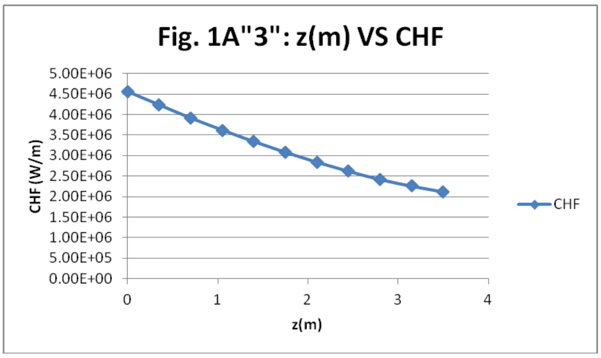


Fig. 1A"3": z(m) VS CHF
CHF (W/m)
z(m)
CHF

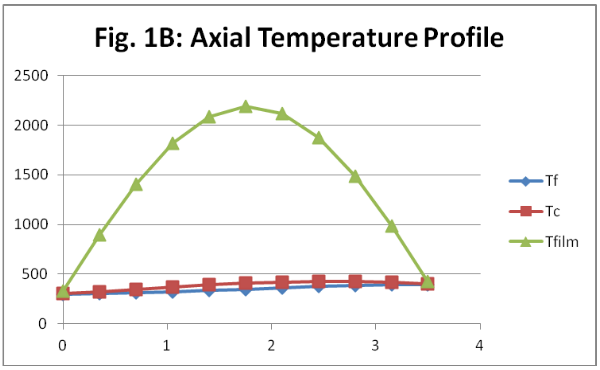


Fig. 1B: Axial Temperature Profile
Tf
Tc
Tfilm

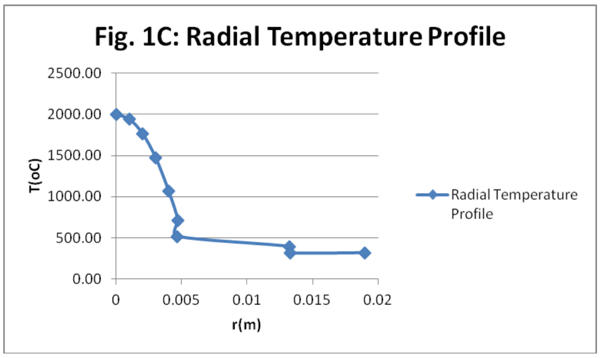


Fig. 1C: Radial Temperature Profile
T(oC)
r(m)
Radial Temperature Profile

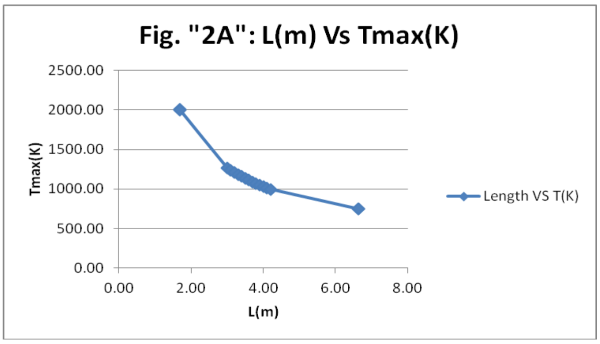


Fig. "2A": L(m) Vs Tmax(K)
Tmax(K)
L(m)
Length VS T(K)

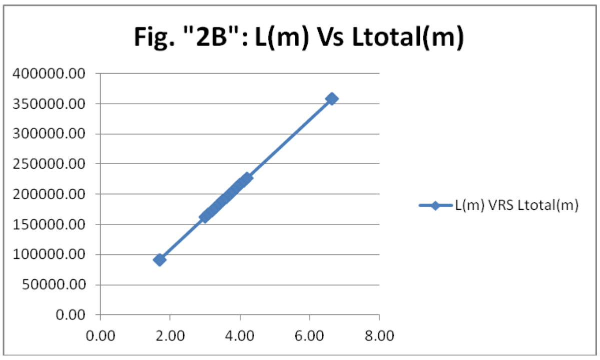


Fig. "2B": L(m) Vs Ltotal(m)
L(m) VRS Ltotal(m)

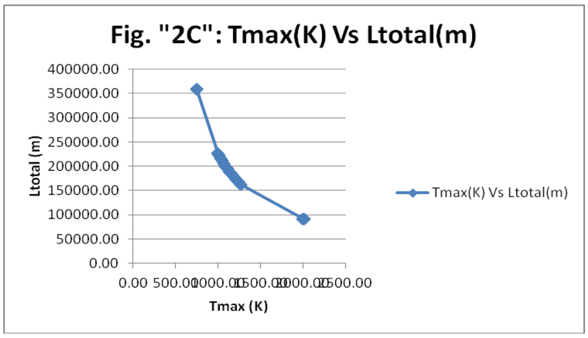


Fig. "2C": Tmax(K) Vs Ltotal(m)
Ltotal (m)
Tmax (K)
Tmax(K) Vs Ltotal(m)

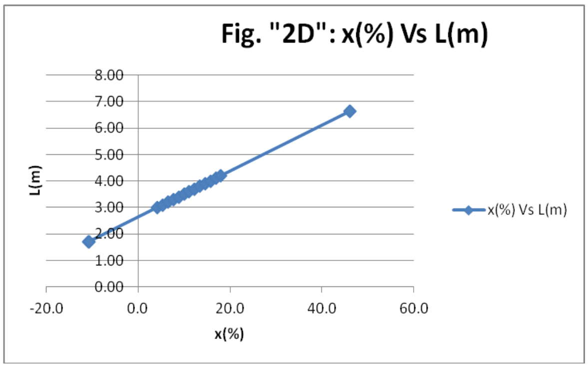


Fig. "2D": x(%) Vs L(m)
L(m)
x(%)
x(%) Vs L(m)

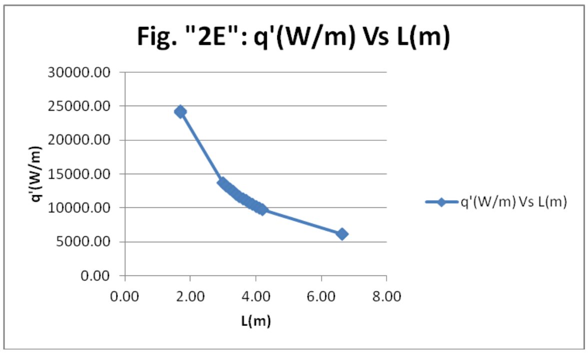


Fig. "2E": q'(W/m) Vs L(m)
q'(W/m)
L(m)
q'(W/m) Vs L(m)

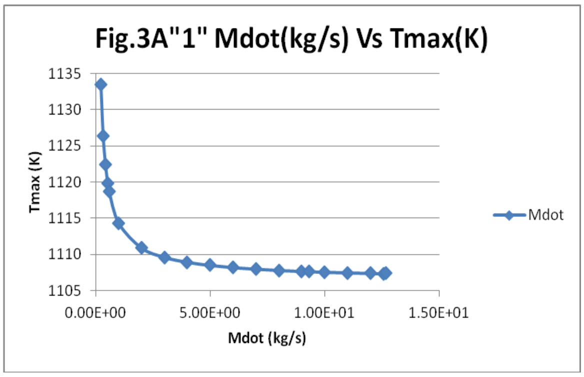


Fig.3A"1" Mdot(kg/s) Vs Tmax(K)
Tmax (K)
Mdot (kg/s)
Mdot

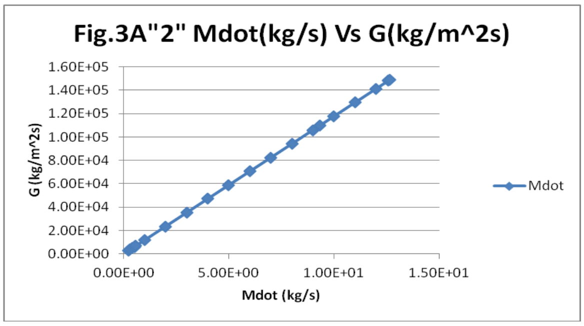


Fig.3A"2" Mdot(kg/s) Vs G(kg/m^2s)
G (kg/m^2s)
Mdot (kg/s)
Mdot

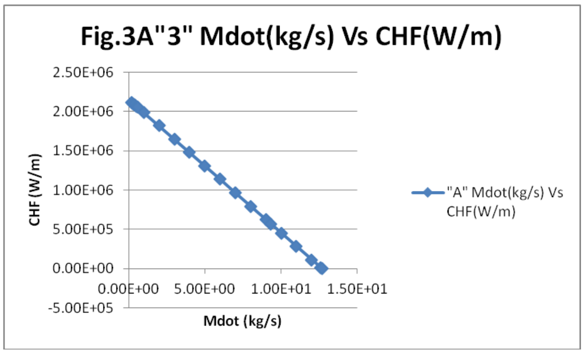


Fig.3A"3" Mdot(kg/s) Vs CHF(W/m)
CHF (W/m)
Mdot (kg/s)
"A" Mdot(kg/s) Vs CHF(W/m)

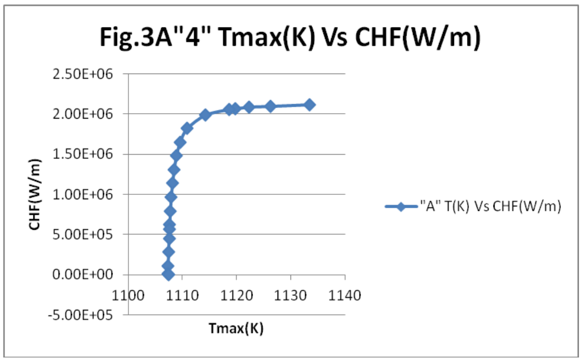


Fig.3A"4" Tmax(K) Vs CHF(W/m)
CHF(W/m)
Tmax(K)
"A" T(K) Vs CHF(W/m)

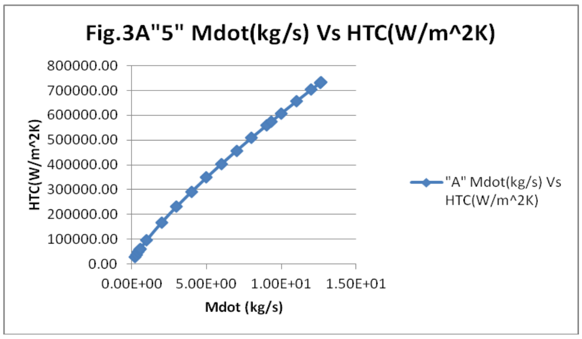


Fig.3A"5" Mdot(kg/s) Vs HTC(W/m^2K)
HTC(W/m^2K)
Mdot (kg/s)
"A" Mdot(kg/s) Vs HTC(W/m^2K)

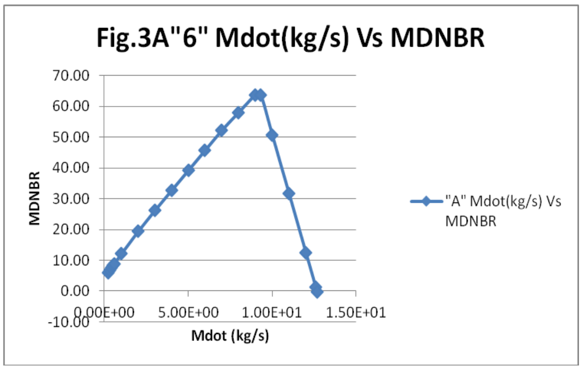


Fig.3A"6" Mdot(kg/s) Vs MDNBR
MDNBR
Mdot (kg/s)
"A" Mdot(kg/s) Vs MDNBR

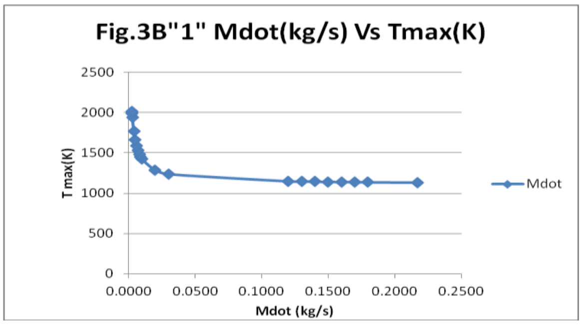


Fig.3B"1" Mdot(kg/s) Vs Tmax(K)
Tmax(K)
Mdot (kg/s)
Mdot

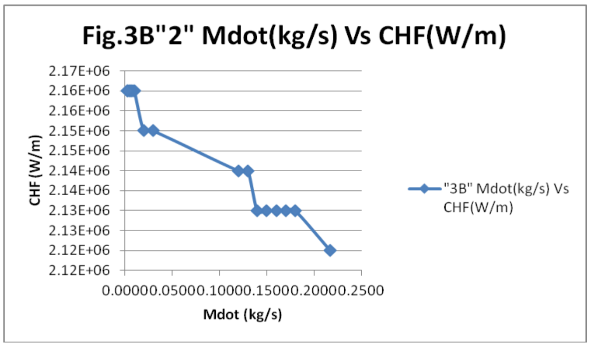


Fig.3B"2" Mdot(kg/s) Vs CHF(W/m)
CHF (W/m)
Mdot (kg/s)
"3B" Mdot(kg/s) Vs CHF(W/m)

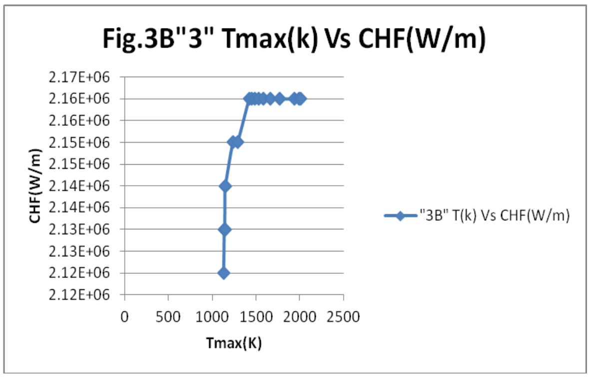
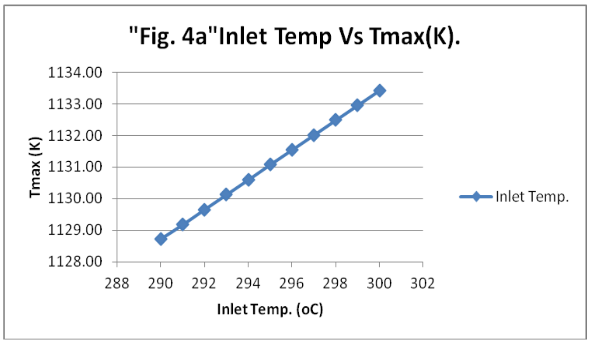
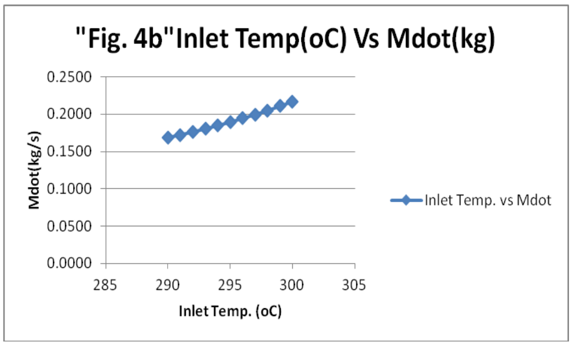
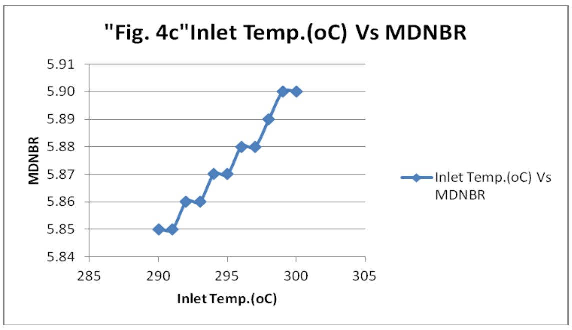
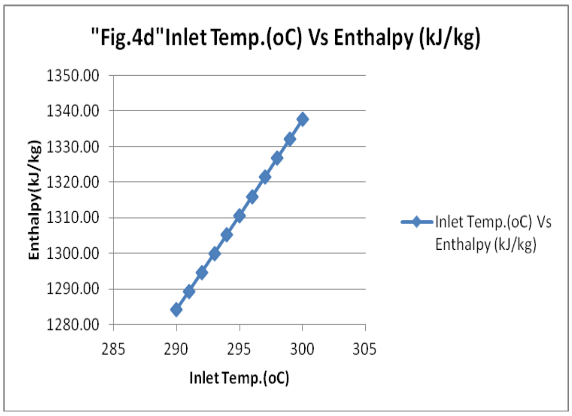


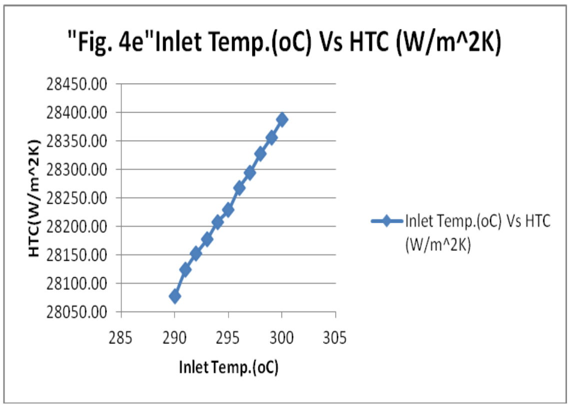
Fig.3B"3" Tmax(k) Vs CHF(W/m)
CHF(W/m)
Tmax(K)
"3B" T(k) Vs CHF(W/m)


"Fig. 4a"Inlet Temp Vs Tmax(K).
Tmax (K)
Inlet Temp. (oC)
Inlet Temp.


"Fig. 4b"Inlet Temp(oC) Vs Mdot(kg)
Mdot(kg/s)
0.2500
0.2000
0.1500
0.1000
0.0500
0.0000
285
290
295
300
305
Inlet Temp. (oC)
Inlet Temp. vs Mdot


"Fig. 4c"Inlet Temp.(oC) Vs MDNBR
MDNBR
5.91
5.90
5.89
5.88
5.87
5.86
5.85
5.84
285
290
295
300
305
Inlet Temp.(oC)
Inlet Temp.(oC) Vs MDNBR


"Fig.4d"Inlet Temp.(oC) Vs Enthalpy (kJ/kg)
Enthalpy(kJ/kg)
1350.00
1340.00
1330.00
1320.00
1310.00
1300.00
1290.00
1280.00
285
290
295
300
305
Inlet Temp.(oC)
Inlet Temp.(oC) Vs Enthalpy (kJ/kg)


"Fig. 4e"Inlet Temp.(oC) Vs HTC (W/m^2K)
HTC(W/m^2K)
28450.00
28400.00
28350.00
28300.00
28250.00
28200.00
28150.00
28100.00
28050.00
285
290
295
300
305
Inlet Temp.(oC)
Inlet Temp.(oC) Vs HTC (W/m^2K)