UNIVERSITI PUTRA MALAYSIA

CFD MODELING OF PERFORMANCE ENHANCEMENT FOR NUCLEAR RESEARCH REACTOR COOLING SYSTEM

ROS LI BIN DARMA WAN

FK 2015 159
CFD MODELING OF PERFORMANCE ENHANCEMENT FOR NUCLEAR RESEARCH REACTOR COOLING SYSTEM

By

ROSLI BIN DARMAWAN

Thesis Submitted to the School of Graduate Studies, Universiti Putra Malaysia, in Fulfilment of the Requirements for the Degree of Doctor of Philosophy

March 2015
COPYRIGHT

All material contained within the thesis, including without limitation text, logos, icons, photographs and all other artwork, is copyright material of Universiti Putra Malaysia unless otherwise stated. Use may be made of any material contained within the thesis for non-commercial purposes from the copyright holder. Commercial use of material may only be made with the express, prior, written permission of Universiti Putra Malaysia.

Copyright © Universiti Putra Malaysia
Abstract of thesis presented to the Senate of Universiti Putra Malaysia in fulfilment of the requirement for the degree of Doctor of Philosophy

CFD MODELING OF PERFORMANCE ENHANCEMENT FOR NUCLEAR RESEARCH REACTOR COOLING SYSTEM

By

ROS LI BIN DAR MAWA N

March 2015

Chair: Nor Mariah Adam, PhD, PE.

Faculty: Engineering

The Reactor TRIGA PUSPATI (RTP), a pool-type research reactor started operation in 1982 with maximum steady state power of 1 MW is classified as a low flux research reactor with very limited applications. Many of such reactors have undertaken upgrading exercises to increase the capacity which involved the installation of higher capacity heat removal system. This approach requires higher cost as well as major modification of the reactor pool. Other alternatives which require less modification have been investigated.

The RTP cooling pool operates on normal water at 32°C bulk temperature in laminar flow steady state condition, surrounded by adiabatic wall, floor and open top exposes to room temperature. The heat source comes from the reactor core at 550W/m² heat flux and cooled down by natural convection. These conditions were modeled using CFD computational code FLUENT V6.3. The conservation equations for fluid flow, momentum, continuity and energy equation were solved numerically to predict the hydrodynamic and thermal behaviors of the model. The modeling of natural convection employed Boussinesq approximation model for the buoyancy term to achieve faster convergence. A prototype of 1/10th scale model was developed to verify the simulation results. A dimensional analysis using Buckingham Π theorem was conducted to develop a dimensionless correlation to characterize the system and to ensure the geometric and kinematic similarity of the model.

The simulation results showed similar temperature and velocity profiles with similar cases available in the literature. The measured data from the scaled model and the CFD simulation showed good agreement. The results reveal the flow regimes, temperature profile and the mechanism of the natural convection
formation inside the reactor cooling pool. Four (4) cooling optimization techniques were simulated at normal operating heat flux (550W/m$^2$) and at higher heat flux of 1100W/m$^2$ and 1650W/m$^2$ to assess its cooling performance. The results show that the installation of in-pool pumps, conduction rods and low temperature cooling inside the existing pool system may be able to enhance the cooling capacity up to 1.5 to 3 times the normal heat flux; whereas the installation of in-pool heat exchanger only performed slightly better than normal operation.
PEMODELAN CFD UNTUK PENINGKATAN PRESTASI SISTEM PENYEJUK REAKTOR PENYELIDIKAN NUKLEAR

Oleh

ROS LI BIN DARMAWAN

March 2015

Pengerusi: Nor Mariah Adam, PhD, PE.

Fakulti: Kejuruteraan


Keputusan simulasi telah menunjukkan profil suhu dan halaju yang sama dengan kes-kes hampir serupa di dalam literatur. Data dari ujikaji model skala dan dari simulasi CFD telah menunjukkan persamaan yang baik. Keputusan
telah menunjukkan regim pengaliran, profil suhu dan mekanisma pembentukan olakan semulajadi di dalam kolam penyejukan reaktor. Empat (4) kaedah bantuan penyejukan telah disimulasi pada fluks haba operasi biasa (550W/m\(^2\)) dan pada fluks haba lebih tinggi, 1100W/m\(^2\) dan 1650W/m\(^2\) untuk menilai keupayaan penyejukannya. Keputusan simulasi menunjukkan bahawa dengan kaedah pemasangan pam dalaman, rod konduksi dan penyejukan suhu rendah di dalam sistem kolam penyejuk sedia ada, telah meningkatkan keupayaan penyejukan 1.5 hingga 3 kali lebih baik dari fluks haba biasa; manakala pemasangan penukar haba dalaman hanya berprestasi lebih baik sedikit dari operasi biasa.
ACKNOWLEDGEMENTS

I would like to express my gratitude to my supervisor, Associate Professor Ir. Dr. Nor Mariah Adam for her valuable guidance throughout the duration of this study. I would like to acknowledge the member of Supervisory Committee, Associate Professor Dr. Nuraini Abdul Aziz, and Associate Professor Dr. M Khairol Anuar M Ariffin, for their help, advice and guidance in due course of preparing this thesis.

I would like to thank Mr. Mohd Arif Hamzah, Mr. Fadil Ismail and Mr. Muhammad Nor, the technicians of MINT mechanical laboratory, for giving me the opportunity to conduct the experiment in their laboratory, their invaluable help and sharing of experience throughout the study. To Engineering Faculty of UPM, and many others who have made contributions, please be known that you have my deepest appreciation.

Finally, I would like to thank my beloved family, my wife Jamaliah, my son Muhammad Azam and daughters Nur Wani, Nur Afrina and Nur Adlina Najwa, for their love, sacrifice, support, patience and encouragement throughout the duration of the study.
I certify that a Thesis Examination Committee has met on 13th March 2015 to conduct the final examination of Rosli bin Darmawan on his thesis entitled “CFD Modeling of Performance Enhancement for Nuclear Research Reactor Cooling System” in accordance with the Universities and University Colleges Act 1971 and the Constitution of the Universiti Putra Malaysia [P.U.(A) 106] 15 March 1998. The Committee recommends that the student be awarded the Doctor of philosophy.

Members of the Thesis Examination Committee were as follows:

Zulkiflle Leman, PhD
Associate Professor
Faculty of Engineering
Universiti Putra Malaysia
(Chairman)

Tang Sai Hong, PhD
Associate Professor
Faculty of Engineering
Universiti Putra Malaysia
/Internal Examiner

Shamsuddin Sulaiman, PhD
Professor
Faculty of Engineering
Universiti Putra Malaysia
/Internal Examiner

Shaker Abdel Meguid, PhD
Professor
University of Toronto
Canada
(External Examiner)

ZULKARNAIAN ZAINAL, PhD
Professor and Deputy Dean
School of Graduate Studies
Universiti Putra Malaysia

Date: 13 May 2015
This thesis was submitted to the Senate of Universiti Putra Malaysia and has been accepted as fulfilment of the requirement for the degree of Doctor of Philosophy. The members of the Supervisory Committee were as follows:

**Nor Mariah Adam, PhD**
Associate Professor  Ir
Faculty of Engineering
Universiti Putra Malaysia
(Chairman)

**Mohd Khairol Anuar Mohd Ariffin, PhD**
Associate Professor  Ir
Faculty of Engineering
Universiti Putra Malaysia
(Member)

**Nuraini Abdul Aziz, PhD**
Associate Professor
Faculty of Engineering
Universiti Putra Malaysia
(Member)

________________________

**BUJANG KIM HUAT, PhD**
Professor and Dean
School of Graduate Studies
Universiti Putra Malaysia

Date:
Declaration by graduate student

I hereby confirm that:

- this thesis is my original work;
- quotations, illustrations and citations have been duly referenced;
- this thesis has not been submitted previously or concurrently for any other degree at any other institutions;
- intellectual property from the thesis and copyright of thesis are fully-owned by Universiti Putra Malaysia, as according to the Universiti Putra Malaysia (Research) Rules 2012;
- written permission must be obtained from supervisor and the office of Deputy Vice-Chancellor (Research and Innovation) before thesis is published (in the form of written, printed or in electronic form) including books, journals, modules, proceedings, popular writings, seminar papers, manuscripts, posters, reports, lecture notes, learning modules or any other materials as stated in the Universiti Putra Malaysia (Research) Rules 2012;
- there is no plagiarism or data falsification/fabrication in the thesis, and scholarly integrity is upheld as according to the Universiti Putra Malaysia (Graduate Studies) Rules 2003 (Revision 2012-2013) and the Universiti Putra Malaysia (Research) Rules 2012. The thesis has undergone plagiarism detection software.

Signature: ________________________ Date: __________________

Name and Matric No.: _________________________________________
Declaration by Members of Supervisory Committee

This is to confirm that:

- the research conducted and the writing of this thesis was under our supervision;
- supervision responsibilities as stated in the Universiti Putra Malaysia (Graduate Studies) Rules 2003 (Revision 2012-2013) are adhered to.

Signature: __________________________
Name of Chairman of Supervisory Committee: __________________________

Signature: __________________________
Name of Member of Supervisory Committee: __________________________

Signature: __________________________
Name of Member of Supervisory Committee: __________________________
# TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>Section</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>ABSTRACT</td>
<td>i</td>
</tr>
<tr>
<td>ABSTRAK</td>
<td>iii</td>
</tr>
<tr>
<td>ACKNOWLEDGEMENTS</td>
<td>v</td>
</tr>
<tr>
<td>APPROVAL</td>
<td>vi</td>
</tr>
<tr>
<td>DECLARATION</td>
<td>viii</td>
</tr>
<tr>
<td>LIST OF TABLES</td>
<td>xiii</td>
</tr>
<tr>
<td>LIST OF FIGURES</td>
<td>xiv</td>
</tr>
<tr>
<td>LIST OF ABBREVIATIONS</td>
<td>xvii</td>
</tr>
<tr>
<td>LIST OF NOMENCLATURE</td>
<td>xix</td>
</tr>
</tbody>
</table>

## CHAPTER

### 1 INTRODUCTION

1.1 Background                                                          1  
1.2 Problem Statement                                                  2  
1.3 Objectives                                                          3  
1.4 Scope and Limitations                                              4  
1.5 Significance of the study                                           5  
1.6 Thesis Layout                                                       6  

### 2 LITERATURE REVIEW

2.1 Introduction                                                        7  
2.2 Nuclear Reactor and Safety                                          7  
2.3 Fundamental Design of Reactor Safety                               8  
2.4 Nuclear Research Reactors and Its Applications                      9  
2.5 Reactor Capacity                                                    10  
2.6 Reactor TRIGA Puspati (RTP)                                          11  
2.6.1 Reactor TRIGA Puspati Safety                                     11  
2.6.2 Reactor TRIGA Puspati Neutron Flux                                14  
2.6.3 Reactor TRIGA Puspati Cooling System                              17  
2.7 Research Reactor Upgrading Activities                               21  
2.8 Works on TRIGA Reactors                                             22  
2.9 Reactor Cooling System Studies                                     24  
2.10 Computational Simulation on Reactor Analysis                      28  
2.10.1 System Level 1D Simulation Codes                                 28  
2.10.2 CFD Application on Reactor Analysis                              29  
2.10.3 Development in CFD Codes for nuclear industry                   32  

x
2.10.4 CFD Applications for Reactor TRIGA Puspati (RTP) 34
2.11 Natural Convection Studies on Enclosures 35
2.12 Conclusion 37

3 THEORETICAL CONSIDERATIONS 38
3.1 Introduction 38
3.2 Dimensional Analysis 38
  3.2.1 Buckingham Pi Theorem 38
  3.2.2 Development of dimensionless \( \Pi \) terms 39
  3.2.3 Geometric similarity 41
  3.2.4 Kinematic similarity 42
3.3 CFD Modeling 43
  3.3.1 Reactor pool model 44
  3.3.2 Governing Equation 47
  3.3.3 Solver Scheme 51
    3.3.3.1 Control volume 52
    3.3.3.2 Cell interpolation scheme 52
    3.3.3.3 Under-relaxation factor 52
    3.3.3.4 Convergence criteria 52
3.4 Closing Summary 53

4 METHODOLOGY 54
4.1 Introduction 54
4.2 FLUENT Modeling 57
  4.2.1 Modeling Scheme 58
  4.2.2 Solver Scheme 60
  4.2.3 Modeling Validation 63
4.3 Cooling Enhancement 64
4.4 Experimental set up 65
4.5 Experimental Procedure 73
4.6 Conclusion 74

5 RESULTS AND DISCUSSION 75
5.1 Introduction 75
5.2 Dimensional Analysis 75
  5.2.1 Development of correlation 75
  5.2.2 Experimental scale determination 83
  5.2.3 Summary 84
5.3 Modeling Validation 86
  5.3.1 Grid Independence Study 86
  5.3.2 Modeling Verification 88
5.4 Experimental Results 91
5.5 Natural Convection Phenomena 97
5.6 Cooling Enhancement 99
5.7 Optimal Conditions 105
5.8 Heat Transfer Correlations 110
5.9 Modeling Scheme Codes 115
5.10 Closing Summary 118
6 CONCLUSION AND RECOMMENDATION 119
6.1 Conclusion 119
6.2 Recommendation and Further Works 121

REFERENCES 123

APPENDICES 146
A Cooling Coil Design Calculation 146
B AutoLISP Programming Codes 150
C AutoLISP Modeling Codes Output 156
D Regression Analysis of the Heat Transfer Correlation 160
E Statistical Analysis on the Experiments 175
F Supplier’s Calibration Report 180

BIODATA OF STUDENT 183
LIST OF PUBLICATIONS 184
# LIST OF TABLES

<table>
<thead>
<tr>
<th>Table</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.1</td>
<td>Neutron flux capacity and its typical applications</td>
</tr>
<tr>
<td>3.1</td>
<td>Operating data of the plant</td>
</tr>
<tr>
<td>4.1</td>
<td>Modeling scheme of the simulation models</td>
</tr>
<tr>
<td>4.2</td>
<td>Experimental set up conditions of all case studies</td>
</tr>
<tr>
<td>5.1</td>
<td>Data of the system simulation</td>
</tr>
<tr>
<td>5.2</td>
<td>Analysed data to fit the I1 terms</td>
</tr>
<tr>
<td>5.3</td>
<td>Data of the physical experiment</td>
</tr>
<tr>
<td>5.4</td>
<td>Calculation of data to I1 terms</td>
</tr>
<tr>
<td>5.5</td>
<td>Details of the data from the correlation, simulation and experiment.</td>
</tr>
<tr>
<td>5.6</td>
<td>Data from the real plant</td>
</tr>
<tr>
<td>5.7</td>
<td>Calculation data for minimum and maximum condition at various scale</td>
</tr>
<tr>
<td>5.8</td>
<td>Minimum and maximum power requirement at various scale model</td>
</tr>
<tr>
<td>5.9</td>
<td>Grid independent study data</td>
</tr>
<tr>
<td>5.10</td>
<td>Overall results of enhancement study on 1/10th model</td>
</tr>
<tr>
<td>5.11</td>
<td>Overall results of enhancement study on full model</td>
</tr>
</tbody>
</table>
LIST OF FIGURES

Table | Page
---|---
2.1 | Schematics of Nuclear Power Plant 8
2.2 | Schematics of Nuclear Research Reactor 9
2.3 | Reactor TRIGA PUSPATI facility 11
2.4 | RTP in-core and out-core facilities 15
2.5 | Cross section of RTP central thimble facility 16
2.6 | RTP beam ports and research facilities 17
2.7 | Reactor pool cross sectional diagram 19
2.8 | RTP cooling system schematic 20
2.9 | IPR-R1 TRIGA nodalization layout using the RELAP5 model 29
2.10 | Counter current flow modeling results by experiment and CFD calculation. 30
2.11 | Sample of simulation result for flow vector of RTP primary cooling pipe. 38
3.1 | Schematic of the heat transfer under study 34
3.2 | Vertical configuration of the reactor. 39
3.3 | Cross section of the reactor 45
3.4 | Reactor pool dimensions used for the full reactor model. 46
4.1 | Overall methodology of the study 55
4.2 | 3D CFD model of the RTP pool system 58
4.3 | Schematic of the reactor pool simulation model 59
4.4 | Example of scaled residuals monitor. 61
4.5 | Result from a 2D slice of temperature profile in 3D simulation. 62
4.6 | Result from a 2D slice of stream profile in 3D simulation 62
4.7 | X-Y plot of a temperature profile result 63
4.8 | Determination of the independent grid scheme 64
4.9 | Schematics of the experimental set up 66
4.10 | Overall view of the experimental set up. 68
4.11 | Heating coil 68
4.12 | Thermocouple type K 68
4.13 | Voltage regulator 69
4.14 | Direct temperature display 69
4.15 | The in-pool pump 69
4.16 | The installation of the in-pool pump 69
4.17 | Low temperature cooling rig 70
4.18 | Schematics of the low temperature cooling 71
4.19 | Installation of the cooling coil. 72
4.20 | Installation of the conducting rods 72
4.21 | Portable infrared camera, IR FlexCam Thermal Imager. 72
5.1 | Plot of $\Pi_1 (hx/k)$ and $\Pi_2 (\rho C_p g \beta dT x^3 / k)$ 77
5.2 | Plot of log scale with regression results 78
5.3 Comparison between the correlated equation data and simulation data.
5.4 Comparison between the correlated equation data and simulation data on log scale.
5.5 Comparison between physical experiment, simulation and correlated data
5.6 Calculated heat transfer coefficient and Nusselts number at different grid schemes
5.7 Calculated temperature at different grid schemes
5.8 Calculated core temperature and convergence time
5.9 Typical profile of velocity and temperature in natural convection over a surface
5.10 Temperature and velocity profile of the model
5.11 Stream profiles from Nakhi et. al. (2008) on the left side and the reactor pool model on right side.
5.12 Stream function profiles of the simulation model (left) and velocity vectors by Rodriguez et. al. (2009) on the right hand side.
5.13 Result of normal natural convection cooling
5.14 Result of low temperature cooling
5.15 Result of in-pool heat exchanger cooling
5.16 Result of in-pool pumps cooling
5.17 Result of conduction rods cooling
5.18 Thermal images of the experimental scaled model
5.19 Thermal image of a boiling electric kettle
5.20 Temperature profile (a) and flow stream (b) of natural convection simulation as compared to temperature profile without natural convection simulation (c) where no flow contour produced.
5.21 Trend between natural convection and pure conduction simulation
5.22 Stream profiles and temperature profiles of normal cooling (a and c) and in-pool pumps cooling (b and d).
5.23 Effect of heat flux increment for different techniques on 1/10th scale model
5.24 Stream and temperature profiles of low temperature cooling
5.25 Stream and temperature profiles for conduction rods technique
5.26 Stream and temperature profiles for in-pool heat exchanger technique.
5.27 Effect of heat flux increment for different techniques on full scale model
<table>
<thead>
<tr>
<th>Topic</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pump capacity at different heat flux</td>
<td>106</td>
</tr>
<tr>
<td>Heat exchanger capacity at different heat flux</td>
<td>106</td>
</tr>
<tr>
<td>Effect of position on in-pool pump technique</td>
<td>107</td>
</tr>
<tr>
<td>Effect of position on in-pool heat exchanger technique</td>
<td>107</td>
</tr>
<tr>
<td>Effect of quantity on in-pool heat exchanger technique</td>
<td>108</td>
</tr>
<tr>
<td>Effect of inlet capacity on low temperature cooling technique</td>
<td>109</td>
</tr>
<tr>
<td>Effect of dimension on conduction rods technique</td>
<td>109</td>
</tr>
<tr>
<td>Heat transfer correlation of the experimental model at 26°C operating condition.</td>
<td>111</td>
</tr>
<tr>
<td>Heat transfer correlation of the real plant at 32°C and 45°C operating condition.</td>
<td>112</td>
</tr>
<tr>
<td>Heat transfer correlation of the real plant for all the cooling enhancement techniques at 32°C operating condition.</td>
<td>114</td>
</tr>
<tr>
<td>Sample of the programming code</td>
<td>116</td>
</tr>
<tr>
<td>Part of the display on output of the code</td>
<td>117</td>
</tr>
</tbody>
</table>
# LIST OF ABBREVIATIONS

<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>AELB</td>
<td>Atomic Energy Licensing Board</td>
</tr>
<tr>
<td>AMG</td>
<td>Algebraic Multigrid</td>
</tr>
<tr>
<td>ASME</td>
<td>American Society of Mechanical Engineers</td>
</tr>
<tr>
<td>AWS</td>
<td>Auxiliary Water Source</td>
</tr>
<tr>
<td>BAEC</td>
<td>Bangladesh Atomic Energy Commission</td>
</tr>
<tr>
<td>BPG</td>
<td>Best Practice Guidelines</td>
</tr>
<tr>
<td>BWR</td>
<td>Boiling Water Reactor</td>
</tr>
<tr>
<td>CFD</td>
<td>Computational Fluid Dynamics</td>
</tr>
<tr>
<td>CHF</td>
<td>Critical Heat Flux</td>
</tr>
<tr>
<td>CT</td>
<td>Central Thimble</td>
</tr>
<tr>
<td>DNAA</td>
<td>Delayed Neutron Activation Analysis</td>
</tr>
<tr>
<td>DNBR</td>
<td>Departure to Nucleate Boiling Ratio</td>
</tr>
<tr>
<td>ECCS</td>
<td>Emergency Core Cooling System</td>
</tr>
<tr>
<td>ECORA</td>
<td>Evaluation of CFD Methods for Reactor Safety Analysis</td>
</tr>
<tr>
<td>FBR</td>
<td>fast breeder reactor</td>
</tr>
<tr>
<td>FDM</td>
<td>Finite difference method</td>
</tr>
<tr>
<td>FEM</td>
<td>Finite element method</td>
</tr>
<tr>
<td>FVM</td>
<td>Finite volume method</td>
</tr>
<tr>
<td>HTTR</td>
<td>High Temperature Testing Reactor</td>
</tr>
<tr>
<td>HTGR</td>
<td>High Temperature Gas-Cooled Reactor</td>
</tr>
<tr>
<td>IAEA</td>
<td>International Atomic Energy Agency</td>
</tr>
<tr>
<td>IVR</td>
<td>in vessel retention</td>
</tr>
<tr>
<td>LHSS</td>
<td>Latent Heat Storage System</td>
</tr>
<tr>
<td>LOCA</td>
<td>Loss of Coolant Accident</td>
</tr>
<tr>
<td>LOFA</td>
<td>Loss of Flow Accident</td>
</tr>
<tr>
<td>LMTD</td>
<td>log mean temperature difference</td>
</tr>
<tr>
<td>LWR</td>
<td>Light Water Reactors</td>
</tr>
<tr>
<td>MIT</td>
<td>Massachusetts Institute of Technology</td>
</tr>
<tr>
<td>MNSR</td>
<td>Miniature Neutron Source Reactor</td>
</tr>
<tr>
<td>MOX</td>
<td>Mixed Oxides</td>
</tr>
<tr>
<td>MYRRHA</td>
<td>Multipurpose hYbrid Research Reactor for High-tech Application</td>
</tr>
<tr>
<td>NAA</td>
<td>neutron activation analysis</td>
</tr>
<tr>
<td>NEA</td>
<td>Nuclear Energy Agency</td>
</tr>
<tr>
<td>NEA-CSNI</td>
<td>Nuclear Energy Agency Committee on the Safety of Nuclear Installation</td>
</tr>
<tr>
<td>OLC</td>
<td>operating limit condition</td>
</tr>
<tr>
<td>PBMR</td>
<td>Pebble Bed Modular Reactor</td>
</tr>
<tr>
<td>PTS</td>
<td>Pneumatic Transfer System</td>
</tr>
<tr>
<td>PUSPATI</td>
<td>Pusat Penyelidikan Atom Tun Dr Ismail</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurized Water Reactors</td>
</tr>
<tr>
<td>RCCS</td>
<td>Reactor Cavity Cooling System</td>
</tr>
<tr>
<td>RTP</td>
<td>Reactor TRIGA PUSPATI</td>
</tr>
<tr>
<td>SAR</td>
<td>Safety Analysis Report</td>
</tr>
<tr>
<td>TES</td>
<td>Thermal Energy Storage</td>
</tr>
<tr>
<td>USA</td>
<td>United States of America</td>
</tr>
</tbody>
</table>
LIST OF NOMENCLATURE

The following is a list of definitions of the main symbols used in this thesis. SI units are considered in the study.

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Description</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>A</td>
<td>Surface area</td>
<td>[m$^2$]</td>
</tr>
<tr>
<td>$C_p$</td>
<td>Specific heat</td>
<td>[J kg$^{-1}$ K$^{-1}$]</td>
</tr>
<tr>
<td>$dT$</td>
<td>Temperature difference</td>
<td>[K]</td>
</tr>
<tr>
<td>D</td>
<td>Diameter of cylinder</td>
<td>[m]</td>
</tr>
<tr>
<td>$E$</td>
<td>Energy due to heat, pressure and work</td>
<td>[W]</td>
</tr>
<tr>
<td>$g$</td>
<td>Gravitational acceleration</td>
<td>[m s$^{-2}$]</td>
</tr>
<tr>
<td>$h$</td>
<td>convective heat transfer coefficient</td>
<td>[W m$^{-2}$ K$^{-1}$]</td>
</tr>
<tr>
<td>$h$</td>
<td>Sensible enthalpy</td>
<td>[kJ/kg]</td>
</tr>
<tr>
<td>H</td>
<td>Height of cylinder</td>
<td>[m]</td>
</tr>
<tr>
<td>$k$</td>
<td>fluid thermal conductivity</td>
<td>[W m$^{-1}$ K$^{-1}$]</td>
</tr>
<tr>
<td>L</td>
<td>Length of body</td>
<td>[m]</td>
</tr>
<tr>
<td>$p$</td>
<td>Pressure</td>
<td>[kg/m$^3$]</td>
</tr>
<tr>
<td>Q</td>
<td>Power applied</td>
<td>[W]</td>
</tr>
<tr>
<td>$q''$</td>
<td>Surface heat flux</td>
<td>[W m$^{-2}$]</td>
</tr>
<tr>
<td>$q'''$</td>
<td>Volumetric heat flux</td>
<td>[W m$^{-3}$]</td>
</tr>
<tr>
<td>$r$</td>
<td>Radius</td>
<td>[m]</td>
</tr>
<tr>
<td>$S_h$</td>
<td>Volumetric heat source</td>
<td>[W m$^{-3}$]</td>
</tr>
<tr>
<td>$T$</td>
<td>Fluid temperature</td>
<td>[K]</td>
</tr>
<tr>
<td>$T_c$</td>
<td>Core temperature</td>
<td>[K]</td>
</tr>
<tr>
<td>$T_o$</td>
<td>Operating temperature</td>
<td>[K]</td>
</tr>
<tr>
<td>$T_s$</td>
<td>Surface temperature</td>
<td>[K]</td>
</tr>
<tr>
<td>$T_f$</td>
<td>Fluid temperature</td>
<td>[K]</td>
</tr>
<tr>
<td>$T_\infty$</td>
<td>Bulk temperature</td>
<td>[K]</td>
</tr>
<tr>
<td>$T_{ref}$</td>
<td>Reference temperature</td>
<td>[K]</td>
</tr>
<tr>
<td>$v$</td>
<td>Velocity</td>
<td>[m s$^{-1}$]</td>
</tr>
<tr>
<td>$V_{max}$</td>
<td>Maximum velocity</td>
<td>[m s$^{-1}$]</td>
</tr>
<tr>
<td>x</td>
<td>Length</td>
<td>[m]</td>
</tr>
<tr>
<td>Y</td>
<td>Mass fraction</td>
<td>[kg]</td>
</tr>
</tbody>
</table>

Greek Letters

<table>
<thead>
<tr>
<th>Letter</th>
<th>Description</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\rho$</td>
<td>Fluid density</td>
<td>[kg/m$^3$]</td>
</tr>
<tr>
<td>$\rho_o$</td>
<td>Operating density</td>
<td>[kg/m$^3$]</td>
</tr>
<tr>
<td>$\rho_\infty$</td>
<td>Bulk density</td>
<td>[kg/m$^3$]</td>
</tr>
<tr>
<td>$\mu$</td>
<td>Kinematic viscosity</td>
<td>[Nm s$^{-1}$]</td>
</tr>
<tr>
<td>$\beta$</td>
<td>Fluid thermal expansion</td>
<td>[K$^{-1}$]</td>
</tr>
<tr>
<td>$\tau$</td>
<td>Stress tensor</td>
<td>[Pa]</td>
</tr>
<tr>
<td>$\nu$</td>
<td>Fluid specific volume</td>
<td>[m$^3$]</td>
</tr>
<tr>
<td>$\alpha$</td>
<td>Thermal diffusivity</td>
<td>[m$^2$ s$^{-1}$]</td>
</tr>
<tr>
<td>$I$</td>
<td>Unit tensor</td>
<td></td>
</tr>
</tbody>
</table>
\( \theta \) Dimensionless temperature
\( \pi \) Pi term in Buckingham theory
\( \Delta \) Difference in quantity
\( \emptyset \) Unknown
CHAPTER 1

INTRODUCTION

1.1 Background

The first nuclear research reactor was officially operated on 2 December 1942 in Chicago, U.S.A. At present, there are 246 research reactors in operation in about 58 countries around the globe (IAEA, 2013). From the inception of International Atomic Energy Agency (IAEA) in 1957 the emphasis has always been on the safe operation of the reactors. Much has been achieved in the development of safety guides and requirements for nuclear reactors from the operational practice and experience of the countries involved. The main objective is to ensure the safe operation of the nuclear reactors for the protection of health, life and property in the development and application of nuclear energy for peaceful purposes (IAEA, 2013).

However, worldwide publicity on a few nuclear accidents such as Three Miles Island, Chernobyl, Fukushima as well as the notorious Hiroshima and Nagasaki bombing have always bring about general public fear on anything related to nuclear. Most findings on the nuclear reactors accidents are closely related to the reactor cooling system (Saenko, 2011, Hiroyashu, 2007, Balonov, 2007). In Three Miles Island case, the defective instrumentation has resulted in operators’ error which eventually led to the melting of part of the reactor core due to lack of coolant inside the reactor. Later in 1986, the worst nuclear accident happened in Chernobyl nuclear reactor. A nuclear explosion had ruptured the reactor vessel and the reactor building, which released large amount of radioactive contaminant to the surrounding area. This radioactive pollutant was then spread to the Western Soviet Union, Eastern Europe, Western Europe, Northern Europe, and Eastern North America. Ukraine, Belarus and Russia received the most contamination resulted in the evacuation and resettlement of about 336,000 people (WNA, 2013). The sequence of event in the accident revealed that the flaw in the control rod design had initially reduced the amount of coolant present, which means that the shut-down actually increased the reaction rate. This had led to energy spike and damaging some of the fuel rods. The fuel rod fragments prevented the insertion of the control rods, thus unable to stop the reaction which eventually led to the nuclear explosion (WNA, 2009 and IAEA, 1992).

The latest unfortunate accident of Fukushima Daiichi has also rendered the same story. The 9.0 Richter scale earthquake has flooded the nuclear facility and damaging all the power supply systems. Without power supply, all emergency cooling systems failed to operate and decreasing the coolant level inside the core. Part of the fuel rods believed to have been melted and some of the radioactive contaminants were released inside the building. Later, several hydrogen explosions have damage the reactor building and released the pollutant to the surrounding area (WNA, 2014). Thus, taking the lesson from the above accidents, the most emphasized area in nuclear reactor safety
analysis is the loss of coolant flow accident (LOFA) and loss of coolant accident (LOCA) (Nuclear Malaysia, 2008). Any effort to further analyse the cooling behavior of a reactor pool would be beneficial to understand its mechanics throughout an operation of the reactor which will enable necessary action or modification to be taken if the need arise.

The above discussion on nuclear accidents showed that one of the most important safety features in nuclear reactor design is its cooling system. The study or analysis on cooling systems involves thermal and hydraulics mechanism of the cooling process. It involves the study on the heat transfer process inside the reactor core. It also involves the hydraulics behavior of the coolant fluid as a result of the thermal process and the effect of dimensional configuration inside the system. The process could be in a single or multi-phased coolant fluid. The process may happen inside a complex structure and involves multi-components. Because of the potentially complicated nature of thermal and hydraulic engineering analysis, the methods used to simulate the behavior of the thermal and fluid flow usually involve complex computational modeling and simulation.

In the early development of nuclear reactor the investigation and analysis were conducted by analytical calculations and physical experimentations. Later, with the emergence of computing technology, 1D system codes were widely used until the last decade. The lack of detail predictions especially on component and physical level has attracted the usage of 3D simulation codes to nuclear reactor safety analysis. While CFD codes have been widely used outside the nuclear industry, it is yet to be developed to achieve the necessary credibility in nuclear industry. Thus, the use of CFD simulation in any of nuclear reactor case study would contribute toward the development of CFD usage in the nuclear industry.

1.2 Problem Statement

Malaysia started to construct its own nuclear research reactor in 1972 and completed for operation in 1982. The Reactor TRIGA PUSPATI (RTP) is the only nuclear research reactor in the country. It is an open pool cooling type, with thermal capacity of 1 Mega Watt. The reactor is used for nuclear research application and educational purposes. The safe operation of the reactor is under the jurisdiction of the Atomic Energy Licensing Board (AELB) with the assistance from International Atomic Energy Agency (IAEA). Among the most important report that is required by the licensing authority is the Safety Analysis Report (SAR).

Malaysian RTP has the first Safety Analysis Report (SAR) back in 1983. The analysis was based on the manufacturer’s initial analysis. Most of the content based on qualitative analysis and lack of quantitative supports (PUSPATI, 1983). The second and subsequent revisions were conducted in 2003, 2004, 2005, 2007 and 2008. Again, the specific analysis on reactor safety was not
changed. Recently in 2011 another revision was conducted. Unfortunately there was no actual quantitative analysis conducted on the present configuration of the reactor. After more than 30 years in operation, the fuel configurations have been changed several times to maintain the desired power level. The safety analysis for the SAR is still based on the manufacturer's analysis back in 1983. Moreover, some of the computational analysis conducted using obsolete computer codes (Nuclear Malaysia, 2008). Thus, the safety analysis does not represent the present state of the reactor. In order to understand the actual condition of the reactor, the relevant quantitative analysis shall be conducted. SAR analysis required for research reactor consists of comprehensive analysis on each potential risk ranges from electrical power, reactor operating systems including primary and secondary systems, reactor auxiliary system, support systems, containment building as well as environmental factors surrounding the reactor. However, the most severe condition which needs to be analysed for most reactors are the loss of flow accident (LOFA) and loss of coolant accident (LOCA). These two conditions occurred when the reactor began to lose its cooling capability. Thus, the study on the reactor cooling system would be a novel effort to actually understand the safety condition of PUSPATI TRIGA research reactor. It will also become a credible supporting data for the current RTP’s SAR.

One of the shortcomings of the current RTP is its low neutron flux which offers limited research and development applications. There are planning to upgrade the reactor power in order to produce higher neutron flux (Iorgulis et. al., 2011, Hairi et. al., 2011). A higher power reactor would require bigger cooling pool to control the core temperature. This will incur additional cost as well as construction space. In order to avoid these additional requirements a technical feasibility study has to be conducted to enhance the cooling system to accommodate for a higher flux requirements.

In view of these two prevailing issues, there is a need to conduct quantitative analysis on the reactor cooling pool condition for safety assessment as well and enhancement study on the current cooling system to cater for higher flux reactor.

1.3 Objectives

Taking into account the previous studies conducted by Ravnik (1995), Iorgulis et. al. (2010) and M. Hairie et. al. (2011) on the requirement of neutron flux and the required cooling capacity for RTP upgrading, the following objectives are established.

The main objective of this project is to improve the cooling capacity to at least twofold of the existing reactor cooling pool capacity, to enable it to be used for higher flux reactor, and to come up with a quantitative analysis on the characteristics and behavior of the RTP pool cooling system.
The above will be accomplished through the following specific objectives:

1. To characterize the thermal and dynamics properties of the reactor core and reactor pool under the present core configuration.

2. To model and simulate the reactor core and reactor pool behavior using computational fluid dynamics method to investigate the thermal and dynamics behavior of the cooling system.

3. To verify the simulation model through an experimental investigation on the thermal behavior of a scale model of the reactor pool cooling system under the present operating condition of RTP.

4. To enhance the cooling capacity of the current reactor cooling pool to at least twice the current capacity to be used for higher flux reactors by manipulation of the present pool conditions and dynamics of the cooling water.

1.4 Scope and Limitations

The following are the scope and limitation of the study based on the actual simulation and experimental condition.

1. The simulation study is conducted only to the reactor pool and its components rather than the whole reactor cooling system which consists of other components such as the heat exchanger, cooling tower, purification system and the pipes and valves.

2. The reactor operation is assumed at maximum power and operated in steady state condition where the highest heat flux would be produced.

3. The study is on single phase fluid condition as it is the normal operating condition of Reaktor TRIGA Puspati (RTP).

4. In general, the pool system is at laminar state condition under normal operation. However, any other additional mechanisms introduced into the system may change this condition, which will have to be investigated further and accounted for.

5. The simulation is limited to the available computational power and capacity.

6. The study is limited to the experimental model capacity, operating condition, control parameters and instrumentation technical specifications.
1.5 Significance of the Study

The feasibility study to use existing Reactor TIGA PUSPATI (RTP) cooling pool system for a higher flux reactor is part of the effort to improve the safety of the reactor. This effort may have several impacts to the industry, the reactor engineering community as well as the natural convection field.

As mentioned earlier most findings on nuclear reactor accidents are closely related to the reactor cooling system (Saenko, 2011, Hiroyashu, 2007, Balonov, 2007). Thus, the understanding of the behavior of the reactor pool cooling system which is part of this study, is very important to ensure the further development and improvement on safety system of nuclear reactors.

Most of the research reactors undergoing power or flux upgrading have increased the primary and secondary cooling capacity through replacement of heat exchangers, cooling towers and piping circuits (Zulqurnain et. al, 2009; Toszer, 2009; Israr et. al, 2009; Yazid et. al, 2009). This will incur additional cost as well as new space requirement. The study to enhance the pool cooling capacity of current RTP without increasing all the auxiliary equipments for higher flux reactor, would bring about a new system application which will benefit the research reactor community in upgrading activities as well as for new research reactor designs.

The study for RTP cooling pool enhancement involved the understanding of the dynamics of the heat and mass transfer inside the system which is mainly based on natural convection process. A review on present natural convection works as discussed in detail in section 2.11 of Chapter 2, revealed that most of the studies are done on rectangle cases with external heat sources; and very few cases on cylindrical shape. The study on RTP will not only contribute to the expansion of the study area but also venturing into a new unique configuration of natural convection study.

The following are the direct and indirect output from the study that will contribute to the nuclear reactor technology and natural convection technology:

- The direct output of the study would be the baseline data on the characteristic of Reactor TRIGA Puspati core and pool cooling system under normal operating condition.

- A novel CFD model of RTP Pool Cooling System which could be used for further analysis of the reactor safety as well as other similar TRIGA reactors.

- A novel thermal study on open top vertical cylinder with internal heat source.
• A proposed system for cooling enhancement to be used by the same pool configuration for a higher power reactor.

• A set of new heat transfer correlations (HTC) for the unique case and enhancement techniques established.

• AUTOLISP codes for basic setting of RTP modeling scheme for natural convection simulation.

The output and data obtained from the study would be useful not only for the RTP development, but also as a baseline data for further development of nuclear reactors either to upgrade into higher power or for future design of nuclear power plant.

The safety management on major facility such as Puspati TRIGA reactor would be improved further with the experience gained in conducting the study. The present condition, in the absence or lack of fundamental study on the reactor system, any evaluation in regard of safety has usually conducted based on superficial judgment. Thus, the quantitative data obtained would assist further understanding of the system safety requirements.

The relation between Puspati TRIGA reactor or Nuclear Malaysia as a research reactor operating organization with other safeguard counterparts will be improved with the novel effort to conduct real quantitative study on the reactor present operating condition pertaining to the safety condition of the reactor. This may bring more benefit to Nuclear Malaysia in term of technical and financial supports from those organizations for any future undertakings.

1.6 Thesis Layout

Chapter one contains the introduction, problem statement, objectives, scope and limitations, output and benefit of the research. Chapter two includes a critical review on previous works and explains the related nuclear reactor technology which includes the safety system, application, modeling and the parameters affecting the system. A review on the recent works on natural convection has also been conducted. Chapter three and four presents the theoretical aspect and methodology involved throughout the study. It covers the numerical theory, modeling scheme, solver scheme, model validation methods as well as the experimental set up and procedures. The dimensional analysis method of the experimental model and cooling enhancement method was also discussed. Chapter five discusses the results of the study which includes the results of the dimensional analysis, development of heat transfer correlations, the modeling validation and verification, experimental results, natural convection modeling, cooling enhancement regimes and the developed modeling scheme codes. Finally, chapter six contains the conclusions and recommendations for future works.
REFERENCES


COSMOL. (2011). Heat transfer by free convection. COSMOL Inc.


Frisch, W., & Gros, G. (2001). Improving the safety of future nuclear fission power plants. Fusion Engineering and Design, 56-57(0), 83-93. doi:10.1016/S0920-3796(01)00238-1


Oliver, Y.P. (2011). Development of an educational tool for the fixed bed adsorption problem. Final Year Project, National University of Singapore.


Orlov, M.L. (1996). Multiple linear regression analysis using Microsoft excel. Chemistry Department, Oregon State University, OR.


