UNIVERSITI PUTRA MALAYSIA

CFD MODELING OF PERFORMANCE ENHANCEMENT FOR NUCLEAR RESEARCH REACTOR COOLING SYSTEM

ROSLI BIN DARMAWAN

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By

ROSLI BIN DARMAWAN

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ROSLI BIN DARMAWAN

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^{o}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/m^2 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.

Abstrak tesis yang dikemukakan kepada Senat Universiti Putra Malaysia sebagai memenuhi keperluan untuk ijazah Doktor Falsafah

PEMODELAN CFD UNTUK PENINGKATAN PRESTASI SISTEM PENYEJUK REAKTOR PENYELIDIKAN NUKLEAR

Oleh

ROSLI BIN DARMAWAN

March 2015

Pengerusi: Nor Mariah Adam, PhD, PE.

Fakulti: Kejuruteraan

Reaktor TRIGA PUSPATI (RTP), sebuah reaktor penyelidikan nuklear jenis kolam yang mula beroperasi pada 1982 dengan kuasa termal 1 MW, adalah dikelaskan sebagai reaktor fluks rendah yang mempunyai applikasi yang sangat terhad. Kebanyakan reaktor seumpamanya telah melalui proses penambahbaikan untuk meningkatkan kapasiti yang melibatkan pemasangan sistem pembuangan haba berkapasiti tinggi. Pendekatan ini melibatkan kos lebih tinggi dan pengubahsuaian besar ke atas kolam reaktor. Kaedah lain yang hanya memerlukan sedikit pengubahsuian telah dikaji.

Kolam penyejukan RTP beroperasi dengan air biasa pada suhu 32^{o}C, beraliran laminar dan berkeadaan mantap, dikelilingi dinding dan lantai adiabatik juga bahagian atas terbuka yang terdedah pada suhu bilik. Punca haba adalah dari teras reaktor dengan fluks haba 550W/m^{2} dan disejukkan melalui proses olakan semulajadi. Keadaan ini telah dimodel menggunakan kod pengkomputeran CFD FLUENT V6.3. Persamaan keabadian aliran, persamaan momentum, persamaan keterusan dan persamaan tenaga diselesaikan untuk menjangka kelakuan hidrodinamik dan termal model tersebut. Pemodelan olakan semulajadi telah menggunakan model anggaran Boussinesq untuk terma apungan demi mencapai penyelesaian lebih pantas. Sebuah model prototaip berskala 1/10 telah dibangunkan untuk tujuan verifikasi keputusan simulasi. Analisa dimensi menggunakan teori Buckingham Π telah dilaksanakan untuk membangun kolerasi tanpa dimensi bagi pencirian sistem juga memastikan kesamaan geometri dan kinematik model tersebut.

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.

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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)

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

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF ABBREVIATIONS

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

USNRC	United States Nuclear Reactors Commission
VHTR	Very High Temperature gas-cooled Reactor
WNA	World Nuclear Association

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.

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

Greek Letters

ρ	Fluid density	[kg/m^3]
ρ_o	Operating density	[kg/m^3]
ρ_∞	Bulk density	[kg/m^3]
μ	Kinematic viscosity	[Nm s^{-1}]
β	Fluid thermal expansion	[K^{-1}]
τ	Stress tensor	[Pa]
ν	Fluid specific volume	[m^3]
α	Thermal diffusivity	[m^2 s^{-1}]
I	Unit tensor	

θ	Dimensionless temperature
π	Pi term in Buckingham theory
Δ	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[a], 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[b], 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 ([a]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[b] 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 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.

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