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Thermal-fluid modeling of the Missouri S&T reactor

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THERMAL-FLUID MODELING OF THE MISSOURI S&T REACTOR

by

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(3) Pages 43-71, *Prediction of Missouri S&T Reactor's natural convection with porous media approximation.*

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(4) Pages 72-89, *Supply chain feasibility analysis of small modular reactor technology*.

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ABSTRACT

Thermal-fluid modeling of the Missouri University of Science and Technology Reactor (MSTR) was carried out using a computational fluid dynamics code (CFD), STAR-CCM+. First, a three-dimensional parallel-plate model was developed, and the cosine-shaped heat flux was applied to the MSTR core. Simulation results for fluid flow under natural convection condition show coolant temperature and velocity as a function of core power. A characteristic equation for the parallel-plate model was obtained based on Forchheimer's flow equation. The inertial resistance tensor and viscous resistance tensor were found to be 281005 kg/m^4 and 7121.6 kg/m^3 respectively. The MSTR core was then defined as a porous region with porosity 0.7027. A second model was developed to study convection within a section of the MSTR includes 3 fuel elements (power density of 1.86E+6 Wm^{-3}) in one third of the reactor pool volume. For validation work, both plume temperature and pool temperature measurements were recorded at several locations within the MSTR pool. At 200kW, the temperature field was consistent with the pool temperature data at 15 locations. A third model included the workings of an eductor outlet from, and inlet into the active cooling system to predict heat removal capability. The major contribution of this study is to explain the thermal flow in the MSTR channels and pool, and to provide a framework for supporting reactor license renewal, and power uprate plans.

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1. INTRODUCTION

Thermal fluid and criticality analysis of a nuclear reactor is performed to study the safety aspects of reactor operations [1, 2, 3]. The analysis provides evidence for the safety of a new reactor design as well as of the reactor's operational ability to keep within its thermal safety limits. In planning for a modified design such as reactor power uprate and core reconfiguration, several studies are carried out to predict the reactor's behavior at the new power and compared with the current operating level [4-7]. Two areas of study that are inextricably connected in any reactor system are reactor kinetics and thermal hydraulics [8, 9]. Both subjects each have a set of assumptions and governing equations that define the fission reactor behavior, and heat removal mechanisms. Computer codes are widely used to study and predict reactor performance in various normal and accident conditions [10-14]. Analyses using different codes provide a framework in which the reactor is expected to work, and support the process of amending an operating license to increase the reactor maximum thermal power level.

Passive cooling systems have been a part of many nuclear designs including core cooling, and steam supply system [15]. Recent designs of water cooled and moderated nuclear fission reactors, including Small Modular Reactors (SMR) have gained significant interest. Most SMR designs utilize natural convection mode for cooling the reactor core. These designs are driven by the demand for safer reactors with the ability to maintain safety under worst emergency situation and would not damage the core nor release radioactivity [16, 17]. In these systems, natural convection is a dominant process to remove heat from the heat-generating core [16]. One of the key concerns in reactor operations and reactor safety is maintaining temperature control of the core, and regulating the coolant’s temperature through a reactor’s cooling system [8, 9]. Thermal hydraulic studies and nuclear safety analysis often utilize system codes, subchannel codes and computational fluid dynamics (CFD) codes to understand a reactor’s behavior and predict the range of thermal parameters under various operating conditions [9]. There is an increasing level of acceptance to use the CFD method to analyze core behavior in nuclear power reactors as well as research reactors [9-12]. The selection of CFD codes as a nuclear thermal hydraulics analysis tool has many benefits. The code allows in-depth analysis of local temperature and flow fields around fuel geometries and in internal components of a reactor [17-22].

The Missouri University of Science and Technology Reactor (MSTR) is a 200kW research reactor, and have been in operation since 1961 [3]. There are initial plans to increase its reactor power to either 500kW or 1MW reactor. This power uprate will require extensive safety analysis. This work is the preliminary thermal hydraulics safety analysis of MSTR for a possible power uprate. However, the results from this study can be readily extended for any natural convection based core cooling system like the one proposed for SMR. CFD models were developed and provided analysis of the natural convection cooling of the MSTR core.

In July 2013, the Small Modular Reactor Research and Education Consortium (SMRrec) was established at Missouri S&T. Together with Ameren and Westinghouse, a project was initiated to support a state-wide strategy to position Missouri as a manufacturing hub for small modular reactors (SMRs). One of the goals for the project is to evaluate the status of the supply chain for SMR industry if Missouri were to serve as the manufacturing hub. This project successfully developed a supply chain model for Missouri SMR operations and is reported as part of this dissertation.

Several studies have been completed to investigate the MSTR (formerly known as University of Missouri-Rolla Reactor (UMMR)) core [4, 5, 23-26]. An extensive safety analyses was performed at the time of fuel conversion from HEU to LEU fuel [4]. Safety analyses on the core loading of the UMMR were studied using PARET and CONVEC codes [5]. Previous work done on the MSTR also included; axial flux measurements, addition of active cooling system, bench marking of neutron energy spectrum and development of and validation MCNP models for the entire core. However, temperature considerations were limited in these models [24, 25]. The neutron cross-sections were defined at a core-averaged temperature as opposed to using accurate temperature profile of the core. In addition, core flow measurements have not been successfully carried out so as to map the flow field in the reactor pool and around the core. The work reported in this dissertation relates to the development of a thermal-fluid model of the core through the use of computational fluid dynamics codes, STAR-CCM+, and the temperature measurements to validate the CFD models [27-29, Appendix A-C]. The modeling of the entire core adopted the porous media approach [9]. This approach is necessary to circumvent the extensive computer resource requirements which are often not readily available. It is intended that the results from this work provide a framework for reactor power upgrade, and an opportunity for further research on coupling the thermal hydraulics and neutronics codes.

The small modular reactor technology has received substantial endorsement from the U.S. Department of Energy (DOE) through the provision of funds to SMR vendors

[30]. The SMR funding seeks to facilitate the commercialization and deployment of small modular reactor (SMR) technologies through the SMR Licensing Technical Support program [30]. While the SMR technology has been designed to be safer and scalable, no SMRs have been deployed and have yet to achieve commercial success. A major factor in successful deployment is having a sustainable supply chain that is both reliable and able to respond to market demand effectively. A new supply chain model has been developed for Missouri to establish itself as a manufacturing hub for SMRs. The model is designed to be able to quantify key implementation obstacles and key growth areas.

1.1 OVERVIEW OF MISSOURI S&T RESEARCH REACTOR (MSTR)

The Missouri University of Science and Technology Reactor (MSTR) is a material testing reactor (MTR) (Figure 1.1). Several features of this type of reactor are:

- Light water moderation
- Natural convection cooling
- Open pool
- Plate-type fuel

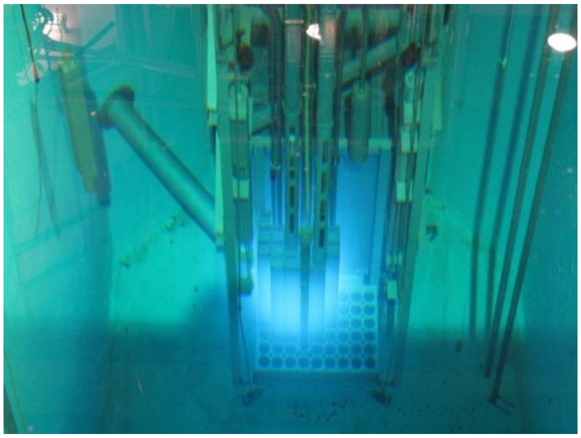

Figure 1.1. MSTR core submerged in the open pool

The next sections discuss the specific characteristics of the MSTR; the fuel type, core configuration, and reactor cooling system are the main consideration in the CFD modeling of the MSTR.

1.2 FUEL CHARACTERISTICS

The reactor core consists of fifteen fuel elements, four control rods, two irradiation fuel elements A standard fuel element has 18 curved fuel plates and a control rod fuel element consists of 10 curved fuel plates (Figure 1.2). The irradiation fuel element contains 9 fuel plates. In all fuel elements, the plates are encased in an aluminum sleeve, which allows water (coolant) to flow through the gaps between the plates to remove the heat generated from fission. The core cooling is by natural convection, and the heated pool water evaporates into the reactor space [3].

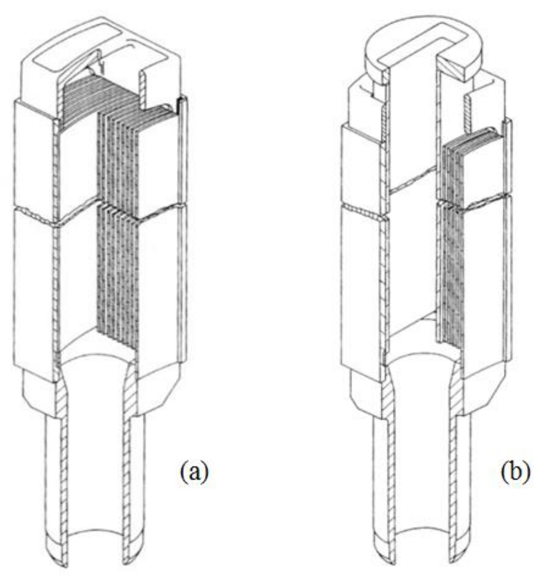

Figure 1.2 (a) Standard Fuel Element and (b) Control Rod Fuel Element

The configuration of the MSTR core in this study is 120W where the fuel elements and control rod fuel element are arranged in a 9 x 5 grid (Figure 1.3). There are a total of 310 fuel plates and approximately 295 channels through which coolant flows (Figure 1.4). The MSTR Safety Analysis Report (SAR) states that "The element holes, which have a 6.91 cm (2.42 in) diameter, pass through the grid plate to permit circulation of coolant through the core. The holes which do not hold an element are not plugged. Smaller auxiliary coolant holes, which have a 2.22 cm (0.875 in) diameter, are provided between the larger element holes to permit coolant flow between outside plates of the fueled elements in the interior of the core" (Figure 1.4) [3].

	1	2	3	4	5	6	7	8	9
A									
B						Source			
C					C4	F5	F1	F17	
D				F4	F8	F14	C1	F10	F2
E				F9	C3	F12	C2	F7	F3
F				CRT	F15	HC	F13	BRT	F6

Figure 1.3 MSTR 120W Core configuration (F:Fuel elements, C:Control rods, CRT/BRT: Cadmium/Bare Rabbit Tube, HC: Hot Cell)

The fuel material is made from low-enriched Uranium Silicide-Aluminium dispersion type (LEU U_3Si_2-Al) with 19.75% uranium enrichment and has a total heat generation area of about 30 m2. Uranium silicide is produced by melting together uranium metal and high-purity silicon. The LEU fuel consist of 19.75 ± 0.2 wt% 235U enrichment of the uranium metal, and the silicon content of the U3Si2 is $7.5^{+0.4}_{-0.1}\ wt\%$. While HEU fuels have 20 vol.% fuel or less in the meat, many LEU fuels have about 45 vol.% fuel. The Uranium density of U3Si2 corresponding to 45 vol.% is 5.1 g U/cm3. Table 1.1 lists the geometrical specifications of the MSTR curved plate fuels [3, 31].

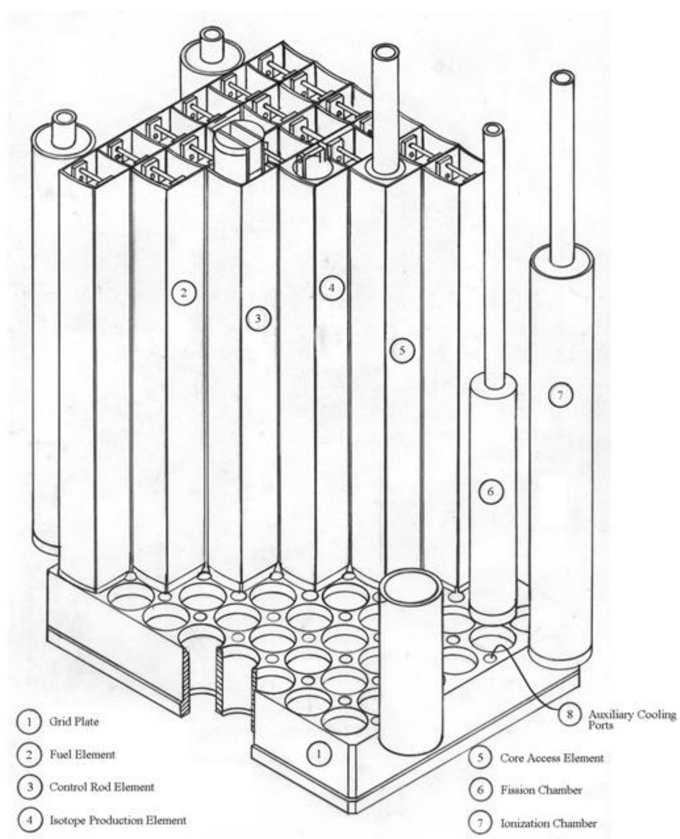

Figure 1.4 Fuel elements and control rod fuel element placement in the grid plate

A safety evaluation report, NUREG-1313 [32], described the fabrication process of the uranium silicide fuel which is made by substitution of UAl_x with U_3Si_2 in the fuel system. The NUREG report states that “The uranium-silicide fuel is produced by melting stoichiometric amounts of uranium and silicon, followed by comminution to produce a powder. The fuel powder is mixed with aluminum powder and formed under pressure into a powder metallurgical compact. The compact is placed in an aluminum picture frame and aluminum cover plates and hot and cold rolled to produce the fuel plate” [32].

The specifications and performance of the LEU fuel has been reported extensively in various technical reports [31, 32]. Previous testing and evaluation of the LEU U_3Si_2-Al fuel at Oak Ridge Research Reactor had investigated fuel performance by irradiation of miniplates and full-sized plates. The plates were subjected to heat generation rates up to 1.4 MW/m^2, and burnups of up to 98 percent of the uranium-235. Post-irradiation examination of the LEU fuels was performed by researchers from the Argonne National Laboratory. The analysis includes visual inspection and dimensional measurements, channel gap thickness measurements, gamma scans, plate thickness measurements, blister threshold temperature tests, metallography, and isotrophic burnup analyses. The LEU fuel is highly suitable for non-power reactors due to several features; having extremely stable swelling behavior of the U3Si2 phase dominated in all cases, small thickness changes (112µm in the regions of 98% burnup)and blister threshold temperature ≥ 550°C.

Table 1.1 Geometrical specifications of LEU U_3Si_2-Al fuel

Fuel Meat Thickness	0.51mm
Fuel Meat Width	61.0mm
Fuel Meat Length	610.0mm
Number of Plates per Standard Fuel Element	18
Cladding Thickness (Aluminum Alloy 6061)	0.381mm
Plate Thickness	1.27mm
Channel Gap Spacing	3.15mm

Fuel features to look out for during inspection include checking the cladding surface for corrosion effects; these are pitting corrosion and uniform-plate corrosion. A minimum cladding thickness is maintained so as to avoid release of fission products [31]. For the CFD simulation work, the channel gap spacing is assumed to be unchanged over the years of MSTR operation.

1.3 REACTOR SAFETY SYSTEM

Under the facility operating license issued by the U.S. Nuclear Regulatory Commission (NRC), the MSTR operates under specific limitations and fulfill equipment requirements for safe reactor operation and for handling abnormal situations. One of the safety boundaries instituted is to ensure that the integrity of the fuel cladding is maintained to guard against an uncontrolled release of fission products. One safety condition requires that fuel element cladding temperature shall not exceed 510°C (950°F). It is also established that reactor core has a negative moderator reactivity that provides an increase in excess reactivity when the reactor pool temperature lowers. Thus, maintaining the reactor pool temperature at a minimum of 15.5°C ensures that the excess reactivity will not significantly increase [3].

1.4 COOLING SYSTEM

The reactor coolant system consists of the reactor pool (primary cooling system), a demineralizer that keeps the water quality within limits, and a Nitrogen-16 (N-16) control system is in place to actively disperse the N-16 generated inside the reactor pool. These components keep the core cooled as well as to allow the reactor to be operated in a safe condition. The large pool is a heat sink for heat removal from the reactor by natural circulation, as well as a source of water for core cooling. The MSTR core cooling is achieved by natural convection. The open pool type reactor holds approximately 30,000 gallons of water [3]. Heat generated from the fuel elements are transferred to the pool water, and the heated water evaporates slowly into the reactor bay area i.e. evaporation is the ultimate heat dissipation mechanism. The reactor operates by natural convection to remove heat from the core. Differences in fluid density and body force (gravity) force the hot fluid to move upwards and cooler fluid to move down in the large pool establishing a circulation pattern driven by buoyancy.

1.5 SCOPE OF WORK

The cooling system plays a major role in maintaining safe reactor operations by removing the heat from the core. The knowledge of temperature variations and flow distribution in the reactor help reactor operators monitor and maintain thermal control at various operating power levels. The research goals of this work are to support power uprate plans for Missouri S&T's reactor as well as provide related modeling framework for the Small Modular Reactor Research and Education Consortium (SMRrec).

The objectives and contributions of this work are:

- Develop a model of the Missouri S&T Reactor to study the temperature and flow fields via computational fluid dynamics modelling and simulation
- Obtain porous parameter for modeling the reactor core as a porous media
- Provide thermal-fluid parameters for porous media model under normal operations
- Perform steady-state CFD simulations
- Validate the porous media model by pool water temperature measurement
- Provide tools to support future license renewal/ power uprate plans
- Develop a supply chain model for small modular reactors in the state of Missouri

1.6 DISSERTATION OUTLINE

This dissertation describes thermal-fluid modeling of the MSTR, and Small Modular Reactor supply chain modeling for SMRrec in the following paper sections. Three papers have been dedicated for CFD analysis on natural convection in the MSTR, and one paper on a new Missouri supply chain model for SMR technology.

2. LITERATURE SURVEY

Fluid dynamics and heat transfer are the core parts in any thermal hydraulic analysis of a nuclear reactor. The tools used for thermal hydraulic analysis uses state-of-the-art computer codes that can simulate both steady-state and transient behavior of the reactor. Studies have shown that computational fluid dynamics codes provided in-depth understanding of heat processes and fluid flow in complex geometries of nuclear reactors [10-12, 17-22].

This section consists of three subsections. The first discusses the previous work carried out on the Missouri University of Science and Technology Reactor (MSTR). These previous studies by former and current faculty, staff and students focused on obtaining the reactor's neutron flux profile through computer simulation as well as through experiments [4, 5, 23-25]. The studies also investigated the feedback effects of the reactor under normal and accident conditions. Part of the results from these work were used to provide support for the relicensing application of the MSTR (formerly known as University of Missouri-Rolla Reactor or UMRR) in 2004 and formed the main part of the authors master theses. The current license for the MSTR expires in 2029, however, there are initial plans for a reactor power uprate for MSTR. The work reported in this dissertation was carried out to complement the previous work described in section 2.1, and provide a framework in which the thermal-fluid parameters are obtained to support future expansion plans.

The second section discusses about thermal hydraulic analyses done on other research reactors in the United States. A typical thermal hydraulics code used is the Reactor Excursion and Leak Analysis Program (RELAP). This code is build-in with thermal hydraulic correlations that is used to study the neutronics and thermal hydraulics behavior of PWR and BWR under loss of coolant accident (LOCA). It combines both neutronics and thermal hydraulics relationships to profile reactor coolant and core behavior. For example, the code is used to predict how transients and postulated accidents influence reactor operations [73]. In the last decade, computational fluid dynamics (CFD) gained acceptance in the nuclear industry as a reliable computational tool for reactor safety analysis [1-2, 10-12]. An overview of the capability of CFD to study thermal hydraulic behavior in nuclear reactors is presented in section 2.2.

The third section is an overview of the current status of the SMR technology, and the background for an SMR industry. Several nuclear industry issues, including the case of adopting the Westinghouse SMR are discussed in Paper IV in the publication sections.

2.1 PREVIOUS WORK RELATED TO MISSOURI S&T REACTOR

The Missouri University of Science and Technology Reactor (MSTR), formerly known as University of Missouri-Rolla Reactor (UMRR), have been in existence since 1961 [3]. It has been mainly used as a teaching and training reactor; experimental facilities are also used by students and faculty to carry out research projects. The objectives of the previous studies on MSTR have been directed towards characterizing the core through computer simulation as well as experimental work. The simulation work obtained neutron fluxes, reactor parameters, reactor transients, hot channel factor and burnup calculations. Simulation and experimental work with regards to coolant/moderator temperature distribution is limited and can be explored in detail for this research. Such study could assist in having accurate neutronics cross-sections and thermal feedback estimates.

Prior to 1992, the MSTR was fueled with high enriched uranium (HEU) and had a 100W core configuration. The HEU fuel was U_3O_8-Al enriched with 90% ^{235}U. The 100W core consisted of 14 fuel elements, 4 control elements and 1 half element; a standard fuel element has 10 plates, a control element has 6 plates, and a half element has 5 fuel plates [3].

Corvington (1989) performed neutronics calculations for the fuel conversion from HEU to LEU [4]. In his study, he used 2DB-UM and LEOPARD codes to predict several reactor safety parameters such as power peaking factor, moderator and void coefficient as well as core multiplication factor and neutron flux profile for both HEU and LEU cores. The 2DB-UM is a two-dimensional neutron diffusion code that solves multigroup diffusion equation using cell-centered finite difference equations. LEOPARD is a code that calculates neutron spectrum and group constants for light water reactors; it utilizes two-energy or four-energy group cross section sets [13]. The maximum power peaking factor occurred at the control element C3. Three components of the power peaking factor (radial, elemental and axial) for the HEU and LEU cores were calculated gave a total power peaking factor of 2.00 and 2.22 respectively. The allowed power peaking limits stated in the UMRR Safety Analysis Report (SAR) is between 3.0 and 4.0. Temperature coefficient of the moderator is the sum of two component coefficients, the moderator density coefficient and moderator temperature coefficient. The predicted moderator temperature coefficient for the LEU was 40% smaller than for HEU fuel, however, the desired negative reactivity feedback was achieved. Details of the calculated and measured thermal flux profile are discussed in Corvington's thesis. The findings by Corvington formed part of the SAR document when the MSTR core changed to low enriched uranium (LEU) fuel and core configuration to 101W.

An accident analysis of the UMRR was performed by Carroll (2004) using PARET and CONVEC codes [5]. The Program for the Analysis of Reactor Transients or PARET was used to predict the behavior of reactor under accident conditions and used to investigate reactor reactivity transients. Power profile and temperature values for coolant, fuel and cladding were obtained. Three accident conditions for UMRR were chosen:

(i) Insertion of excess reactivity. This was initiated by the accidental placement of a fuel element near the reactor core with a worth $1.90 or 1.50%Δk/k, reactor assumed to be operating at full power at accident initiation @ 200kWt, 400kWt and 500kWt for each case moderator inlet temperature @ 70F, 80F and 90F.

(ii) Using CONVEC, the loss of coolant accident condition for UMRR was investigated. CONVEC is a computer program designed to obtain solutions of transient two-dimensional incompressible fluid flow problems as well as energy equations [14]. The program is based on finite element method, and is able to predict fuel, clad and coolant temperatures at a loss of coolant accident condition. Caroll (2004) studied the behavior of the UMRR during a situation where there was a rapid loss of coolant and the reactor is operating at a power with only air cooling

(iii) Startup accident where a reactivity insertion occurs due to withdrawal of control elements at a rate of $0.36 per second. The peak power and temperatures of coolant, clad and fuel were calculated at 200kW and 400kW power levels with varying moderator inlet temperatures 70F, 80F and 90F. The

> results show that there is a cyclic power peaking due to the transient occurred after 3 seconds and stabilizes due to negative feedback effects of the reactor. The cladding temperatures remain well below the melting point of 588C. Fuel integrity is maintained and no fission product release occurs.

Kulage (2010) performed calculations to estimate the neutron flux spectrum of the MSTR using the SAND-II program and MCNP codes [23]. The spectrum was also experimentally measured using foil flux monitors. The thermal, intermediate and fast neutron power fluxes were estimated to be 2.94E+12±1.9E+10, 1.86E+12±3.7E+10 and 2.65E+12±3.0E+3 neutrons per square centimeter per second [23].

Richardson (2012) in his thesis reported a cosine-shaped neutron flux profile for MSTR. The neutron cross-sections were defined at a core-averaged temperature [24]. This conservative approach provided an MCNP model that is relatively close to the flux profile obtained from experiments. Subsequently, to account for the axial temperature variation, two different temperatures used for the top and bottom half of the core to represent axial temperature profile. This approach improved results from the MCNP model by 2.7%. It was suggested in his thesis that using the actual temperature variation in the core could provide a better MCNP model of the MSTR. This suggestion was taken up in this dissertation whereby a thermal-fluid model of the hottest channel in the MSTR was developed. The temperature profile was reported in the manuscripts published in the ANS meeting and conference proceedings found in the paper publication sections [24].

Finally, O'Bryant (2012) determined the hot channel factor for both a clean core and burnup corrected core [25]. It was found that the hottest channel is located between the 6th and 7th fuel plates of control rod number 1 (refer Figure 1.3). The ratio of maximum to average value of energy deposition is 1.85 and 1.71 for clean core and burnup corrected core respectively. The corrected model revealed that the hottest channel remained at the same location that was determined for the clean core model.

2.2 PREVIOUS RELATED WORK ON OTHER REACTORS

Several studies have been reported in the literature on thermal hydraulic analysis of research and test reactors. For example, Yan (2011) investigated the 20MW Australian Replacement Research Reactor using the computational fluid dynamics code, ANSYS Fluent [18]. In his study, the reactor core was approximated as a porous media. He found the characteristic equation to model the core as a porous media by examining forced convection in both laminar and turbulent flow regimes. He proposed a modified k-ϵ turbulence model for porous media in the FLUENT code using user-defined functions. This finding was applied to study advanced Gen IV reactors, whereby the general porous media model in FLUENT was modified to define turbulence in porous media. The neutronic and CFD codes, RELAP-3D and Fluent, were coupled and used to simulate the primary coolant system in the Gas Turbine-Modular Helium Reactor [18].

Yoon & Park (2008) used the CFD code, CFX, to improve the 3-D CFD model that predicts the temperature distributions of the moderator inside the Calandria vessel [20]. The matrix of Calandria tubes located in the core region required a large number of computational cells to model. The porous media approach for the core region was made to overcome computational limitations. Buoyancy force was modeled as a source term in the momentum equations. Using Boussinesq approximation, density was assumed to be a linear function of the temperature. Subsequently, they analyzed the moderator transient for the 35% Reactor Inlet Header (RIH) break without Emergency Core Cooling (ECC) injection so as to determine whether the fuel channel integrity is maintained. They successfully developed a CFD moderator analysis model for Canada deuterium uranium (CANDU) reactors. CFX was used to study steady-state moderator circulation under operating conditions and the local moderator subcooling during a LOCA transient [20].

J. Chang (2005) developed a CFD model for Pennsylvania State's Breazeale Nuclear Reactor using 3D FLOW. The analysis of temperature behavior during steady state and pulsing show cooling of the core was by both axial and strong cross-flow due to thermal expansion of the coolant [21]. In addition to coolant flow modeling, a stand-alone fuel rod model predicted temperature distribution in the fuel rod and thermal response during both steady-state and pulsing operation. The predictions by the models were shown to correlate to temperature and velocity data.

A study by Krepper (2002) for flows under natural convection in large pools revealed that is possible to calculate temperature oscillations caused by heat plumes [22]. It was reported that the heating up process in a long and large horizontal cylinder (6m x 2m dia.) can be simulated in a qualitative manner. His calculation provided insight into the effects of temperature stratification in an emergency condenser. Stratification increases pressure in the containment of a nuclear reactor and therefore an alternative arrangement of guide plates was suggested. This arrangement could establish natural circulation by chimney effect, hence, act as a passive measure.

Tung et.al. (2014) performed CFD analysis on a prismatic gas-cooled very high temperature reactor (VHTR). Their findings showed the heat flow behavior of a prismatic very high temperature reactor (VTHR) during a loss of flow accident (LOFA). Their modeling strategies utilized the symmetry of the core as well as reduced heights in modeling plenums to capture effects of natural circulation [19].

In general, the cross flow for MSTR is expected to be minimum because each channel is separated by solid walls. However, the gap between two fuel elements is subjected to the same phenomena as reported for a power reactor, for example at high power levels the flow difference between the fuel elements could be large enough to initiate turbulent interchange, likewise significant power difference between fuel elements or imperfect alignment of fuel element lead to pressure difference and consequently diversion cross flow. The combined effect can be lumped in the parameters of a porous media model and the detailed thermal hydraulic analysis is avoided.

2.3 SMALL MODULAR REACTOR TECHNOLOGY

Small Modular Reactors (SMRs) is an emerging class of nuclear reactors that are being developed to meet the world’s energy demands [33]. A nuclear reactor is considered to be under the category of SMR if it is having electrical output of less than 300MWe [33]. A typical large nuclear power plant is rated at 1000MWe or more. There are many types of modular reactors being developed around the world, and the designs that have emerged utilize diverse reactor technologies. SMR designs can be grouped according to the way the reactor core is being cooled; the primary coolant can be ordinary water, heavy water, gas (Helium), and liquid metal (molten salt). The SMR categories are listed below [33]:

- Light Water Reactor (LWR)
- Heavy Water Reactor (HWR)
- Gas Cooled Reactor (GCR)
- Liquid Metal Cooled Reactor(LMCR)

There are 131 small and medium sized reactors in operation in 26 countries with a total capacity of 59GWe [33]. Out of all these reactors, there are 32 different designs reported in the International Atomic Energy Agency's (IAEA's) SMR design status report [33]. The US has four LWR-type, two GCR-type, and two LMCR-type reactor designs [30, 33].

Table 2.1 SMR Designs in the US [30, 33]

Small Modular Reactor (SMR)	**Containment Dimensions**	**Number of fuel assemblies**	**Plant Design Life (Years)**	**Passive Heat Removal**
Westinghouse SMR (800MWt/225MWe)	Height 89ft Dia. 32ft	89 (17x17 PWR design)	-	7 days
NuScale (160MWt/45MWe)	Height 65ft Dia. 9ft	37 (17x17 PWR design)	60	3 days
Generation mPower (B&W) (530MWt/180MWe)	-	69 (17x17 PWR design)	60	14 days
Holtec Inherently Safe Modular Underground Reactor (HI-SMUR), SMR-160 (160MWe)	Height 100ft Dia. 50ft	32 (17x17 PWR design)	80	Indefinite cooling

*Pressurized Water Reactor (PWR)

The four different light water SMRs in the US are developed by Westinghouse Electric LLC, NuScale Power Inc, Babcock & Wilcox, and Holtec International respectively (Table 2.1). Typical units for measuring the output of a power plant are MWe and MWt. Megawatts electric (MWe) refers to MW of electrical output, and megawatts thermal (MWt) quantifies the MW of thermal output. For example, the mPower (Babcock & Wilcox, USA) SMR has a design thermal capacity of 530MWt and an electrical capacity of 180MWe.

Reliance on natural convection for emergency and in some cases normal operation for heat removal is a common feature for all of these designs [33]. The SMR is designed to remove excess heat by natural convection in the event of an emergency without human intervention or the use of active heat removal systems such as pumps. In addition, the inherent passive safety design of the SMRs can absorb powerful earthquakes, tsunami, and tornadoes without compromising the core's integrity. All the major US SMR designs are listed in Table 2.1. All four are integral pressurized water reactor (iPWR) designs; these designs are compact versions of the regular PWRs [30, 33-35]. Compared to the large NPP design, the iPWR design is simpler, and combines the entire reactor and the nuclear steam supply system into one reactor vessel. The reactor vessel is located underground and this position protects the reactor from external threats from airplane crash and projectiles.

The development of an efficient SMR supply chain is paramount to the timely and successful construction of small modular reactors. It is also important to understand the processes involved when a customer places an order of an SMR, and who are involved at each phase. Figure 2.1 shows the stages of an SMR customer order. The major stakeholders in a SMR supply chain are the suppliers, SMR vendors, government, and the customer. The suppliers are categorized according to components and subcomponents of a new nuclear reactor and the categories under which their products fall under. In the SMR supply chain map, the nuclear power plant components are divided into four major groups:

- Nuclear Island: All nuclear grade or safety-related products and components
- Turbine Island: All parts/components related to heat-to-electricity conversion
- Balance of Plants: All parts/components related to cooling system, electrical switchyard, etc.
- Site Development and Construction: Site preparation, construction equipment, supplies and support

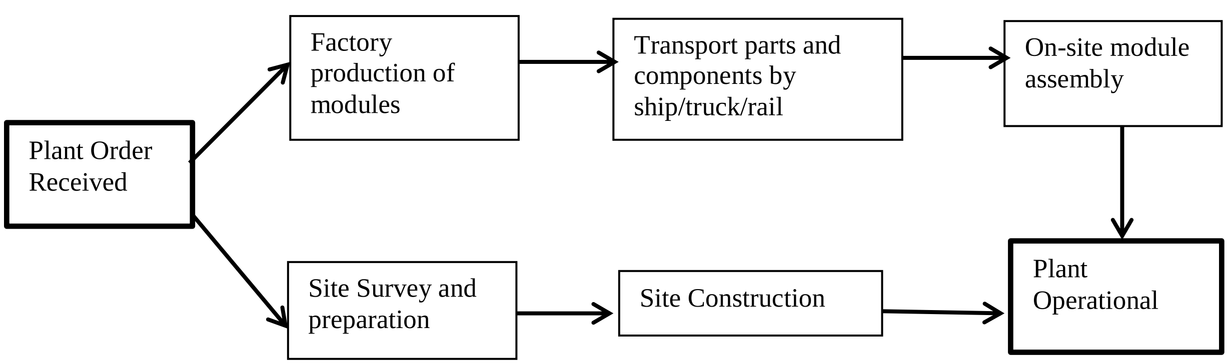

Figure 2.1 Stages of an order for an SMR

I. CFD MODELING OF A COOLANT CHANNEL FOR MISSOURI S&T REACTOR

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1. Introduction

An accurate and comprehensive nuclear core modeling of the Missouri University of Science and Technology Research Reactor (MSTR) utilizes both thermal hydraulics and neutronics codes. Previous Monte Carlo N-Particle (MCNP) simulation work performed by O'Bryant (2012) had obtained a core map, estimating the energy deposition in the fuel assembly, as a result of the gamma and neutron interactions within the fuel plates [1]. Temperature induces two types of feedback to the neutronics. Firstly, the change in the density of coolant (and hence the moderator) can impact the neutron slowing down density and therefore the energy spectrum. Secondly, the neutron cross section may show some temperature dependence. The purpose of this study is to determine the need of these corrections for accurate modeling of the core. In this initial study, the focus is only on the second impact that is the need for modifying neutron cross sections. From very fine 3D computational cells, temperature and flow profiles are resolved from FLUENT simulations. These are used to determine the refinement needs of the MCNP model, and enable a detailed hot channel prediction for the current core configuration.

2. Description of Work

2.1 MSTR Geometry

The MSTR core cooling is currently achieved by natural convection. The open pool type reactor holds approximately 30,000 gallons of water and evaporation is the ultimate heat dissipation mechanism. The reactor core consists of 15 fuel elements and four control rods. A standard fuel element has 18 curved fuel plates and a half element consists of 10 curved fuel plates (Figure 1). The plates are encased in an aluminum sleeve, which allows water (coolant) to flow through the gaps between the plates to remove the heat generated from fission. There are a total of 310 fuel plates and approximately 295 channels through which coolant flows. The fuel material is made from uranium silicide, U_3Si_2Al with 19.75% uranium enrichment and has total heat generation area of about 23 m^2. Fuel plates consist of U_3Si_2-Al fuel "meat" sandwiched in aluminum clad. The fuel meat dimensions are approximately 0.05 cm x 6.10 cm x 60.96 cm (0.02 in x 2.4 in x 24 in). The cladding is a layer of aluminum alloy 6061 which is 0.038 cm (0.015 in) thick. The overall plate thickness is about 0.13 cm (0.05 in). The gap between two fuel plates is approximately 0.315cm [2].

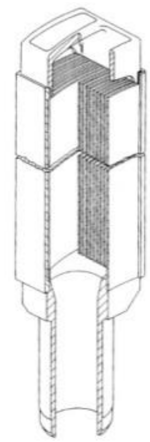

Figure 1. Cross sectional view of a fuel element with 18 fuel plates

2.2 Computer Simulation

The scope of this work is to simulate coolant flow into a narrow channel that arises from convective heating from two curved fuel plates. A computer aided design (CAD) model of the fuel plates were made using ANSYS DesignModeler [3]. The mesh generator chosen for this study is ANSYS Meshing and had utilized sizing to generate finer mesh at the face boundaries [3]. At 200kW, the estimated Rayleigh number of the setup is in the range of 10^4 [4]. For the simulation of coolant flow through the channel, a laminar flow was assumed. The flow solver ANSYS Fluent 14.0 was used to study the flow and convection process in detail [3]. Preliminary CFD calculations were performed using Intel Core2 Duo CPU E7200 @ 2.53GHz that operates under a 64bit operating system with 4GB RAM. Essentially, FLUENT solves the governing integral equations for the conservation of mass, momentum, and energy.

The flow and heating processes were modeled in a computational domain representing 1.2m x 0.4m x 0.5m volume. The grid consisted of about 500,000 cells and 90,000 nodes. This grid size was determined from a grid study that varied the fineness of the grid. Increasing cells over 500,000 gave little improvement in the results (Figure 2). With the plate length (L) to channel width (W) ratio of about 200 (L/W), the cells have a skewness average of less than 0.3 and quality of less than 0.85. The coolant temperature 294K (21°C) was used, which corresponds to the regulated coolant temperature at the reactor. Boundary conditions at the fuel plate surfaces were set at 10,000 W/m^2 and the other bounding walls were adiabatic. The heat flux of 10,000 W/m^2 was based on a 1.15 hot channel factor and an average heat flux of approximately 8,700 W/m^2. This initial study focused on identifying the need for neutron cross section correction. The ΔT values were required to determine the potential need for cross section correction. The use of cosine shaped heat flux variation in the axial direction is more realistic and would increase the fidelity of modeling. Small velocity 1E-6 ms^{-1} was given at the lower boundary and the top boundary was pressure outlet at atmospheric pressure. Coupled flow and heat transfer calculations were implemented to capture the convection phenomena.

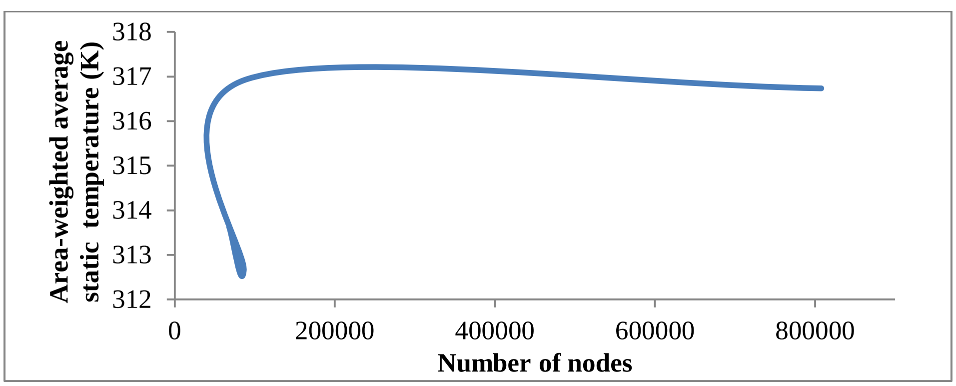

Figure 2. Independent grid study using heat flux 8700 W/m^2

3. Results

To investigate the coolant flow, three axial planes were chosen on y-axis and three planes on z-axis. On y-axis, the three planes show the point of entry, mid-point and point of exit of the coolant. On the z-axis, the three cross sections were made at the left, middle and right sections of the channels. Figure 3 shows a partial temperature profile of the channel. It can be seen that the heating from the fuel plates induces an upward coolant flow, which is represented by the heat plumes that rises up to the top boundary of the domain.

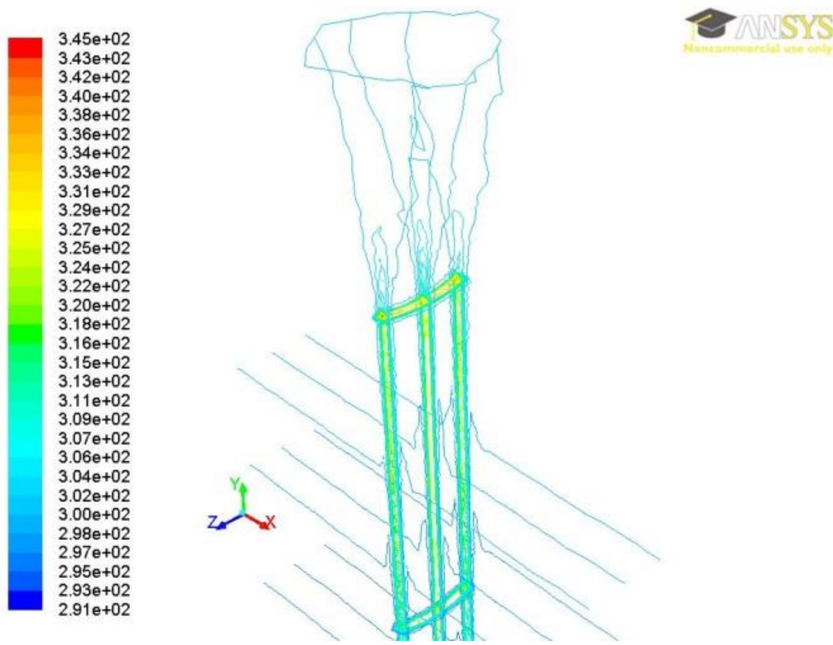

Figure 3 Temperature distribution along the length of the channel

Three line axial locations were chosen to show the heat flow progression. These lines are 1mm apart and are located within the coolant flow path. Gradual heating of the coolant is shown as it moves up the plate length. The coolant temperature at the entry point (bottom part of channel) is 294K (21°C). The maximum temperature at the exit point (top part of the channel) is ~340K (67°C). The results show a temperature rise of 46K between the entry point and the exit point of the channel. The sudden temperature variation at the coolant exit is under investigation.

The temperature on the fuel plate surfaces were calculated using area-weighted surface integral and were found to be 329K (55.85°C) and 327K (54°C) for the inner and outer plate surfaces respectively. These predicted temperatures from FLUENT are close to the expected average value of 325K (52°C), which is calculated based on published correlations [4]. Figure 3. Partial temperature profiles at the top half of the fuel plates. Three cross-section planes are shown. Temperatures are in Kelvin.

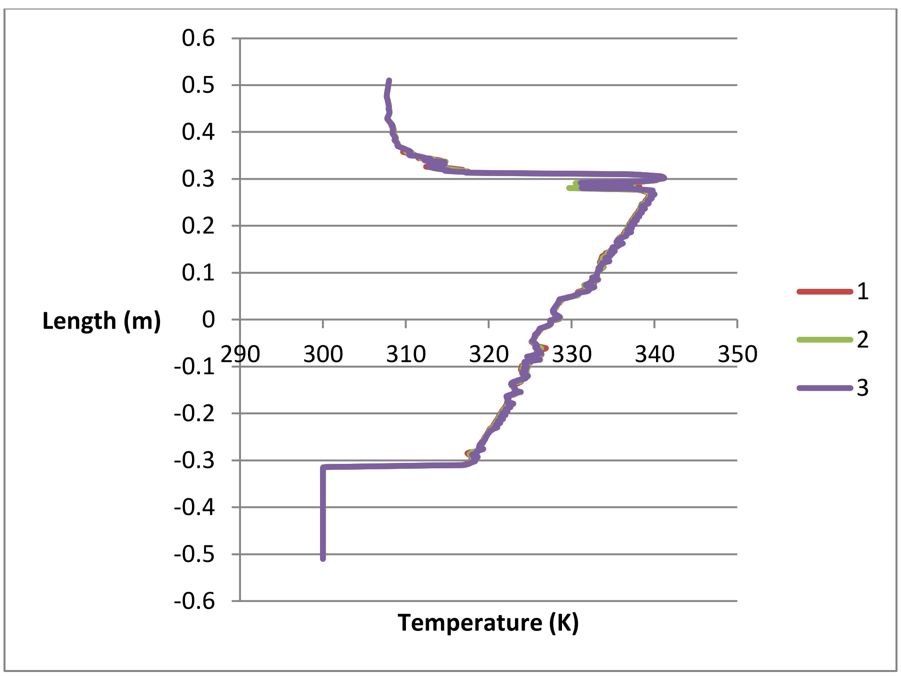

Figure 4. Temperature profile along the length of the plate. Results are shown for three line locations (1, 2 and 3) within the channel

As there is no physical temperature measurement made on the fuel plate itself, these estimates provide a way to estimate safety limits on reactor operation. The results from FLUENT will provide insight into the safety analysis of any future power upgrade efforts.

Figure 5 shows the velocity profile along the full length of the fuel plate. Zero velocity was recorded on the plate surfaces and this corresponds to no slip condition. Low flow is recorded at the entry point (bottom part of the channel) and proceeds to gain momentum as it gets heated up. Velocities between 0.03 ms^{-1} and 0.05 ms^{-1} were recorded above the top mid-section of the channel. This allows for slow mixing at the top of a large domain and would probably not affect the mechanical integrity of the fuel plates. From the velocity and temperature profiles along with the associated density change, a Froude number of approximately one was obtained. This indicates the effect of buoyance driven mixing in the pool water.

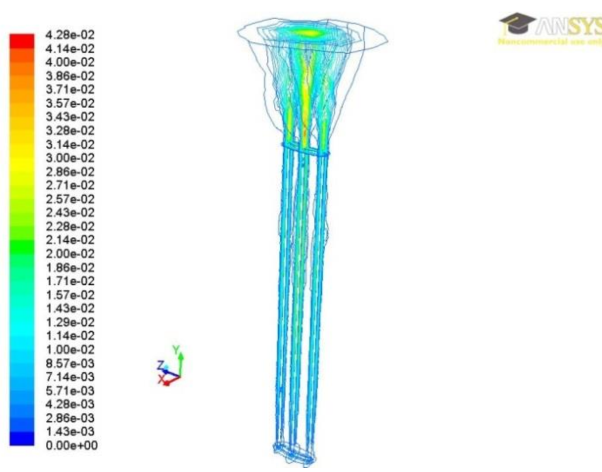

Figure 5. Velocity profiles along the fuel plates. Three cross-sections are shown. Velocities are in meter per second

4. Summary

Simulations were carried out to determine the need for temperature corrections to accurately model the MSTR core. This preliminary study using uniform heat flux had obtained temperature and velocity profiles of the coolant flow into a narrow channel that arises from convective heating from two curved fuel plates. The result of the simulations indicated that there is no need for neutron cross section correction for MCNP simulations. Less than 50K temperature is sufficiently small to ignore any impact on the neutronics. The water density change of approximately 2% may require some correction in the model.

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II. A PARALLEL PLATE MODEL USING A POROUS MEDIA APPROACH

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Abstract

For reactor operators, maintaining safe operation and having control over operations under normal and accident conditions is paramount over the lifetime of the reactor. Safety analysis provides a systematic way to study flow stability boundaries, temperature limits, transients that cause core damage as well as the overall behavior of a nuclear core. This analysis usually employs various computational codes, and is required when there is a major modification to the core, including for reactor power upgrade. Recent progress in computational fluid dynamics codes extends its capabilities to resolving local temperature and flow fields in various types of nuclear cores. The Missouri University of Science and Technology Reactor (MSTR) is looking into increasing its reactor operational power level, and is seeking an update to the current computational tools to support the power uprate plans.

1. Introduction

The Missouri University of Science and Technology Reactor (MSTR) is a 200kW research reactor. The core has a total of fifteen fuel assemblies and four control rods that can generate up to a maximum total flux of $4.36 \times 10^{12} \pm 2.84 \times 10^{11}$ neutrons/cm^2/s [1]

[2]. The reactor coolant system consists of the reactor pool (primary cooling system), a demineralizer that keeps the water quality within limits, and a Nitrogen-16 (N-16) control system is in place to actively disperse the N-16 generated inside the reactor pool [2]. These components work in tandem to keep the core cooled as well as to allow the reactor to be operated in a safe condition. Cooling of the core is achieved through natural convection.

In a paper by Richardson et. al. (2012), he obtained a cosine-shaped neutron flux profile for MSTR [1]. The neutron cross-sections were defined at a core-averaged temperature. This conservative approach provided an MCNP model that is relatively close to the flux profile obtained from experiments. Subsequently, to account for the axial temperature variation, two temperatures used for the top and bottom half respectively improved the MCNP model by 2.7% [1]. The non-uniform heat flux present along the axial length of the fuel plate presented the need to modify neutron cross sections so as to obtain high-fidelity neutronics as well as thermal hydraulic core model.

Part of the problem of modeling and simulation of the thermal hydraulics of large systems like a nuclear reactor is the sheer size of reactors. The MSTR pool is rectangular-shaped, and is approximately 5.79 m (19 ft) long, 2.74 m (9 ft) wide and 8.23 m (27 ft) deep. It houses the reactor, a beam port, and a thermal column. It contains about 113.56 kiloliters (30,000 gallons) of demineralized water. Modelling the convective behavior of such large pool requires enormous computer memory, and takes up time to perform the calculations. One of the strategies that can be used to counter these limitations is adopting a porous media approach to model the core, therefore minimizing the number of computational cells that are needed to model the whole reactor.

The MSTR core can be approximated as a porous media due to its homogenous configuration; made up of a uniform arrangement of fuel plates interspaced with cooling channels. It becomes an array of pores where coolant (water) flows through. In this work, we opted to model the “unit cell” of the MSTR core: consisting of two fuel plates and a coolant channel. The reactor core consists of an array of parallel-plates and channels. The pressure difference across each channel is approximately the same since there is no cross-flow effect. This allows for the same flow-rate in each channel. This work aims to obtain an accurate prediction of the pressure drop at various operational power levels, and obtain a characteristic equation of the core as a porous media.

2. Fuel Plate Geometry

The MSTR core has a total of 310 fuel plates and approximately 295 channels through which the coolant flows. The fuel material is made from uranium silicide, U_3Si_2Al with 19.75% uranium enrichment and has total heat generation area of about 30 m^2. Fuel plates consist of U_3Si_2-Al fuel “meat” sandwiched in aluminum clad. The fuel meat dimensions are approximately 0.05 cm x 6.10 cm x 60.96 cm (0.02 in x 2.4 in x 24 in) [2]. The cladding is a layer of aluminum alloy 6061 which is 0.038 cm (0.015 in) thick. The overall plate thickness is about 0.13 cm (0.05 in). The gap between two fuel plates is approximately 0.315cm [2].

3. Porous Media

Porosity is an attribute of a medium whereby voids are present within that solid media. Any system that consists of solids and interconnected voids could be described as porous at varying degrees of permeability. A nuclear core is made up of a fuel assembly, control rods, and support structures; the geometry of which affects the flow field at the core. Heat generated in the core is removed through flow channels present in the fuel assembly into the surrounding coolant fluid. A nuclear core can be assumed to behave like a porous medium due to the array of solid fuel and the flow of coolant through its many channels. It is a network of solids and fluids. An investigation of a nuclear core by porous media approach would show the macroscopic behavior of the core. Parameters associated with the region are volume-averaged thereby allowing analysis of large regions. The region is representative of the collective flow behavior through the coolant channels. In effect, we apply the macroscopic transport theory on the large system by making volume analysis at the scale of one coolant channel.

Porous media may be studied experimentally or through computational models. In an experimental setup, the porous media is subjected to flow conditions that produces pressure gradients to mimic actual physical processes. Similarly, computer models aims to reproduce physical processes that are less expensive to carry out and avoid costly experimental setups for various conditions to be investigated [3].

An important property of porous media is permeability, which is a measure of a porous medium's capability of transferring fluid throughout the pore space within the medium. Pressure losses in a porous medium due to viscous effects are described by Darcy Law [3]. Equation (2) describes a linear relationship between pressure gradient and filtration velocity derived from Darcy's Law [3].

$$-\frac{dP}{dx} = \frac{\mu}{\kappa} u_f \quad \text{for Re} < 1 \qquad (2)$$

where dP/dx (N/m^3) is the pressure gradient along the x axis or length of the column, μ ($N\text{-}s/m^2$) is the fluid viscosity, u_f (m/s) is the filtration velocity or ratio of total pore space volume flow rate (m^3/s) to total pore space area (m^2), and κ(m^2) is the average medium permeability. Permeability is also expressed in Darcy (D) or milliDarcy (mD) units where 1 D = $9.86 \times 10^{-13} m^2$.

For higher flow rates (Re > 1) in porous media, the pressure gradient begins to deviate from a linear relationship. A quadratic term is added to Darcy's law to describe this deviation as seen in equation (3).

$$-\frac{dP}{dx} = \frac{\mu}{\kappa} u_f + \rho \beta u_f^2 \quad \text{for Re} > 1 \qquad (3)$$

This is known as Forchheimer's equation and β (m^{-1}) is often referred to as Forchheimer's coefficient. The quadratic term relates pressure losses within a porous media to inertial dissipation.

4. CFD Model

The commercial computational fluid dynamics package, STAR-CCM+, was used to model the fuel plates as well as simulate the convective heat transfer from the fuel plates into the surrounding coolant water [5]. Essentially, STAR-CCM+ is a flow solver for the governing integral conservation equations of mass, momentum and energy [5]. Polyhedral and embedded thin mesher models were used for mesh continua, with thin solid thickness defined as 3.15mm corresponding to the gap between fuel plates. At

200kW, the estimated Rayleigh number of the setup is in the range of 10^{11}[6]. The CFD calculations were performed using Intel i7 @ 2.2GHz with 8GB RAM.

The flow and heating processes were modeled in a computational domain representing 0.85m x 0.01205m x 0.082m volume. The plate length (L) to channel width (W) ratio is approximately 200 (L/W). The polyhedral and embedded thin mesher obtained good resolution of the heat flow process. A coolant temperature of 294K (21°C) was used, which corresponds to the regulated coolant temperature at the reactor. Boundary conditions applied at the fuel plate surfaces were non-uniform heat flux and follow the flux profile of the active length of the core (Figures 1 and 2). The side walls bounding the fuel plates were taken to be symmetric planes. The lower and upper boundaries were kept at the constant temperature of 294K. Coupled flow and heat transfer calculations were implemented to capture the convection phenomena.

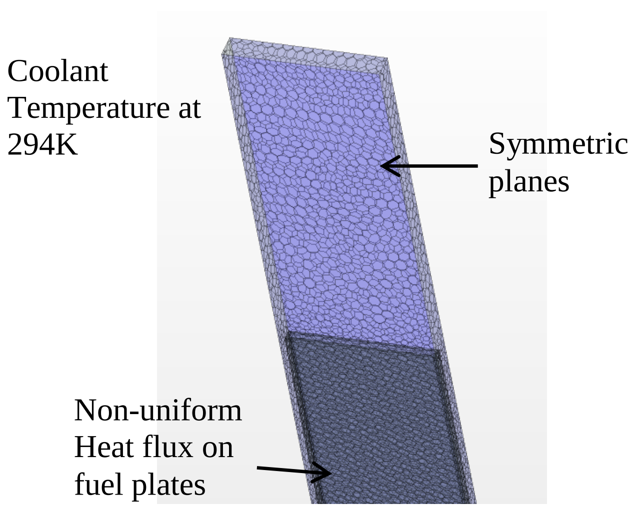

Figure 1. Boundary conditions on fuel plates (top section)

The effect of the porous medium on the flow was defined using lumped parameters. The parameters are typically taken to be resistance coefficients for a source term in the momentum equation. The inertial and viscous coefficients are required for the porous source term in the momentum equation. The macroscopic effect of the porous medium on the overall fluid flow is of interest, not the details of the internal flow.

The porous source term appears in the momentum equations of the coupled and segregated flow solvers

$$f_p = -P.v \qquad (4)$$

where P is the porous resistance tensor [5]. Porous resistance tensor is given by

$$P = P_v + P_i|v| \qquad (5)$$

where

P_v is the viscous (linear) resistance tensors

P_i is the inertial (quadratic resistance tensors

In the porous region, the theoretical pressure drop per unit length can be determined using the equation below [5].

$$\frac{\Delta P}{L} = -(P_i|v| + P_v)v \qquad (6)$$

The goal of this study was to simulate coolant flow into a narrow channel that arises from convective heating from two fuel plates, and to obtain the associated pressure drop in the channel. A curve fit of $\frac{\Delta P}{L}$ versus v^2 and v was applied to equation 6. From the pressure gradient, the viscous and resistance tensors were predicted from a characteristic equation on a curve fit on a plot pressure drop per length versus average channel velocity. This geometry (two fuel plates and a channel) was taken as the smallest unit of the core region.

The pressure drop (between the channel entrance and channel exit) and volume-averaged velocity was predicted as well as the maximum exit velocity. From the two variables, the equation that describes their relations was obtained. The equation was correlated to the viscous resistance factor and the inertial resistance factor. The factors formed the basis to model the fuel assembly using porous media approach. Polynomial curve fit for Forchheimer's equation was obtained for the two plates-one channel model.

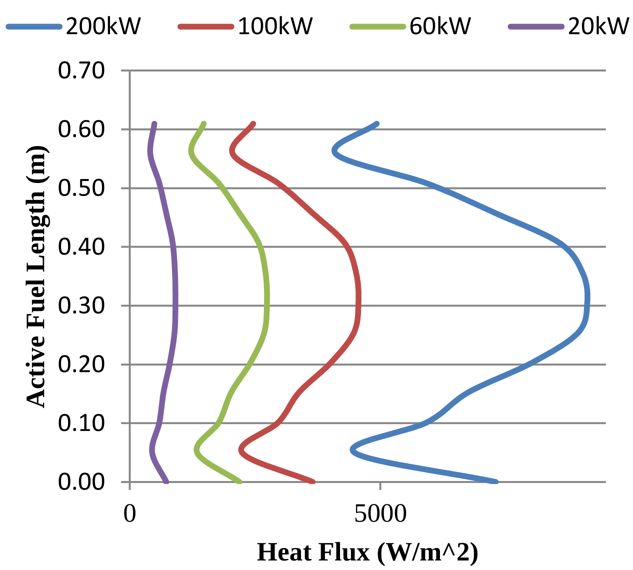

Figure 2. Heat flux profile at various nuclear power level

4.1 Grid Study

A mesh consisting of 19714 cells was chosen to calculate all reported values in this study. This mesh size was determined from a study that varied the fineness of the grid. Increasing cells over 20,000 gave little improvement in the calculations for exit temperature. The chosen mesh gave 20% computational time savings compared to the mesh with 72018 cells (Figure 3).

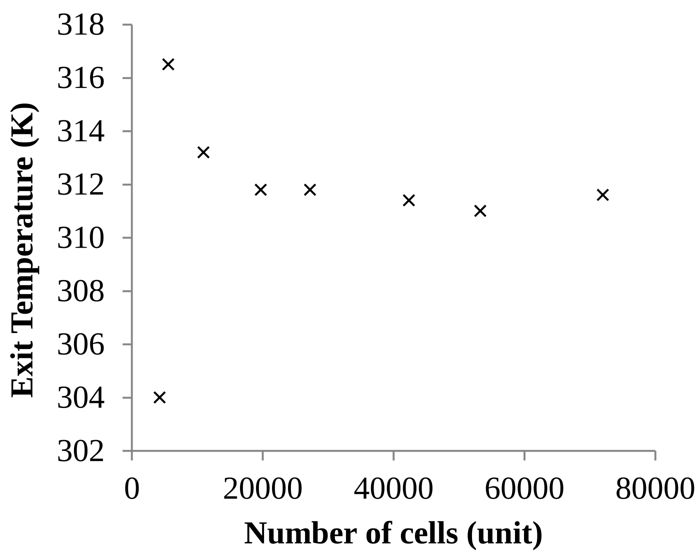

Figure 3. Grid study to check for independence

5. Results and Discussions

The number of pore space was determined using the volume porosity as described by Todreas & Kazimi (1990). Volume porosity was calculated to be 0.7027, and was used on the porous model [3].

$$\gamma_V \equiv \frac{V_f}{V_T} = \frac{Fluid\ volume}{Total\ volume} \qquad (7)$$

Each reactor power level gave an associated heat flux profile. When applied on the plate surface, each cosine-shaped heat flux predicted a correponding pressure difference between the bottom and top of the channel (Figure 4). Hydrostatic pressure effect was not included in the pressure calculation.

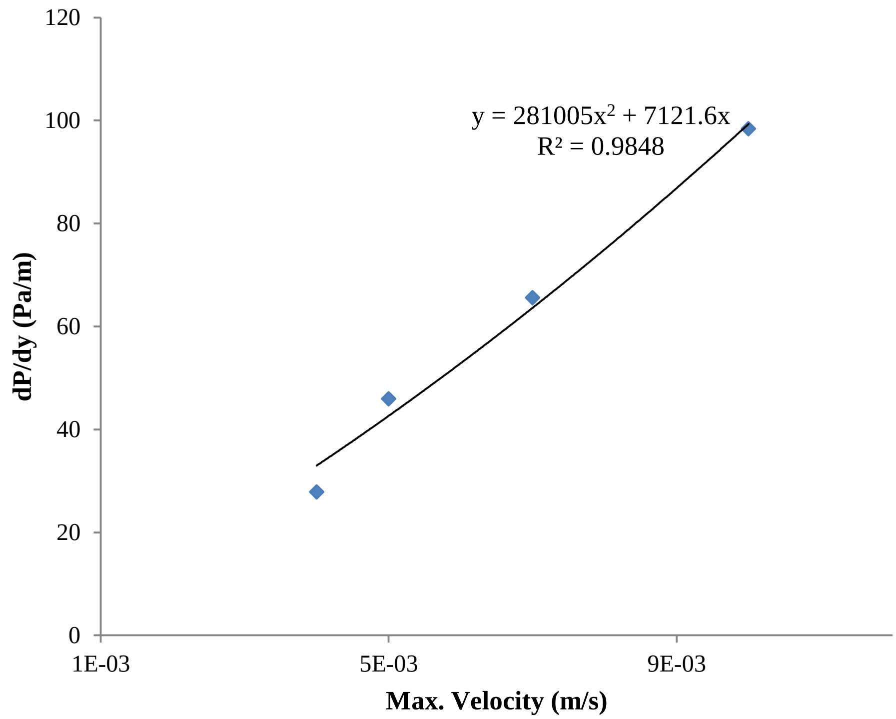

Figure 4. Pressure drop per length with maximum velocity at channel exit

The results predicted the viscous resistance tensor, P_v and the inertial resistance tensors, P_i to be 281005 kg/m^4 and 7121.6 kg/m^3 respectively. Due to the nature of free convection, the inertial resistance tensor was a magnitude higher than the viscous resistance. These values were then used in a (replacement) porous model for the two-plates, one channel (2P1C) model, and pressure drop values were then obtained (Table 1).

6. Summary

The results of the CFD model for a two-plate, one channel were used to predict pressure drop that occurs in a single channel. The viscous and inertial resistance factors showed a relatively close pressure drop was achieved with the equivalent porous media model. The next step is to model part of the MSTR core using the porous factors, and validating it experimental values from the reactor.

Table I. Results used for porous coefficients

Reactor Power	Max. Velocity (m/s)	$dp/dy_{2p1Cmodel}$ (Pa/m)	dp/dy_{porous} (Pa/m)
200kW	0.030	91.8	94.3
100kW	0.028	78.7	73.8
60kW	0.017	47.5	43.4
20kW	0.008	19.3	21.1

Acknowledgement

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III. PREDICTION OF MISSOURI S&T's NATURAL CONVECTION WITH POROUS MEDIA APPROXIMATION

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Abstract

The Missouri University of Science and Technology (Missouri S&T) is considering a power uprate of its 200kW research reactor (MSTR). To support this goal, preliminary CFD analysis was carried out to complement neutronics analysis on the current reactor. A three-dimensional parallel-plate model was developed using STAR-CCM+ v 8.04, and steady-state simulations for fluid flow under natural convection were performed. Cosine-shaped heat flux as a function of reactor power was applied on fuel plates. Temperature field in the hot channel were calculated at 200kW, 100kW, 60kW and 20kW power levels, and the resulting temperature profiles described the heat flow from the fuel plates into the surrounding coolant/moderator. To model the entire reactor, porous media approximation at the core was applied to reduce the computation cost. Using CFD simulation for four power levels, the inertial resistance tensor and viscous resistance tensor were found to be 281005 kg/m^4 and 7121.6 kg/m^3 respectively. Subsequently, the parallel-plate section was replaced with a porous section. The pressure drop within the channel for both cases was found to be within 10% of each other. For the investigation of the heat flow in the MSTR pool, a porous region core was defined by

both resistance tensors and porosity of 0.7027. A section of MSTR with 3 fuel elements and a power density of 1.86E+6 Wm^{-3} was modeled with one third of the reactor pool. Temperature measurements were made to validate the simulation results and at 200kW. The average temperature difference between the measured values and the simulated results was 0.29 K. The maximum difference between the simulation results and the measurements was observed to be less than 2 K at 0.9 m from the bottom of the core which is also 0.3 m above the top of the fuel. After porous media model validation, flow field in the reactor pool were generated with the new active cooling system operated at 35% pumping capacity. These results will provide a framework for power uprate safety analysis.

Keywords: Research reactor, Porous media, natural convection, CFD

1. Introduction

Reactor behavior is described through coupling between neutron kinetics and thermal hydraulics (Lamarsh and Barrata, 2001). It has been reported that computational fluid dynamics (CFD) codes provided in-depth understanding of fluid flow and heat transfer processes in nuclear reactors (INL, 2006; NEA, 2007; Smith, 2010). Yan (Yan, 2011) investigated the 20MW Australian Replacement Research Reactor using ANSYS Fluent to model the forced convection by using porous media approximation under both laminar and turbulent flow conditions. The neutronic and CFD codes, RELAP-3D and Fluent, were coupled and used to simulate the primary coolant system in the Gas Turbine-Modular Helium Reactor (Yan, 2011). Recent CFD results by Y.-H. Tung et. al. (2014) described the heat flow behavior of a prismatic very high temperature reactor (VTHR) during a loss of flow accident (LOFA). Their modeling strategies utilized the symmetry of the core with porous medium approximation. In modeling plenums, reduced upper and lower plena heights were reported to be sufficient to capture effects of natural circulation. Yoon and Park (2008) used the CFD code, CFX, to improve a three-dimensional CFD model for Canada deuterium uranium (CANDU) reactor by predicting the temperature distributions of the moderator inside its calandria vessel. They analyzed the moderator transient for the 35% Reactor Inlet Header (RIH) break without ECC (Emergency Core Cooling) injection so as to determine whether the fuel channel integrity is maintained. CFX was used to study steady-state moderator circulation under operating conditions and the local moderator subcooling during a LOCA transient. J. Chang (2008) developed a CFD model for Pennsylvania State's Breazeale Nuclear Reactor using 3D FLOW. The analysis of which described steady state temperature fields and showed the pattern of heat flows inside the reactor during pulsing. They also validated the simulation results with temperature and velocity data collected using thermocouples and micro turbine meter respectively. There is an increasing level of acceptance and utilization of the CFD method to characterize core behaviors for various types of nuclear reactors (NEA, 2007; Smith, 2010; IAEA, 2008; IAEA, 2012).

2. Overview of the Missouri S&T Reactor

The Missouri University of Science and Technology Reactor (MSTR) has been in operation for over 50 years. During this time the reactor went through a major change when the fuel was replaced from HEU to LEU. During various safety studies and re-licensing effort the core was investigated using both thermal hydraulics and neutronics code (Corvington, 1989; Carroll, 2004; Kulage, 2010; Richardson, 2012; O'Bryant et.al., 2012; Sipaun et.al., 2013; Castano et. al., 2013). Corvington performed neutronics calculations for the fuel conversion from HEU (Fig. 1) to LEU (Fig. 2). In his study, he used 2DB-UM and LEOPARD codes to predict several reactor safety parameters such as power peaking factor, moderator and void coefficient as well as core multiplication factor and neutron flux profile for both HEU and LEU cores (Corvington, 1989).

Accident analyses of this reactor were performed by Carroll (2004) using PARET and CONVEC codes, where the investigation of reactor behavior under accident conditions and reactivity transients were performed. Kulage (2010) performed calculations to estimate the neutron flux spectrum of the MSTR using the SAND-II program and MCNP codes. The spectrum was also experimentally measured using foil flux monitors. The thermal, intermediate and fast fluxes at full power were estimated to be 2.94E+12±1.9E+10, 1.86E+12±3.7E+10 and 2.65E+12±3.0E+3 n.cm^{-2} s^{-1} respectively (Kulage, 2010).

Richardson and co-workers (2012) developed an MCNP model for MSTR to predict the reactor's neutron flux profile. Using neutron cross-sections at the average core temperature, the predicted flux profile was consistent with the experimental data. To achieve greater model accuracy, they also reported a 2.7% increase in modeling fidelity when they used two temperature regions for the top and bottom sections of the core (Richardson et.al., 2012).

O'Bryant (2012) determined the hot channel factor for both a clean core and burnup corrected MSTR core model. It was found that the hottest channel is located between the 6th and 7th fuel plates of control rod one (CR1 located at D7, see Fig. 2). The ratio of maximum to average value of energy deposition was reported to be 1.85 and 1.71 for clean core and burnup corrected core respectively. There was no shift of the hottest channel which remained at position D7 for the burn-up corrected core as it was for the clean core (O'Bryant, 2012). Compared to the HEU core (Corvington, 1989), the location of the hottest channel in the current LEU core had shifted from location E6 (CR3 of HEU core) to location D7 (CR1 of LEU core) (Fig. 1 and Fig. 2). The shift is one slot above to the right from the previous hot channel location and this is probably due to the extra fuel element added to compensate for the LEU fuel.

At the time of HEU-to-LEU fuel change the entire core was also moved closer to the thermal column and the beam port. The power peaking factors were reported to be 2.22 with the LEOPARD code, and 1.71 with MCNP Code (Corvington, 1989; O'Bryant, 2012). In April 2012, a heat removal system was installed to allow continuous reactor operations. (Castano et. al., 2013). As use of MSTR continues to grow, a power uprate that will provide a higher flux is a next step for the MSTR.

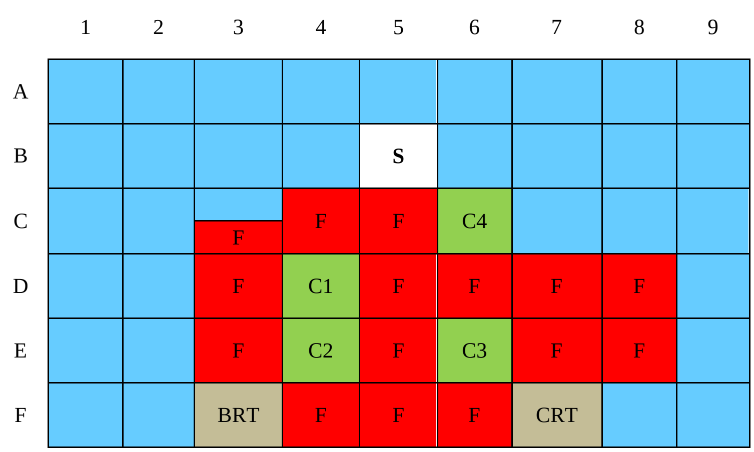

Figure 1. HEU Core: 1 Be Source, 4 Control Rod (C), 14 Fuel Elements (F), and 1 Half Fuel Element

	1	2	3	4	5	6	7	8	9
A									
B						**S**			
C					C4	F5	F1	F17	
D				F4	F8	F14	C1	F10	F2
E				F9	C3	F12	C2	F7	F3
F				CRT	F15	HC	F13	BRT	F6

Figure 2. LEU Core: 1 Be Source, 4 Control Rod (C), 15 Fuel Elements (F)

Several simulation results are described in the following sections with the intention to update and complement previously reported MSTR models. The goal of this study is to provide predictions of MSTR thermal-fluid parameters;

- MSTR temperature fields via CFD modelling and simulation at 200kW
- Natural convection heat flow in a hottest coolant channel at 200kW, 100kW, 60kW, and 20kW
- Application of porous media approach for the core, and determination of porous parameters for CFD calculation using various channel powers.
- Analyze the effectiveness of porous model for power uprate from 200kW to 500kW

2.1 Core Description

The core has a total of fifteen fuel assemblies and four control rods that can generate up to a maximum total flux of $4.36 \times 10^{12} \pm 2.84 \times 10^{11}$ neutrons/cm^2/s (Bonzer and Carroll, 2008). The reactor core is cooled by natural convection. The reactor coolant system consists of the reactor pool, a demineralizer that keeps the water quality within limits, a pool water makeup system and a Nitrogen-16 (N-16) control system to actively disperse N-16 generated in the reactor pool (Bonzer and Caroll, 2008) (Figure 3). These components work in tandem to keep the core cooled as well as to allow the reactor to be operated in a safe condition. The pool water acts as a heat sink and excess heat is removed by evaporation. A cooling system was installed at the MSTR which consist of a heat exchanger, a chiller system, processed water and chilled loops, and a control unit (Castano et. al., 2013). With the new cooling system, the rate of heat removal is improved and reactor operation is no longer limited by the reactor pool heat build-up. The pool temperature decreases 20 degrees Fahrenheit for every 1.5 hours of operation (Castano et. al., 2008). With the new cooling system the pool water can be maintained between 65 F and 75F at maximum power operation allowing a power uprate.

2.2 Fuel Characteristics

The MSTR core has a total of 310 MTR-type fuel plates and approximately 295 channels through which the coolant flows. The fuel material is made from uranium silicide, U_3Si_2Al with 19.75% uranium enrichment and has total heat transfer area of about 30 m^2. One fuel plate consists of a narrow slice of U_3Si_2-Al fuel that is bounded by aluminum cladding. Table 1 list the geometrical specification of a standard element of the LEU U_3Si_2Al fuel.

The fuel dimensions are approximately 0.05 cm x 6.10 cm x 60.96 cm (0.02 in x 2.4 in x 24 in). The cladding is a layer of aluminum alloy 6061 which is 0.038 cm (0.015 in) thick. The overall plate thickness is about 0.13 cm (0.05 in). The gap between two fuel plates is approximately 0.315cm. A standard fuel element has 18 fuel plates (Fig. 4), a half element has 9 fuel plates, and the control rod element has 10 fuel plates (Bonzer and Carroll, 2008).

2.3 Heat Flux

The non-uniform heat flux along the axial length of the fuel plate presented the need to modify neutron cross sections so as to obtain high-fidelity neutronics as well as thermal hydraulic core model (Richardson, 2012). The non-uniform fluxes at four power levels are shown in Fig. 5. The maximum to average heat flux value is 1.31. These flux distributions were applied as thermal conditions on the fuel plates for obtaining convective flow velocities at low power levels of 20kW and 60kW, and at high power levels of 100kW and 200kW.

Table 1. Geometrical specifications for a standard fuel element

Fuel Meat Thickness	0.51mm
Fuel Meat Width	61.0mm
Fuel Meat Length	610.0mm
Number of Plates	18
Cladding Thickness (Aluminum Alloy 6061)	0.381mm
Plate Thickness	1.27mm
Channel gap spacing	3.15 mm

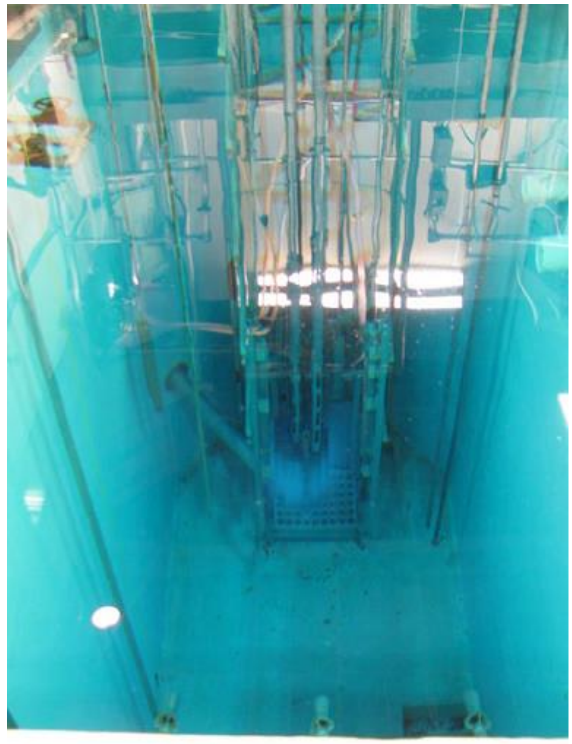

Figure 3. MSTR core has 15 fuel elements, 4 control rods

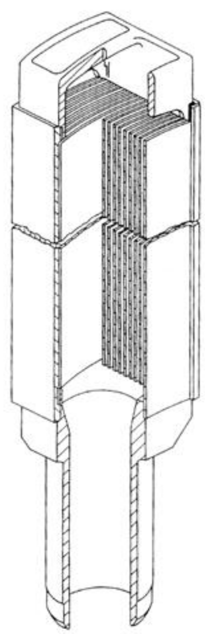

Figure 4. Standard Fuel Element

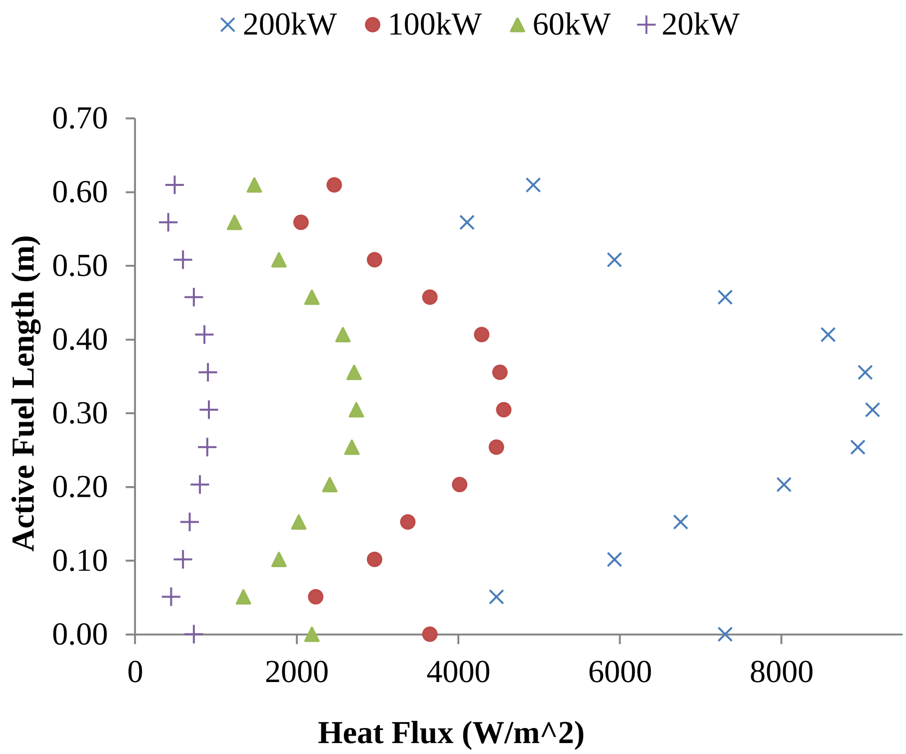

Figure 5. Heat flux profile at various nuclear power level

2.4 Porous Media Approach and Model Development

Part of the problem of thermalhydraulic modeling and simulation of a nuclear reactor is the complexity and large size of reactors. The MSTR pool is rectangular-shaped, and is approximately 5.79 m (19 ft) long, 2.74 m (9 ft) wide and 8.23 m (27 ft) deep (Bonzer and Carroll, 2008). It houses the reactor, a beam port, and a thermal column. It contains about 113.56 cubic meters (30,000 gallons) of demineralized water (Bonzer and Carroll, 2008). Modelling the convective behavior of such large pool requires enormous computer resources, and processing time to perform the calculations. One of the strategies that can be used to counter these limitations is adopting a porous media approach to model the core, therefore minimizing the number of computational cells that are needed to model the whole reactor (NEA, 2007).

The MSTR core can be approximated as a porous media due to its homogenous configuration; made up of a uniform arrangement of fuel plates interspaced with cooling channels. The core consists of an array of pores where coolant (water) flows through. In the first part of this work, we opted to model a “unit cell” of the MSTR core: consisting of two fuel plates and a coolant channel. The reactor core consists of identical array of parallel-plates and channels. Since there is no cross-flow, pressure difference across channels is assumed to be a function of channel power. This assumption allows for determination of flow rate as a function of channel power and hence porous medium approximation with variable channel characteristics could be used for various power levels. A prediction of the pressure drop at various operational power levels, and a characteristic equation of the core as a porous media were obtained. Based on the results of the parallel-plate model, the porous parameters were then used to model the MSTR core, and this model was validated with temperature data of the MSTR pool at 15 locations. The third and final part of the work studied the effect of the active cooling system in the MSTR through a prediction of heat flow and coolant removal rate.

3. CFD Model Description

The Star-CCM+ v 8.04 code was used to solve the energy and flow equations in a finite volume of both a unit cell model of the MSTR and a representative MSTR model at steady-state conditions. The coupled flow model solved the conservation equations for mass, momentum, and energy simultaneously using a pseudo-time-marching approach. The integral equation (Eq. (1)) represents the transport of a scalar quantity ϕ in a continuum. The terms in this equation are transient term and the convective flux on the left side, and the diffusive flux and the volumetric source on the right side (CD-Adapco, 2013).

$$\frac{\partial}{\partial t}\iiint_V \rho\phi dV + \iint_S \rho\phi\vec{u}.d\vec{A} = \iint_S \Gamma_\phi \vec{\nabla}.d\vec{A} + \iiint_V S_\phi dV \qquad (1)$$

In the first CFD model, a unit cell of the MSTR core was built consisting of two fuel plates and one coolant channel (Fig.6). This model was built to simulate the convective heat transfer from the fuel plates into the surrounding coolant. Polyhedral and embedded thin mesher models were used for mesh continua, with thin solid thickness defined as 3.15mm corresponding to the gap between fuel plates. The estimated Rayleigh number of the setup is in the range between and 10^{11} and 10^{12} (Bejan, 1995).The flow and heating processes were modeled in a computational domain representing 0.85m x 0.01205m x 0.082m volume. The plate length (L) to channel width (W) ratio is approximately 200 (L/W). The polyhedral and embedded thin mesher obtained good resolution of the heat flow process. A coolant temperature of 294K (21°C) was used, which corresponds to the regulated coolant temperature at the reactor. Boundary conditions applied at the fuel plate surfaces were non-uniform heat flux and follow the flux profile of the active length of the core (Fig. 5 and Fig. 6). The side walls bounding the fuel plates were taken to be symmetric planes. The lower and upper boundaries were kept at the constant temperature of 294K. Physically, the fuel assembly is submerged in a deep pool, and the top and bottom surfaces experience little temperature changes, essentially forming walls of constant temperatures. The conductive heat transfer within the fuel plates was ignored in order to decrease the computational mesh requirement.

Coupled flow and heat transfer calculations were implemented to capture the convection phenomena.

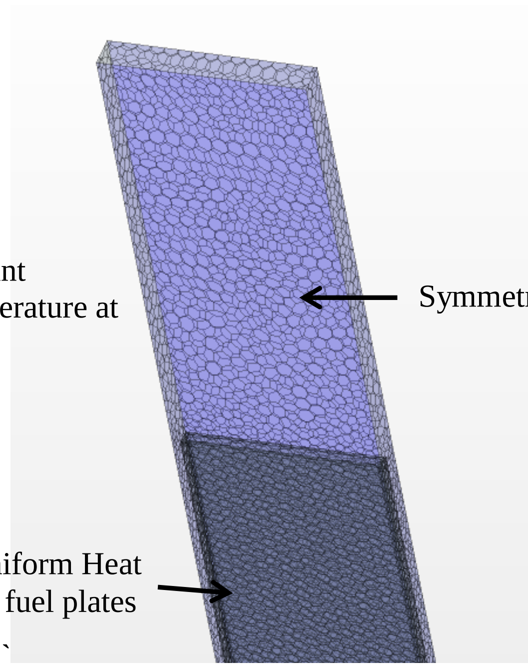

Figure 6. Boundary conditions on fuel plates (top section)

In the second CFD model, a representative model of the MSTR was developed (Fig.7). In the modeling of the MSTR, its core's detailed assembly was replaced with a porous region that mimics the pressure drop and temperature variation as in a detailed core. Figure 7 show a simplified model of the MSTR, consisting of one third of the reactor pool, three fuel elements (1.86E+6 Wm^{-3}), an eductor (Model 46550 Tank Mixing Eductor), a 4" pipe inlet with cone-shaped 6" opening, and a fuel storage area. The inlet serves to remove heated water from the pool into a heat exchanger system. The cooled water is returned to the pool through an eductor that is attached to a pipe which is angled

30° downward from bulkhead wall. The same non-uniform heat flux values (Fig.5) were applied on the porous fuel region and the aluminum sleeves were made adiabatic. Temperature of the reactor walls were at 294K. The walls and the bulkhead surface were maintained at 294K as their locations are far enough from the core that they are not highly affected by the core heat. Top of the pool was set to be adiabatic.

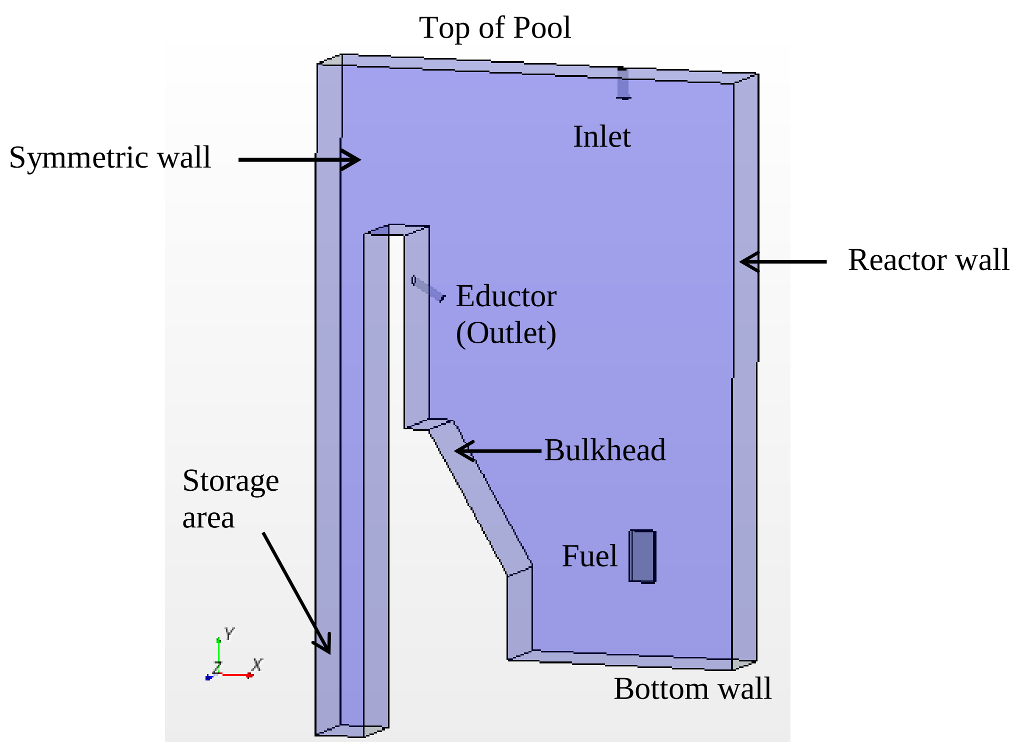

Figure 7. MSTR model consisting of 1/3 of the reactor pool, 3 fuel elements, eductor, and inlet

The eductor was set as a mass flow inlet allowing a coolant mass flow rate of 0.4536 kg/s (1 lbs/s) into the pool at 294 K. The outlet was set as a pressure outlet at atmospheric pressure. Pool temperature was at 300K at initial condition and this corresponds to experimental condition. All CFD calculations were performed using Intel i7 at 2.2GHz with 8GB RAM.

3.1 Porous Media Description

The Navier-Stokes equations are valid for the flow motion inside a porous media that consist of voids and solids. The equations, however, are simplified by considering the porous medium as a continuum in which the velocities and pressures are averaged over small but finite pore volumes (Lage, 1998). An important property of porous media is permeability, which is a measure of a porous medium's capability of transferring fluid throughout the pore space within the medium. Pressure losses in a porous medium due to viscous effects are described by Darcy's Law. Equation (2) describes a linear relationship between pressure gradient and filtration velocity derived from Darcy's Law (Lage, 1998). It is valid for seepage flow and small permeability where the pore Reynolds number based on the local volume average velocity is less than unity.

$$-\frac{dP}{dx} = \frac{\mu}{\kappa} u_f \quad \text{for Re} < 1 \qquad (2)$$

where dP/dx (N/m^3) is the pressure gradient along the x axis or length of the column, μ ($N\text{-}s/m^2$) is the fluid viscosity, u_f (m/s) is the filtration velocity or ratio of total pore space volume flow rate (m^3/s) to total pore space area (m^2), and κ(m^2) is the average medium permeability.

For higher flow rates (Re > 1) in porous media, the pressure gradient begins to deviate from a linear relationship. This departure arises from inertial, viscous and convective effects which were neglected in Darcy's Law for seepage type flow, and results in over prediction of the fluid motion. A quadratic term is added to Darcy's law to account for the effects when fluid Reynolds number increases to more than unity. Equation (3) is known as Forchheimer's equation and β (m^{-1}) is often referred to as Forchheimer's coefficient. The quadratic term relates pressure losses within a porous media to inertial dissipation (Lage, 1998; Huang, 2003).

$$-\frac{dP}{dx} = \frac{\mu}{\kappa} u_f + \rho \beta u_f^2 \quad \text{for Re} > 1 \tag{3}$$

3.2 Porous Media in STAR-CCM+

The effect of the porous medium on the flow was defined using lumped parameters. The parameters are typically taken to be resistance coefficients for a source term in the momentum equation. The inertial and viscous coefficients are required for the porous source term in the momentum equation. The macroscopic effect of the porous medium on the overall fluid flow is studied, and without much emphasis on the details of the internal flow.

The porous source term appears in the momentum equations of the coupled and segregated flow solvers

$$f_p = -P.v \tag{4}$$

where P is the porous resistance tensor. Porous resistance tensor is given by

$$P = P_v + P_i|v| \tag{5}$$

where

P_v is the viscous (linear) resistance tensors

P_i is the inertial (quadratic) resistance tensors

In the porous region, the theoretical pressure drop per unit length can be determined using the equation below.

$$\frac{\Delta P}{L} = -(P_i|v| + P_v)v \tag{6}$$

The simulation captures the behavior of coolant flow into a narrow channel that arises from convective heating from two fuel plates, and to obtain the associated pressure drop in the channel. A curve fit of $\frac{\Delta P}{L}$ versus v^2 and v was applied to equation 6. From the pressure gradient, the viscous and resistance tensors were predicted from a characteristic equation on a curve fit on a plot pressure drop per length versus average channel velocity. This geometry (two fuel plates and a channel) was taken as the smallest unit of the core region.

The pressure drop (between the channel entrance and channel exit) and volume-averaged velocity was predicted as well as the maximum exit velocity at 20kW, 60kW, 100kW, and 200kW. From the two variables, the equation that describes their relations was obtained. The equation was correlated to the viscous resistance factor and the inertial resistance factor. The factors formed the basis to model the fuel assembly using porous media approach. Polynomial curve fit for Forchheimer's equation was obtained for the two plates-one channel model.

3.3 Grid Study

A grid study was carried out to identify the required number of computational cells to obtain grid independent results for the parallel-plate model. These simulations confirmed that the mesh consisting of 19714 cells was satisfactory. Increasing cells over 20,000 gave little improvement in the calculations for exit temperature. The chosen mesh gave 20% computational time savings compared to the largest mesh size with 72018 cells (Fig. 8).

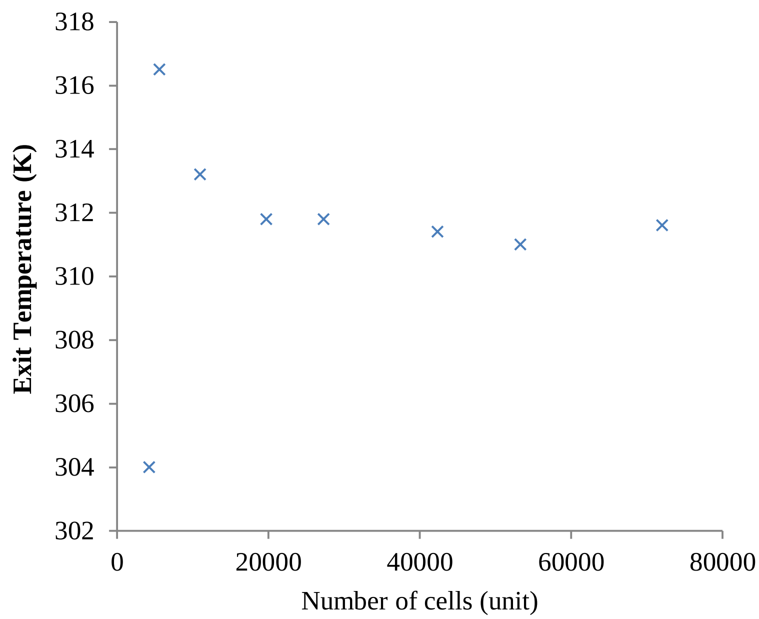

Figure 8. Grid study to check for independence

4. Results and Discussion

In the parallel-plate model, the simulation results indicate that heat transfer between fuel surfaces to the coolant within the channel can be described as conductive at each reactor power levels due to linear temperature changes along the narrow channel gap of 3.15mm (Figure 9). The heated coolant that passed through the gap is shown to move upflow from the top of the channel to the top of fluid domain. It meets the colder fluid at the top, and the slow mixing reduces the exiting coolant temperature to values lower than 340K at 200kW.

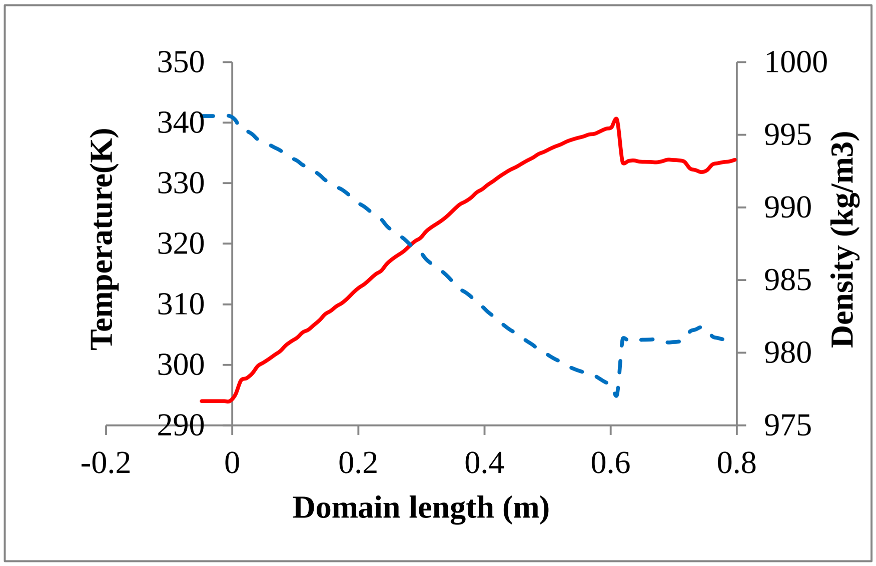

Figure 9. Temperature (solid line) and density (dash lines) variation in the channel at 200kW

Each reactor power level has a unique heat flux distribution associated to it. When applied on the plate surface, each cosine-shaped heat flux produced predicted values of temperature, density, and pressure drop that occurs within the channel at each power level. Figure 10 shows temperature increase along the channel length and figure 11 shows density decreases within the channel as the coolant is heated up due to heat transfer from the fuel plates. To obtain the porous parameters, the pressure difference between the bottom and top of the channel and its related maximum velocity for the channel were obtained (Figure 12). Hydrostatic pressure effect was neglected in the pressure calculation. The number of pore space was determined using the volume porosity as described by Todreas & Kazimi (1990). Volume porosity was calculated to be 0.7027, and was used in the porous model.

$$\gamma_V \equiv \frac{V_f}{V_T} = \frac{Fluid\ volume}{Total\ volume} \tag{7}$$

Figures 10 and 11 provide a characteristic profile of the hot channel behavior in MSTR, and can be used as interface information in the development of coupled codes between thermal hydraulics and neutronics. This result is valid for powers between 20kW and 200kW. The pressure drop and maximum velocity achieved within the channel was obtained, and is presented in Table 2. The predicted porous parameters within this power range are the viscous resistance tensor, P_v and the inertial resistance tensors, P_i and were found to be 281005 kg/m^4 and 7121.6 kg/m^3 respectively.

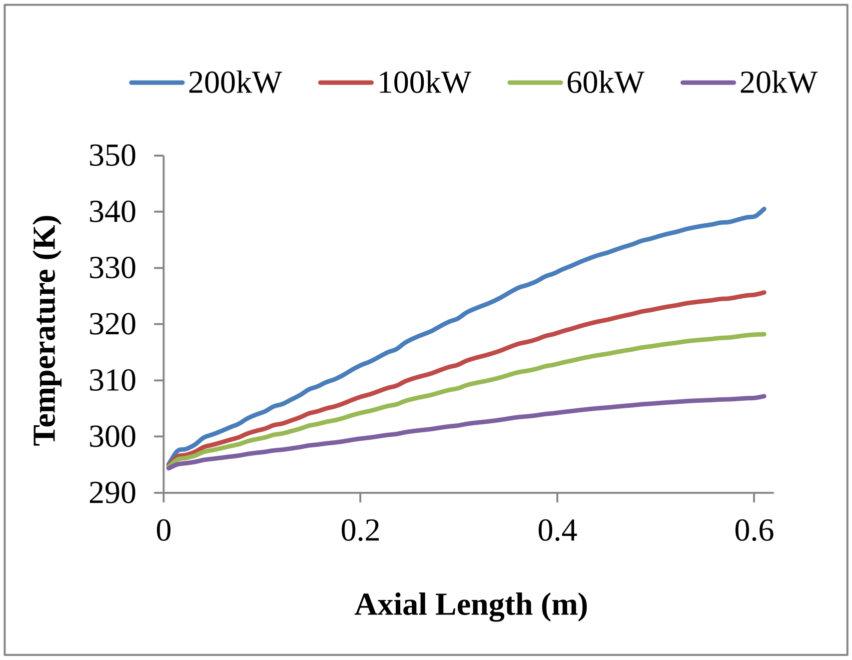

Figure 10. Temperature changes within the coolant channel

Due to the nature of free convection, the inertial resistance tensor was a magnitude higher than the viscous resistance. The tensors were then used in a porous section that replaced the two fuel plates and one channel section (2P1C) in the parallel-plate model. The pressure drop values for both porous section and 2P1C section were within 10% of each other (Table 2).

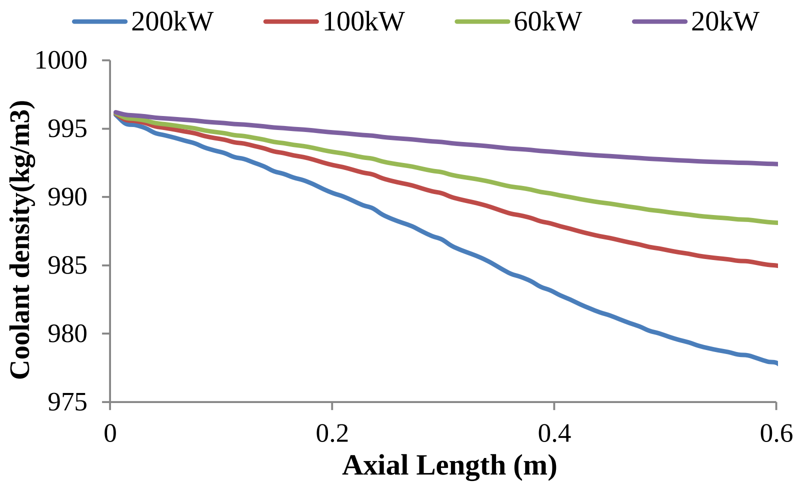

Figure 11. Density changes within the coolant channel

This simulation result suggests that the porous parameters are suitable for predicting temperatures, density, and pressure losses under natural convection condition without detailed fuel assembly modeling. It would be appropriate to use the inertial and viscous resistance factors to model the whole core by replacing fuel assemblies with an equivalent porous region with variable porous parameter as a function of channel power. The porous model does not require geometrical details, however, retains the prediction accuracy for the reactor pool temperatures.

Table 2. Results used to find resistance tensors

Reactor Power	Max. Velocity (m/s)	dp/dy$_{2P1C}$ (Pa/m)	dp/dy$_{porous}$ (Pa/m)	% Differ ence
200kW	0.03	91.8	94.3	2.7
100kW	0.028	78.7	73.8	-6.2
60kW	0.017	47.5	43.4	-8.6
20kW	0.008	19.3	21.1	9.3

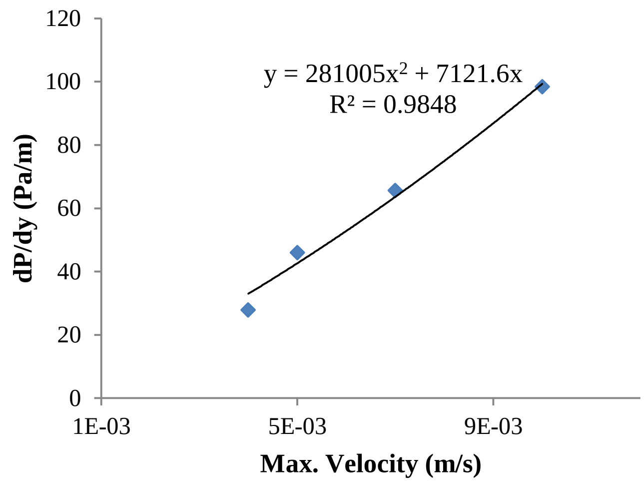

Figure 12. Pressure drop per length with maximum velocity at channel exit

The thermal and velocity predictions for cross-sections of the reactor pool are reported in this section. In addition, temperature data were collected at position C9 (Figure 2) of the MSTR fuel core. A line probe was created in the MSTR model at a location outside the porous core. This line probe was made to correspond to the thermocouple locations along the vertical length of the pool at location C9 (Fig.2). The experimental data were measured using 15 units of K-type thermocouple with position zero corresponding to the bottom of the fuel core which sits on a grid plate. At the time of data collection, the active cooling system was not operated; therefore the reactor was cooled solely by natural convection.

Figure 13 shows the coolant temperature values in the MSTR model is in good agreement with the pool temperature measurements. The source of difference between simulation and experiment values seen at locations before 2 meters is due to the model not taking into account the presence of other physical structures (grid plate, inverted aluminum tower assembly) in the reactor core that impact the flow, induce mixing and hence reducing the temperature stratification in the pool.

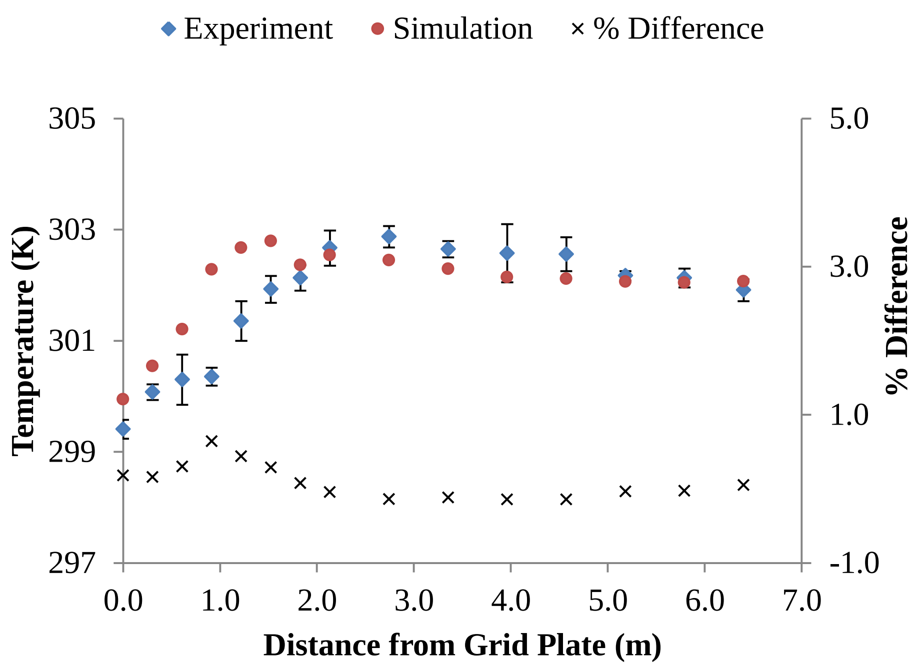

Figure 13. Comparison between experiment and simulation

Figure 14 show a cross section view of the temperature field, and heat upflow from the fuel core into the surrounding coolant. The inlet takes in the heated water. There is a gradual heating up of the pool, while the coolant behind the bulkhead remains fairly constant at 295K. This area is reserved for fuel storage and is intended to be unaffected by the heat removal mechanism of the MSTR. The heating from the fuel core increased the surrounding coolant temperature from 295K at the bottom of the reactor pool to 302K at top of the pool. The reduced the density of coolant/moderator and the warmer fluid is pushed upwards due to buoyancy force, while the colder fluid flow downwards with gravity.

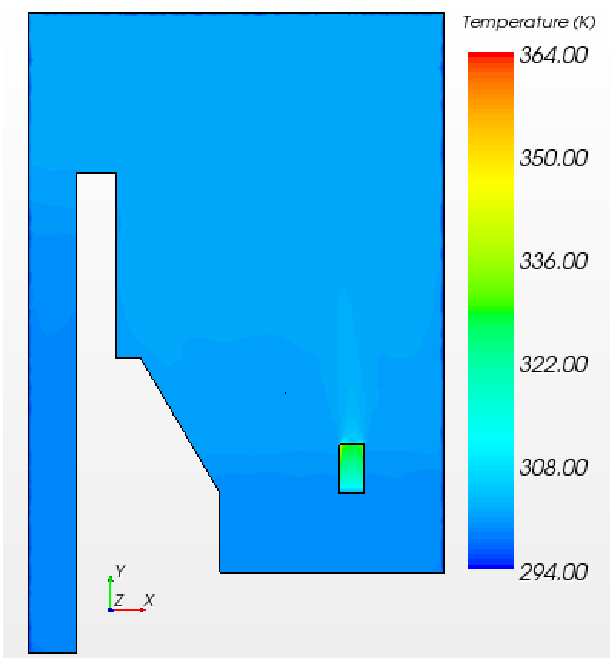

Figure 14. Cross-sectional view cutting through the porous region shows upward heat flow at 200kW(without active cooling)

Figure 15 show the velocity field in the MSTR pool when the reactor is operated at 200kW. Cool water is discharged at a constant rate 0.4536 kgs^{-1} into the pool through the eductor, and is mixed with the bulk pool water. Heated water is drawn to the inlet where the mass flow rate was predicted to be 1.28kgs^{-1}.

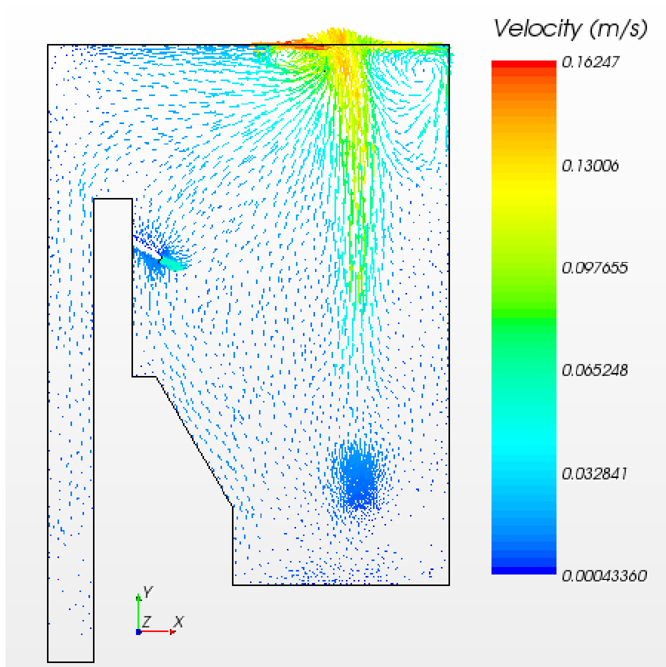

Figure 15. Cross-sectional view cutting through the eductor at 200kW (with active cooling system)

5. Conclusion

CFD has been used to provide thermal behavior predictions in reactors for postulated cases and under varied conditions. The Missouri University of Science and Technology is in the initial stage of a reactor upgrading exercise that is seeking to extend its neutron flux capacity for its reactor-based research. Two CFD models of the Missouri University of Science and Technology Reactor (MSTR) were developed to support the upgrading plans. Modeling and simulation was carried out using a computational fluid dynamics code (CFD), STAR-CCM+ v 8.06. The first model is a unit cell of the core, consisting of two fuel plates and one coolant channel. This three-dimensional parallel-plate model was used in the steady state CFD analysis at 20kW, 60kW, 100kW, and 200kW power levels. The goal of the first model was to obtain porous media parameters for the MSTR core that is valid between 20kW and 200kW power levels. The predicted parameters are the inertial resistance tensor, $P_i$and the viscous resistance tensors, P_v and were found to be 281005 kg/m^4 and 7121.6 kg/m^3 respectively. The channel temperature, velocity and pressure fields were obtained, and simulation results show coolant temperatures and density as a function of core power. In this model, the parallel-plates and channel were then replaced with a porous section. The pressure drop within the channel/section for both cases was within 10% of each other, and indicated that both tensors were adequate for use in modeling the MSTR core using the porous media approach.

The second model is a representation of the entire MSTR including an inlet/outlet from a secondary cooling system which was installed to support reactor power upgrade. A section of MSTR with 3 fuel elements and a power density of 1.86E+6 Wm^{-3} was modeled with one third of the reactor pool. The core was modeled as a porous media by using the porous parameters from the parallel-plate model. For all cases, non-uniform heat flux was applied on the fuel plate surface to reflect the MSTR cosine-shaped flux. At 200kW and without operating the active cooling system, the temperature field was found to be in good agreement with the pool temperature data. The average temperature difference between the measured values and the simulated results was 0.29 K. The maximum difference between the simulation results and the measurements was observed to be less than 2 K at 0.9 m distance from the bottom of the core which is 0.3 m above the top of the fuel. At 35% pump capacity, the simulation results for the MSTR model showed that water is drawn out of the pool at a rate 1.28 kgs^{-1} from the 4" pipe, and predicted a surface temperature of the pool not exceeding 30°C. It was found that the porous parameters were adequate for use in replacing the MSTR core with a porous region, and to investigate coolant flow inside the reactor pool. The simulation results provided thermal-fluid parameters for normal operations and baseline parameters for supporting license renewal as well as power uprate plans.

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IV. SUPPLY CHAIN FEASIBILIY ANALYSIS OF SMALL MODULAR REACTOR TECHNOLOGY

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Abstract

The feasibility of Small Modular Reactors (SMR) in providing a lower-cost power-generation alternative to large nuclear power plants has been suggested widely in recent years, but remains a largely undeveloped technology. Published reports have identified many of the challenges of building SMR technology, with the major ones being industrial capacity, local manufacturing, forging capability, competitiveness in global market, and supply chain management. This paper focuses on the adaption of supply chain management concepts and practices of nuclear power companies, in particular Westinghouse Electric Company LLC, which supports the nucleation of an SMR economy in the state of Missouri. The SCOR framework is used to define supply chain processes and partnerships in Missouri that will be essential in the establishment of SMR operations. The framework builds on findings from a review of open literature including modular specifications, economic reports, and supplier information for the state of Missouri and the U.S. This study investigates several key factors that influence the supply chain process: strategic supplier partnership, achievement of efficiency gains, supplier qualification/supplier development, and sustainability. The findings are intended to provide an overview of existing supply chain management best practices through a case study designed to formulate the development of an SMR supply chain.

Keywords

Small Modular Reactor, Energy Sustainability, Supply Chain, SCOR, Missouri

Introduction

Small Modular Reactors (SMRs) are an emerging class of nuclear reactors that are under development to meet the world's energy demands (IAEA, 2012). Small modular reactors (SMR) of the light water reactor category are also known as integral pressurized water reactors (iPWR) and are compact versions of the well-established design of pressurized water reactors (PWRs). SMR technology has received strong endorsement from the U.S. Department of Energy with funding approvals up to $452 million to develop this innovative new generation of nuclear reactors. Although they produce a fraction of the power generated from a large nuclear power plant (NPP), SMRs are designed to overcome high capital costs of a large nuclear power plant and offer stand-alone capacity for power generation away from large electricity grids (Vujic et. al., 2012; Abdulla et. al., 2013). The cost reduction results from short construction times due to modular construction for SMRs. Modular construction refers to factory-assembly and offsite manufacturing of the reactor components or modules. The SMR modules are then transported to the SMR site for final assembly. Because they are factory-built, the quality of standardized components can be controlled and the production process is expected to increase supply chain efficiency hence reduce the cost for the n^{th} deployment. Moreover, SMRs offer many advantages over traditional reactors due to improved safety features and incremental capacity building. These features open up new markets for non-traditional customers, such as developing countries, that will now be able to opt for nuclear power due to lower capital costs (Bennet, 1987). Industries that need power close to their facilities and customers in remote locations may benefit as well.

New nuclear builds, however, still carry a risk of failure due to several factors including build times, rising cost, and public acceptance (John W. Collins, 2011). Additionally, a totally new design such as the SMR must undergo certification and rigorous testing to comply with regulations set by the nuclear regulatory authority, U.S. Nuclear Regulatory Commission (U.S. NRC, 2014).

Rosner and Goldberg (2011) suggest that successful development of an SMR industry includes job creation and developing a performance metrics for SMR deployment. The development of a performance metrics for SMR deployment provides a means to measure U.S. competitiveness in the global nuclear market. The Westinghouse SMR is a prototype SMR under consideration for use in Missouri. It is an integral pressurized water reactor (iPWR); this design is a compact version of the regular pressurized water reactor (PWR). Compared to the conventional reactors, the iPWR is relatively smaller, and its unique design combines the entire reactor and the nuclear steam supply system into one reactor vessel (Fetterman et. al., 2011). The reactor vessel is located underground and this below-grade position protects the reactor from external threats from airplane crash and projectiles. In addition, the inherent passive safety design of the SMR can absorb powerful earthquakes, tsunami, and tornadoes without compromising the core's integrity. The SMR is designed to remove excess heat by natural convection in the event of an emergency without human intervention or the use of active heat removal systems such as pumps (Westinghouse SMR Brochure; Fetterman, 2011).

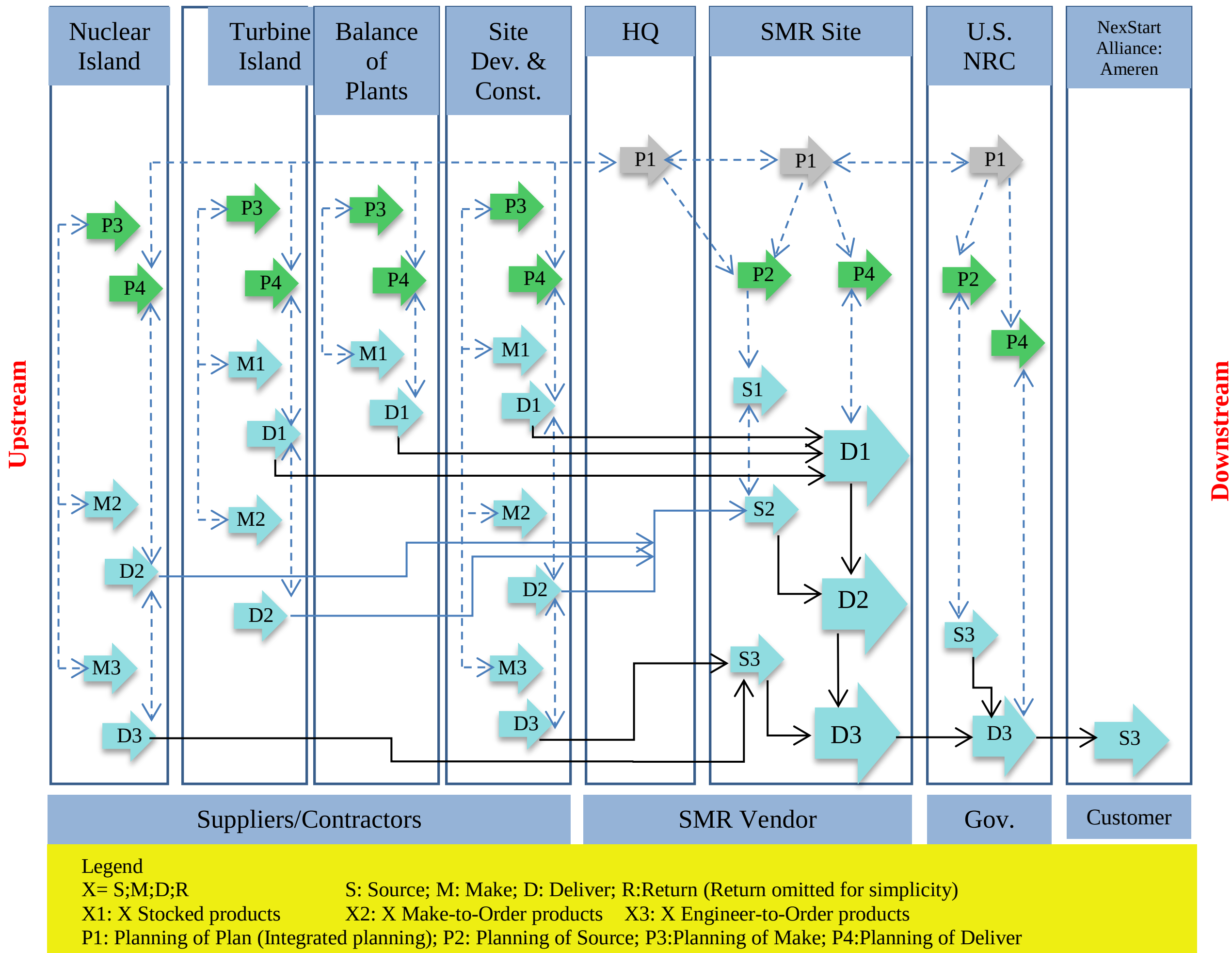
As an early adopter of this state-of-the-art technology, Missouri could derive benefits from a long-term strategy for energy security as well as to reduce CO_2 emissions from current coal plants. SMR technology could potentially bring about a wide-ranging impact on the state's economy by positioning Missouri as a manufacturing hub for SMR components. The feasibility analysis of supply chains to support the emergence of an SMR industry in Missouri is investigated in this research through the Supply Chain Operation Reference (SCOR) framework. The SCOR model is made by defining the supply chain processes and partnerships in Missouri that will be essential in the establishment of SMR operations. This study investigates several key factors that influence the supply chain process: strategic supplier partnership, achievement of efficiency gains, supplier qualification/supplier development, and sustainability. Having built a 1190 megawatt (MW) nuclear power plant (NPP) in Callaway, Ameren Missouri's technical experience in seeing through a complete process for building a NPP and Ameren's operational experience is valuable and applicable for adopting Westinghouse's SMR technology. Westinghouse Electric Company LLC (WEC) formed an energy consortium, NexStart SMR Alliance, comprising utility companies, current and prospective nuclear plant owners as well as investors. WEC have applied to the Department of Energy to invest in its SMR at Callaway Energy Centre in Fulton, Missouri. The findings of this study are intended to provide an overview of existing supply chain management best practices through a case study designed to formulate the development of an SMR supply chain.

SCOR Framework

Supply chain performance are evaluated through identification of supply chain drivers and its metrics. Statistical methods and analytical models are used to estimate operational parameters and to find means for an optimum supply chain configuration (Chopra and Meindl, 2012). To establish a successful SMR operation, we suggest that the supply chain operations and design is measurable, strategic and balanced view that includes all stakeholders. The supply chain model could identify key implementation obstacles, and key areas to promote economic growth. The supply chain design requires a method that could effectively communicate metrics that is understood across organizations, and the supply chain operation reference (SCOR) model fits our objectives (Supply Chain Council, 2006; Huan et.al, 2004). This research paper presents a feasibility analysis of supply chains to support the emergence of an SMR industry in Missouri using the SCOR framework. SCOR is an industry standard for documenting supply chain operations within an organization as well as across organizations. A Level 1 Process Model for Missouri SMR is presented in Exhibit 1. In this model, the key elements in a level 1 diagram are the organizations within the supply chain, and their key business processes. The processes are identified as plan (P), source (S), make (M), deliver(D), and return (R). In general, a company will source goods, transform the goods into another product, and finally deliver the finished product to their customer downstream in the supply chain. Return (R) describes the return process involved when the goods supplied are returned to the manufacturer for any reason (Supply Chain Council, 2006). Finally, the planning process accounts for aggregate demand, and the overall planning for how to channel the goods from sourcing to delivery (Supply Chain Council, 2006).

The SMR SCOR model was constructed by defining the supply chain processes and partnerships in Missouri that will be essential in the establishment of SMR manufacturing operations. The principal organization studied is the SMR vendor, WEC, along with the scope in which it influences other entities where its interaction aims to produce value for the end customers. There are four categories of suppliers involved in building an SMR; they are suppliers and contractors that supply parts, components, modules, and services to the nuclear island (NI), turbine island (TI), balance of plants (BOP), and site preparation and construction (SP&C). Because the WEC SMR is the first-of-its-kind nuclear construction, the U.S. NRC plays a significant role in reviewing and approving the vendors and the overall SMR design through a rigorous certification process to comply with safety regulations. Potential customers are utility companies. In exhibit 1, the conventions for identifying the type of process involved are marked as X1, X2 and X3 where the numbers 1, 2 and 3 refer to stocked items, made-to-order items, and engineered-to-order items. The modules and parts for NI, TI, BOP, and SP&C are manufactured off-site, and modules are shipped to the SMR site for final assembly. The flow of goods is indicated by solid lines, and it flows from suppliers directly to the construction site. From the SCOR model, we suggest several criteria that can be used to study the feasibility of establishing an SMR supply chain. These are strategic partnership, efficiency gain, supplier qualification/development, and sustainability.

Exhibit 1. Level 1 Process Model for Missouri SMR operations



Strategic Supplier Partnership

Within the context of global nuclear industry, nuclear power companies build strategic partnerships to provide an efficient supply chain for new nuclear builds (Global Data, 2013). A diverse network of nuclear plant and technologies suppliers can support a flexible and reliable supply chain to ensure timely delivery of new nuclear plants. WEC operates on a “Buy where we build” business approach where WEC relies on its plant engineering partners, local suppliers and contractors to sustain an extensive nuclear supply chain (WEC, 2009; NEI, 2013). WEC has an established engineering expertise in building nuclear reactors and developing fuel designs with manufacturing locations around the world. Out of five large nuclear power plants (NPPs) currently being built in the U.S., four reactors are of Westinghouse's AP1000 pressurized water reactor design. It received design certification from U.S. NRC in 2011, an important milestone in the process of domestic licensing for the new generation of nuclear reactors. This design is also used to build four new reactors in China, and three new reactors in the United Kingdom (U.K.). Westinghouse' SMR is designed based on the AP1000, where the passive safety system and a 17 x 17 fuel assembly design of the AP1000 are incorporated into the SMR design. The expected operational lifetime of this SMR is 60 years, with a 24 month refueling cycle.

The NexStart SMR Alliance is a partnership between current and prospective nuclear power plant owners and operators that supports the adoption of Westinghouse's SMR technology in Missouri. The selection of an Engineering, Procurement and Construction company is made to manage the submission of a combined and operating license (COL) application to the NRC. The COL allows for construction and operation of new SMR units. Westinghouse has in place an established purchasing and supply management system to select certified and qualified suppliers subsequently be added to an approved suppliers list. Under U.S. NRC regulations 10 CFR 50, the quality assurance criteria for nuclear power plants and fuel reprocessing plants are stipulated. Other criteria are needed to cover all aspects of nuclear facility construction such as ASME's quality assurance requirements for nuclear facility applications, IAEA's safety criteria IAEA-50-C-QA, and ISO 9001:2000. The suppliers for any SMR vendor are required to be certified and qualified to undertake the work required to meet the construction of a nuclear power plant. Among the criteria that are important for supplier application are experiences in nuclear construction, utility, and related industrial work. Vendors such as WEC tend to work with companies/suppliers that have a long working history with WEC and a proven track record in successful NPP construction. Due to the long-term nature of SMR partnership agreements, these suppliers must have stability and staying power to be sustainable supply network partner.

Achievement of Efficiency Gains

The current fleet of nuclear power plants was built using previous era construction technology based on fossil fuel plant construction. It has been suggested that construction for nuclear power plants adopt the ship-building construction technology due to both industries having similar scales of complexity (Seubert, 2011). The technology is based on Product Work Breakdown Structure (PWBS) that improves work flows on the process lanes by establishing work packages in ship building; specifically, hull construction, outfitting, and painting (Seubert, 2011). In nuclear power plant construction, work packages are divided into nuclear and non-nuclear module construction, outfitting, and on-site final assembly. The Westinghouse SMR is based on the approved design of AP1000, the construction of which benefits from modularization. The three levels of modularization are prefabrication, pre-assembly, and module assembly (IAEA, 2011). Cost reduction for SMR is expected to come from several factors:

- Having the modules manufactured at offsite locations, and therefore not requiring additional infrastructure and preparation

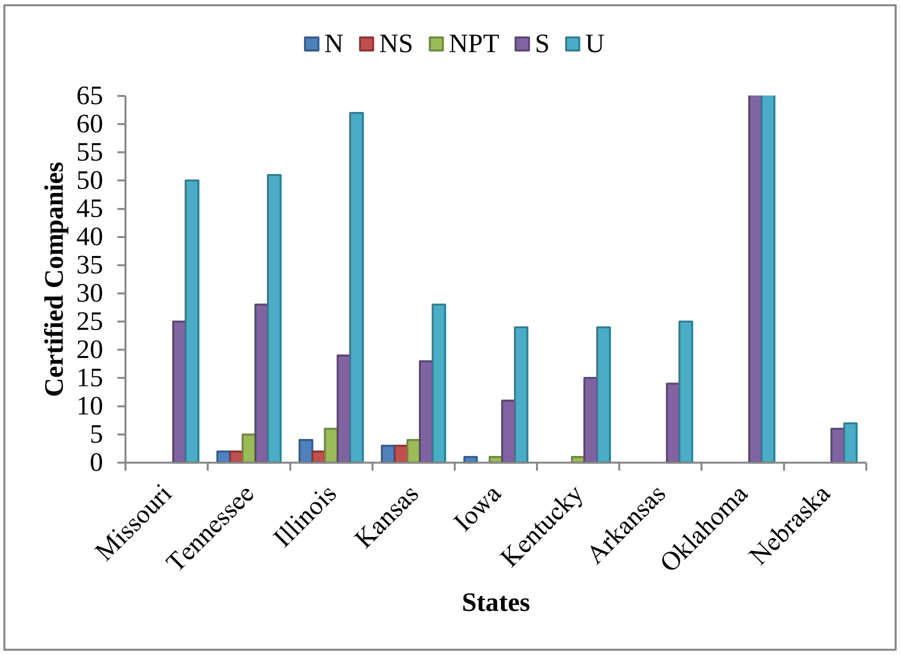
- SMR’s compact design allows for utilizing off-the-shelf components
- Improved quality control for factory-built modules
- Electrical and civil modules are transportable by truck and rail because of its compact mechanical design
- Open-top construction method (i.e. setting reactor pressure vessel directly into place using large crawler crane) leads to greater efficiency and shorter schedule
- Passive safety system that requires less equipment
- Integrated project planning and management

An improvement of supply chain performance is achieved through manufacturing efficiencies and delivery certainty. Overall, the SMR technology is better designed to improve on quality, cost and schedule.

Supplier Qualification/Supplier Development

The regulations for current generation of NPP are stipulated under U.S. NRC’s regulation, Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities”. These include construction permits and operating licenses (under 10 CFR Part 50) and design certifications, combined licenses (COLs), and early site permits (under 10 CFR Part 52, “Licenses, certifications, and Approvals for Nuclear Power Plants”. ASME for Boiler and Pressure Vessel for Nuclear Containment specifies codes for materials, operations and maintenance, inservice inspection, contruction of NPP components. Nuclear component suppliers and technology providers are required to adhere to standards and codes set by the industry as well as NRC in order to fulfil the requirements of domestic licensing. By understanding the innovation in the SMR design, suppliers and contractors are able to respond to specific regulations that govern licensing for SMRs.

Exhibit 2. Missouri Companies with Relevant Certifications (ASME, 2014)



Certification	Description
N	Nuclear vessels, pumps, valves, storage tanks
NS	Nuclear supports
NPT	Fabrication of nuclear appurtenances and supports
S	Manufacture and assembly of power boilers
U	Manufacture of pressure vessels

Current trends and practices in supply chain management show that suppliers and services are not only sourced locally but overseas as well. The goal is to achieve supply chain efficiency across supplier network through process integration, and to be able to respond to demand variability. The supplier selection framework evaluates potential suppliers based on business experience, business type, qualifications, and other relevant criteria determined by the vendors. For example, to apply to be a WEC supplier/subcontractor, companies submit information to indicate interest and may request to be evaluated on resource availability, technical leadership, quality, human performance, and continuous improvement work. Exhibit 2 shows the number of companies having ASME certifications in Missouri and its surrounding states. The result indicates that there are resources and infrastructures available for participation in SMR module manufacturing operations.

Infrastructure and Sustainability

Potential supply chain challenges for an SMR industry are summarized in the following points below:

- Meeting demand and supply,
- Rail and road infrastructure. Shipping allocations for international markets
- Labor availability
- Qualified and skilled workforce

An analysis of Missouri's workforce showed more than 10% difference between labor supply and demand in construction, installation, maintenance and repair (CIMR), production, and management/support work (MERIC, 2014). This trend is seen over almost all regions in the state. The surplus of workers in these areas would be available for retraining and employment in the SMR industry. In the 2013 National Manufacturing and Logistics Report, Missouri was rated among the highest in the nation for sector diversification (Center for Business and Economic Research, 2013). The top three Missouri exports for the 1st Quarter 2014 are in chemical, transportation equipment, and food and kindred products for a total of $1.573 billion (Missouri Department of Economic Development, 2014). Missouri is also among the most connected by Class 1 Railroads with six railroads (BNSF, CSF, KCS, NS, CP, and UP) currently in operation. In addition, there are three foreign trade zones located in Kansas City, St. Louis, and Springfield where major airports are also located.

Factors to be considered for sustainability are financial stability of contract manufacturers and the reliability of third-party logistics firms. Capacity for human resource development in SMR technology could be fulfilled by the universities and technical colleges in Missouri, including Nuclear Reactor Operator Certification at Missouri S&T. There are two university research reactors and one commercial nuclear power reactor in Missouri.

Conclusions

Small modular reactor technology shows potential in creating new market in energy solutions both domestically and overseas. It is designed to have better safety systems (inherently safe power operation), resistance to external threats, decentralized energy delivery, and the capability to overcome large initial investment of a conventional nuclear power plant. It also provides an alternative to shift away from fossil fuels. While the advantages of pursuing the SMR technology are clear and promising, there have been no commercial orders to build and operate commercial SMRs. One of the important issues for commercialization is supply chain management, and the preparation to build a network of components supply to meet demand is vital. Based on the results of this research, Missouri shows promise as a test site for SMR technology.

This paper describes several issues in supply chain of small modular reactor technology: strategic supplier partnership, achievement of efficiency gains, supplier qualification/supplier development, and sustainability. The global nature of the nuclear power industry suggests that companies that provide nuclear technology solutions would rely on strategic partnership and alliances to be able to deliver new power plants in a timely manner. Nuclear vendors are focused on delivering state-of-the-art reactor designs, including fuel designs. Expertise in engineering, procurement, and construction is provided for by companies with relevant industrial plant construction experience, which includes open-top construction. Companies with adequate knowledge in nuclear regulation and have industry certifications such as ASME, ACI, IEEE are required. The opportunities for new suppliers to be involved in SMR industry could be developed through human resource development as well as company certification. In addition, financial stability of suppliers/contractors and delivery reliability of modules are issues to be considered in long-term supply chain management.

By developing a SCOR framework and describing the supply chain processes and partnerships in the SMR industry, Missouri shows a business climate for supporting the nucleation of an SMR economy. The role of Missouri as a global supply base is well supported by the established infrastructure for manufacturing and logistics. The availability of talent and good connectivity is promising for SMR supplier development and the ability to deploy modules efficiently. Previous experience in nuclear construction provides an invaluable understanding of Missouri's capabilities and forms the foundation for future potential supply base. SMR technology provides new opportunities for Missouri to diversify its economy and create export opportunities. Further details are needed to evaluate the feasibility of the SMR supply chain through specific performance criteria that can be evaluated through the SCOR framework.

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3. CONCLUSION

The main goal for this research was to support power uprate of the Missouri S&T reactor, and to study natural convection as a cooling mode in a reactor core. Natural convection based passive safety system is also one of the major attractions in small modular reactor technology. The work reported in the CFD analysis sections followed two main paths; the development of a thermal-fluid model of the MSTR by modeling a “unit cell” of the core representing the hottest channel, and the subsequent modeling of the MSTR core through porous media approach. The models were developed and analyzed using a commercial computational fluid dynamics code, STAR-CCM+. The research goal of this analysis is to develop CFD models that provide predictions of thermal-fluid parameters in the MSTR.

The unit cell consists of two parallel fuel plates and a coolant channel through which heat is removed from the unit cell into a coolant volume. In a preliminary study, a parallel-plate model was developed (Papers I and II). First, a constant heat flux was applied on the fuel plate, and the simulation results compared with the results where non-uniform heat flux was used as a boundary condition. The CFD analysis show that constant heat flux provides a prediction of average coolant temperature, however, does not entirely capture the effects of varying heating source. The MSTR core is characterized by a classical cosine-shaped heat flux that has maximum heating at the core center. There is a small amount of heating retardation away from the center of the fuel plate and translates to relatively slower heating in the channel. The parallel-plate model predicted both temperature and velocity fields for the MSTR core configuration of 120W. CFD simulations for the parallel-plate model were performed at four power levels i.e.

200kW, 100kW, 60kW and 20kW. The temperature fields show a linear increase, and density field decrease linearly with power. The parallel-plate model was also used in determining porous coefficients for the MSTR core under natural convection conditions. Since modeling the entire core is computationally intensive and cost prohibitive, a strategy of developing an MSTR model that minimized the details yet retained the heat transfer features was decided. To this end, modeling the core as a porous media was achieved. Darcy's Law is expressed in terms of flow velocity and the pressure gradient in the porous medium, and permeability. The parallel-plate model predicted velocity field and pressure drop in the channel. Darcy's Law is valid for seepage flow in porous media; for flow that exceeds Reynolds number by unity, it over predicts the actual fluid motion. Thus, the Forchheimer's equation was used whereby it takes into account the boundary and inertial effects that was neglected in Darcy's Law due to the small porosity associated with the medium. From the simulation results, the inertial and viscous resistance tensors were found to be were found to be 281005 kg/m^4 and 7121.6 kg/m^3 respectively.

In the second model, a volume representation of the MSTR pool and core was developed. This model consists of a third of the MSTR pool along with three fuel elements at power density 1.86E+6 Wm^{-3} (Paper III). The fuel section was replaced by porous region by using previously determined porous parameters and porosity 0.702 from the parallel-plate model. Temperature measurements were carried out at three locations within the reactor pool and at three power levels 200kW, 100kW and 10kW. The measurement procedure and analysis for coolant temperature measurements are presented in Appendix A, B, and C. Model validation was successfully performed for this MSTR model. This model was then used to predict the heat removal capacity through an active cooling system. An eductor (outlet) and a pipe (inlet) are part of the cooling system, and they were ‘turned on’ for this third simulation case. This cooling system was installed as part of the reactor upgrade for the MSTR. Flow field in the reactor pool were obtained with the new active cooling system operated at 35% pumping capacity. The simulation results for the MSTR model showed that water is drawn out of the pool at a rate of 1.28kgs^{-1} from the 4” pipe, and predicted a surface temperature of the pool not exceeding 30°C. It was found that the porous parameters were adequate for use in replacing the MSTR core with a porous region, and to investigate coolant flow inside the reactor pool. The CFD simulation results provided thermal-fluid parameters for normal operations and baseline parameters for supporting license renewal as well as power uprate plans.

Because the reactor is designed to use natural convection as the mechanism to remove heat, in the event of future reactor power uprate, additional removal system will need to address the higher pool temperature rise without causing the reactor to shut down due to the negative temperature coefficient. In this case, an active heat removal system was installed that has a 4” inlet pipe with a 6” dia. head that takes in the pool water, goes through a heat exchanger, cooled down and water is returned to the pool through the use of three eductors.

In the final part of this research, a focus on the small modular reactor technology had produced a new Missouri small modular reactor supply chain model. Small modular reactors (SMRs) are the future of advanced light water nuclear reactor. It has been designed to utilize passive safety system for safer reactor operations. The cooling of the entire SMR core can be done through natural convection during emergency. The reactor will be able to achieve safe shutdown without any electrical power or the need for operator actions. Because SMRs are modular by design, the SMR modules are factory-built, and the construction period is expected to be shorter. The goal of this phase of the research was to evaluate the status of the supply chain in the Midwest in general and in the state of Missouri in particular.

While SMR has lower investment cost due to SMR capacity can be built in increments, but there remains a question of the sustainability of the back-end supply chain. Several SMR construction and manufacturing issues were discussed in Paper IV. An SMR supply chain model for the state of Missouri was created based on the supply chain operations reference (SCOR) framework. This model allows quantification of key performance issues and identifies key growth areas in establishing an SMR operation.

APPENDIX A.

TEMPERATURE MEASUREMENTS

A. TEMPERATURE MEASUREMENT

This section describes the methods used to obtain temperature measurements in the MSTR pool. The purpose of this experiment is:

- To obtain temperature distribution of the MSTR reactor pool at 10kW, 100kW and 200kW
- To obtain temperature distribution of the heat plumes at 100kW and 200kW

Temperature measurements were performed using Type K thermocouples. These are a generic type that is commonly used for measuring temperature in −200 °C to +1350 °C / -330 °F to +2460 °F range. A thermocouple tree (TC-tree) consisting of 17 thermocouples attached to a half-inch PVC pipe was extended into the MSTR pool to obtain temperature measurements. The pipe is about 29 feet long, and wires connected to the thermocouples are securely wrapped around the pipe. The thermocouple is arranged so that it is 1 feet apart from each other for thermocouple #1 to #8 and 2 feet apart for thermocouples #9 to #17. Temperature readouts are taken from FLUKE 54 II Thermometer reader. At the end of the pipe, there is a notch that is used to set it on the fuel element or the grid plate. From the measurements, vertical temperature distributions were obtained. Two locations, C9 and D3, were selected at the periphery of the core and the location F14 is above the center of the core (Figure 1). Details of this measurement process are given in Appendix B.

	1	2	3	4	5	6	7	8	9
A									
B						Source			
C					C4	F5	F1	F17	**C9** **X**
D			**D3** **X**	F4	F8	**F14** **X**	C1	F10	F2
E				F9	C3	F12	C2	F7	F3
F				CRT	F15	HC	F13	BRT	F6

Figure 1. Core map with the selected locations for MSTR pool temperature measurements

Results and discussion

The TC-tree was lowered into the pool at locations D3, F14 and C9 (Figure 1). Temperature measurements were recorded, and were plotted against elevation to show coolant temperature variation in the vertical direction. To obtain an accurate temperature distribution at both D3 and C9 positions, the TC-tree was aligned as closely as possible to fuel assemblies F4 and F2 respectively (Figures 2 and 3). This alignment is made so that the temperature changes along the vertical fuel plate can be captured. At position F14, the TC-tree was placed right above the core to obtain heat plume measurements (Figure 4). The total time taken to make continuous measurements was approximately 7 hours, and the reactor was operated in succession by several Student Reactor Operators. During this time the reactor was taken to power gradually from 10kW, 100kW to 200kW.

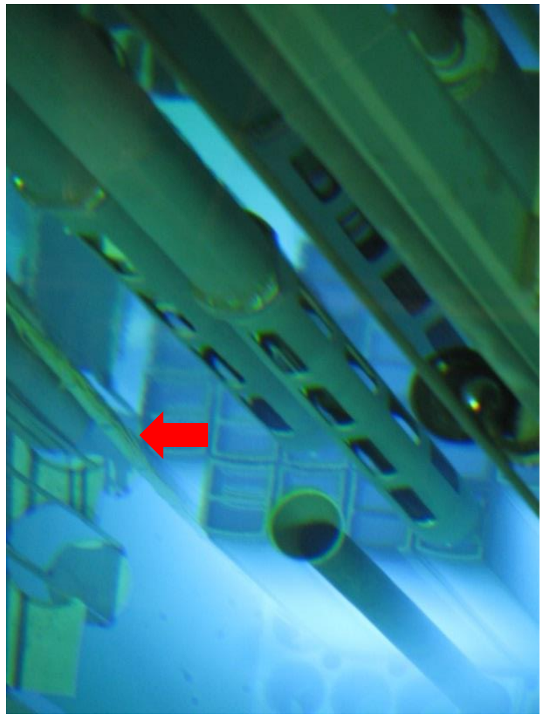

Figure 2. Temperature measurement: TC-tree at position C9

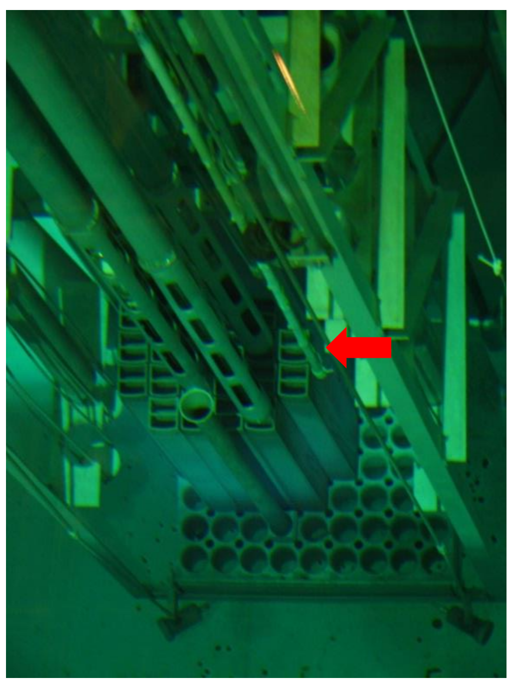

Figure 3. Measurement of heat plume: TC-tree at position D3

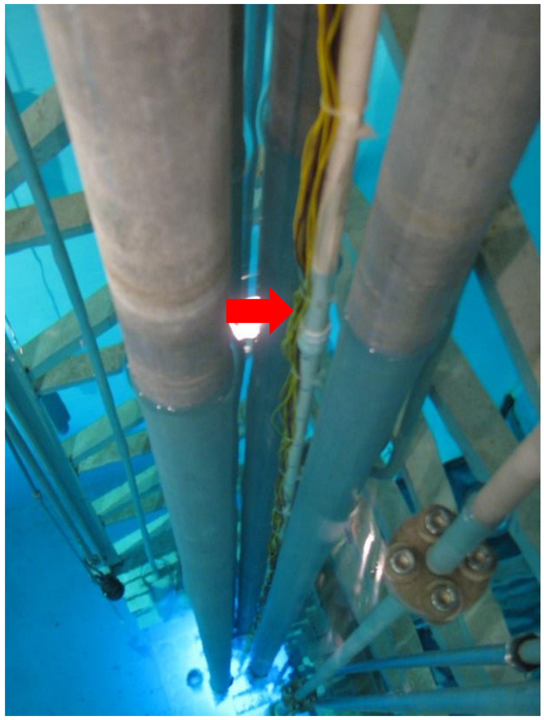

Figure 4. Measurement of the heat plume: TC-tree at position F14

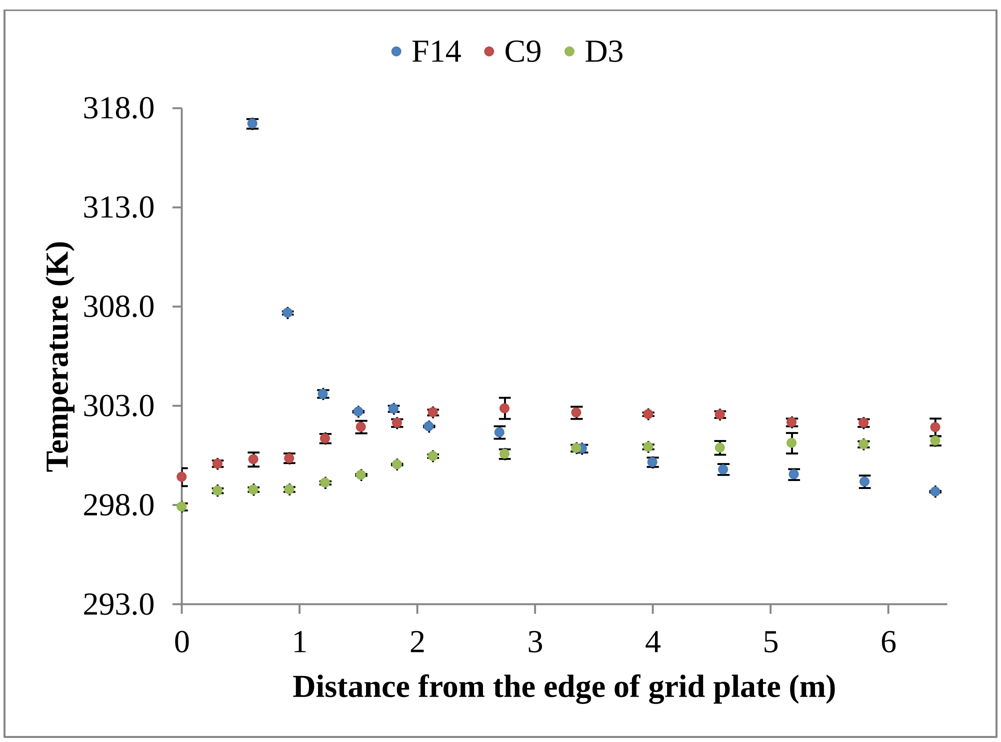

Figure 5. Temperature distribution in locations F14, C9 and D3 at 200kW

It was also noted that the introduction of the TC-tree into the pool had slight effect on the reactor power level whereby the reactor was taken to a higher power than 200kW to counter the reactivity effect.

The results of the temperature measurements at location F14, C9 and D3 are shown in Figure 5. In the core map, location F14 is approximately the center of the MSTR core. Location C9 is a position surrounded by fuels F17 and F2. Location D3 is in the periphery of the core and is adjacent to fuel F4. The coolant temperature rise in F14 is the largest, followed by temperatures in locations C9 and D3. These temperature distributions showed the expected trends, whereby the highest temperature corresponds to the center of the core where the highest flux is located. At C9, the fuels F17 and F2 contributes to a higher temperature rise compared to peripheral temperatures at location D3. The F14 data showed that there is a relatively large drop of temperature (14K or °C) from 317K (44°C) to 303K (30°C). This drop is seen from the top of the core to a distance 1.2 meters away from the core top. At 3.5 meters above F14, the coolant temperature was recorded to be between 298K (25°C) and 300K (27°C). The data suggest that the upward convective flow is strongest at F14, and coolant mixing starts approximately 1.5 meters away from the top of the core. As the heat flow from F14 slows-down, the coolant temperature increases between 300K (27°C) and 303K (30°C) in locations C9 and D3.

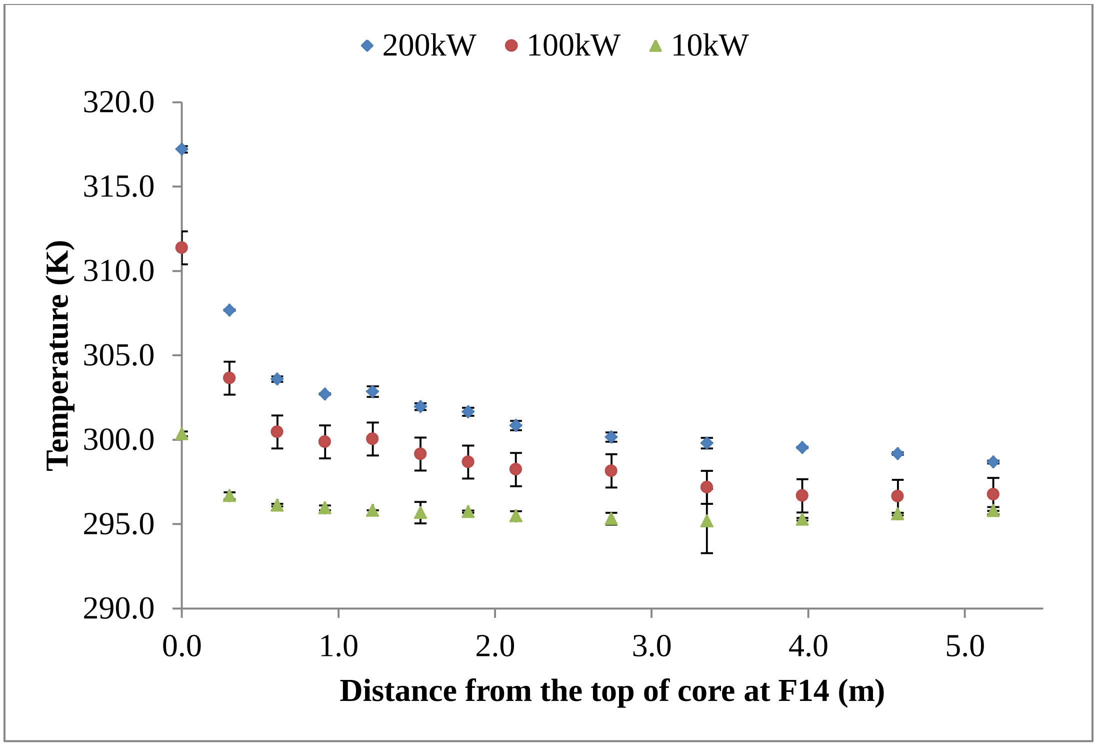

Figure 6. Temperature distribution at 200kW, 100kW and 10kW above the core center

Figure 6 shows the F14 temperature distributions at three power levels, i.e. at 200kW, 100kW and 10kW. The highest temperature recorded is at the position closest to the core; at 200kW, 100kW, 10kW the values are approximately 317K (44°C), 311K (38°C) and 300K (27°C) respectively. The heat dissipates, and coolant temperature comes to equilibrium at about 4.5 meters above the core.

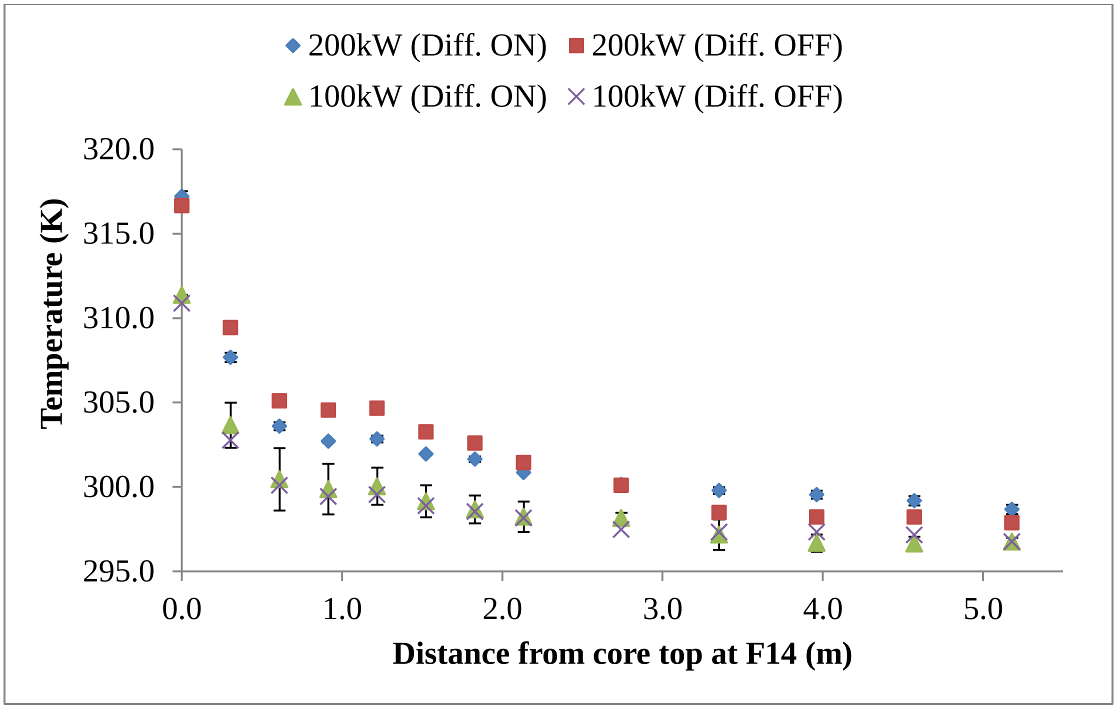

Figure 7. Diffuser effect in location F14

Figure 7 shows the effect of diffusers on the heat flow at location F14. In general, coolant temperatures are expected to be higher when the reactor is operated at 200kW compared to 100kW. The data shows that this expected trend was followed. There are two N-16 pumps (diffusers) located above the core to blow down the surface water and delay the rise of N-16. The N-16 is produced in the water passing through the core by the O-16 (n,p) N-16 reaction. The half-life of N-16 is about 7 seconds. The diffusers are located between the top of the core and the pool surface. There is about 20 feet (6 m) of water from the core top to the surface, and diffusers are located at about 10 feet (3 m) from the core top. The graph shown in figure 7 suggests that the diffusers do not affect the convective flow at locations 3 meters above the core top.

APPENDIX B.

TEMPERATURE MEASUREMENT PROCEDURE

B. TEMPERATURE MEASUREMENT PROCEDURE

Purpose:

1. To obtain temperature distribution of the MSTR reactor pool at __________kW
2. To obtain temperature distribution of the heat plumes at __________kW

Preparation and precautions:

1. Check the batteries of the thermocouple reader - FLUKE 54 II Thermometer. Check functions of reader (Note: instantaneous and average temperature options are available). The reader can read values from two thermocouples (in °C, °F, and K).
2. Thermocouple Type K is used for this measurement.
3. Wear latex gloves when handling the thermocouple tree to prevent contamination to the thermocouples.
4. The numbers on the tree indicate the distance (in feet): zero (0) is at the bottom end, and 25 is at the top end.
5. Move the tree in slow and deliberate manner to prevent unnecessary bending while it is in the water.
6. The N-16 diffuser (water pump) is switched on when reactor operates at 20kW and above. This is done to delay the escape of N-16 into the reactor bay by allowing decay to take place in the pool.

Procedure (Purpose 1):

1. Check core map, and locate the desired measurement locations i.e. periphery of the core.
2. Connect cables from the thermocouple to the reader.

3. Locate the first measurement point, and position the tree.
4. Wait between 10 and 15 minutes to allow for water mixing then start taking readings.
5. Record the temperature reading shown on the reader. Take an average in 1 minute if fluctuations occur.
6. Repeat steps 1 to 5 for other measurement locations.
7. Tabulate and graph the temperature measurements against distance.
8. Obtain the Hourly Operating Log to monitor the Core Inlet Water Temp (°F)(Item 23)

Procedure (Purpose 2):

1. Check core map, and locate the desired measurement locations i.e. above the core.
2. Connect cables from the thermocouple to the reader.
3. Locate the first measurement point, and position the tree.
4. Wait between 10 and 15 minutes to allow for water mixing then start taking readings.
5. Record the temperature reading shown on the reader. Take an average in 1 minute if fluctuations occur.
6. Repeat steps 1 to 5 for other measurement locations.

APPENDIX C.

TEMPERATURE DATA SHEET

C. DATA SHEET

Power Level: ____________ kW

THERMOCOUPLE	LOCATION								
°C / °F / K	A6	B5	B8	D3	D10	G5	G8		
#25									
#23									
#21									
#19									
#17									
#15									
#13									
#11									
#7									
#6									
#5									
#4									
#3									
#2									
#1									
#0									

APPENDIX D.

HEAT TRANSPORT

D. HEAT TRANSPORT

Nuclear fission produces a large amount of heat in the reactor core, and the heat transport out of this core is a key aspect of thermalhydraulic analysis. The design and operations of a reactor is governed by the thermal limits on the fuel temperature and material. Thermalhydraulic behavior of a nuclear reactor is well described by heat transport theory whereby heat is transferred by three methods: conduction, convection, and radiation. Heat is conducted from the fuel to the cladding, and convected to the coolant.

Natural convection is often considered a challenging problem due to the complexity of interaction between buoyancy, gravity and density gradient in the flow field as well as the influence of pressure changes during heat transfer. Pressure drop can be caused by resistance to flow, changes in elevation, density, flow area, and flow direction. Temperature changes are most prominent in the boundary layers near the wall. modify the coolant behavior in a nuclear reactor.

Steady State Heat Transport

Fuel element temperature distribution, T= T(x, y, z) is determined by solving the heat equation. Fourier's Law allows the determination of conduction heat flux from the knowledge of temperature distribution of the medium. The heat flux, q_x'' in x direction is given by equation (1).

$$q_x'' = -k\frac{dT}{dx} \tag{1}$$

The general form of the heat equation is given in equation (2) and in concise form equation (3) below. The net transfer of thermal energy into a control volume, dV and thermal energy generation, $q'' = q''(x, y, z)$ is balanced with the change in thermal energy storage (Figure 1). Thermal diffusivity, α (m²/s) is a property of the fuel that describes how much it conducts relative to its ability to store thermal energy. Materials with high thermal diffusivity rapidly adjust their temperature to that of their surroundings due to quick heat conduction in comparison to their volumetric heat capacity.

$$\frac{\partial}{\partial x}\left(k\frac{\partial T}{\partial x}\right)+\frac{\partial}{\partial y}\left(k\frac{\partial T}{\partial y}\right)+\frac{\partial}{\partial z}\left(k\frac{\partial T}{\partial z}\right)+q''=\rho c_p\frac{\partial T}{\partial t} \quad (2)$$

$$\nabla^2 T + \frac{q''}{k}=\frac{1}{\alpha}\frac{\partial T}{\partial t} \quad (3)$$

where $\alpha=\frac{k}{\rho c_p}$

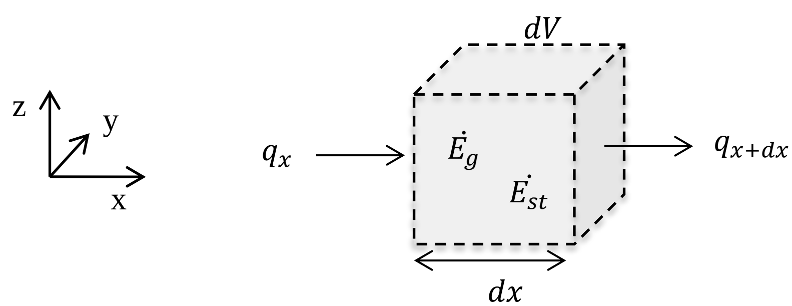

Figure 1. Differential control volume through which energy transfer is by conduction in x-direction

Applying Fourier's Law and the principle of conservation of energy to a differential control volume, the one-dimensional steady state heat conduction equation in a fuel slab can be written as in equation (4).

$$q_x''' = -\mathrm{k}\frac{\mathrm{d}^2\mathrm{T}}{\mathrm{dx}^2} \quad (4)$$

The heat flux, q_x'' (W/m^2) is in the positive x-direction. The temperature gradient, dT/dx (K/m) notation is in the direction of heat flow whereby conduction occurs in the direction of decreasing temperature. The proportionality constant k is the thermal conductivity of the fuel material (W/m-K). The volumetric heat source, q_x''' is written for the slab's heat transfer in the x-direction in Cartesian coordinates. The solution for the fuel temperature along x-axis is obtained by applying the boundary conditions $\left.\frac{dT}{dx}\right|_{x=0} = 0$ and $T = T_c$ at x $= \frac{s}{2}$ (Figure 2).

$$T(x) = \frac{q'''}{2k}\left(\frac{s^2}{4} - x^2\right) + T_c \qquad \text{for } 0 \leq x \leq s/2 \qquad (5)$$

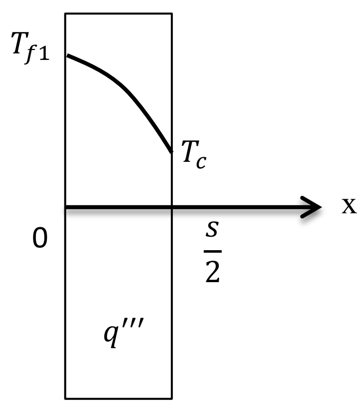

Figure 2. Heat conduction for a fuel slab with thickness, t = s/2

In convection process, the Newton's Law of Cooling describes the heat transferred from the surface of the fuel (cladding) into the coolant (Equation 4). The equation notations are heat transfer coefficient, h (W/m^{2}K), fuel surface temperature, T_s (K), coolant temperature, T_∞(K), and surface area, A (m^2).Combining equation 2 and 5 we obtain the heat transfer coefficient, h (W/m^2K) in equation 6. Several dimensionless parameters were used to estimate heat transfer from the fuel plates to coolant (Table 1).

$$q = hA(T_s - T_\infty) \quad (6)$$

$$q" = h(T_s - T_\infty) \quad (7)$$

$$h = \frac{-k\left(\frac{\partial T}{\partial x}\right)_{y=0}}{T_0 - T_\infty} \quad (8)$$

Flow in the fuel assembly could be described with Navier-Stokes equation and continuity equation. With the assumptions of incompressible fluid, applying boundary layer treatment and considering two dimensional flow in x and y directions; the continuity, momentum and energy equations for the steady state, laminar natural convection along a vertical flat plate (Figure 3) are given in the following governing equations.

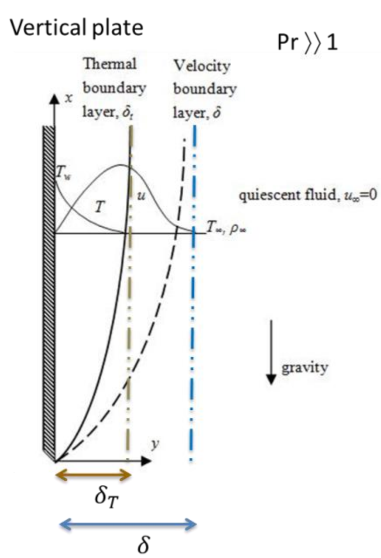

Figure 3. 2D, steady-state, laminar natural convection from a vertical plate

$$\frac{\partial u}{\partial x}+\frac{\partial v}{\partial y}=0 \tag{9}$$

$$u\frac{\partial u}{\partial x}+v\frac{\partial v}{\partial y}=g\beta(T-T_{\infty})+\nu\frac{\partial^2 u}{\partial x^2} \tag{10}$$

$$u\frac{\partial T}{\partial x}+v\frac{\partial T}{\partial y}=\alpha\frac{\partial^2 u}{\partial y^2} \tag{11}$$

$$u(x,0)=v(x,0)=0 \tag{12}$$

$$T(x,0)=T_w(x) \tag{13}$$

$$\frac{\partial T(x,0)}{\partial y}=-\frac{q_w(x)}{k} \tag{14}$$

Calculations of dimensionless parameters such as those listed in Table 1 are commonly used in heat transfer problems. Rayleigh number for an isoflux plate was used in this work to estimate the flow type in the MSTR channels (Paper II and III). There are several detailed treatments for various plate-type problems and these are not repeated here. The following references by S. Ostrach (1952), E. Eckert (1950), A. Bejan (2013) and Incropera and Dewitt (1996) are useful texts to understand laminar free convection flow and heat transfer, and were used to set up hand calculations for the MSTR models. Advanced mathematical treatments on heat convection are found in books by L. M. Jiji (2009) and Je-Chin Han (2012). Several papers with non-uniform heat flux boundary conditions were referenced for understanding and explanation for use in the case of MSTR's cosine-shaped heat flux (Lee & Yovanovich, 1991;Pantokratoras, 2003; Roeland et. al., 2014).

Table 1. Relevant thermal and flow parameters

Free stream pressure gradient	$\frac{dp_\infty}{dx} = -\rho_\infty g$
Thermal expansion coefficient	$\beta = -\frac{1}{\rho}\left(\frac{\partial\rho}{\partial T}\right)_p = -\frac{1}{\rho}\frac{\rho_\infty - \rho}{T_\infty - T}$ $\beta = -\frac{1}{\rho}\left(\frac{\partial\rho}{\partial T}\right)_p = \frac{1}{\rho}\frac{p}{RT^2} = \frac{1}{T}$
Boussinesq approximation	$(\rho_\infty - \rho) \approx \rho\beta(T - T_\infty)$
Grashof number(Isothermal Plate)	$Gr_L \equiv \frac{g\beta(T_s - T_\infty)L^3}{\upsilon^2}$
Critical Rayleigh number	$Ra_{x,c} = Gr_{x,c}Pr = \frac{g\beta(T_s - T_\infty)x^3}{\upsilon\alpha} \approx 10^9$
Average Nusselt number	$\overline{Nu_L}$ $= \left\{0.825 + \frac{0.387Ra_L^{1/6}}{[1 + (0.492/Pr)^{9/16}]^{8/27}}\right\}^2$
Modified Rayleigh number(Isoflux Plate)	$Ra* = \frac{g\beta S^4 q''}{k\alpha\upsilon}$
Nusselt number (Isoflux Plate)	$\overline{Nu_L} = \left\{0.825 + \frac{0.387\left(Ra_L^{1/6}\right)}{\left\{1 + \left\{\frac{0.492}{Pr}\right\}^{9/16}\right\}^{8/27}}\right\}$
Reynolds number	$Re = \frac{\rho VD}{\mu}$

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