Thermal-Hydraulic System Analysis of a Proposed 1 MWth Nuclear Gas Cooled Microreactor

Aaron S. Fernandez

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THERMAL-HYDRAULIC SYSTEM ANALYSIS OF A PROPOSED 1 MWth NUCLEAR GAS COOLED MICROREACTOR

by

Aaron S. Fernandez

Bachelor of Science
United States Military Academy, 2011

Submitted in Partial Fulfillment of the Requirements
For the Degree of Master of Science in
Nuclear Engineering
College of Engineering and Computing
University of South Carolina
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Accepted by:
Travis W. Knight, Director of Thesis
Elwyn Roberts, Reader
Jamil A. Khan, Committee Member
Tracey L. Weldon, Interim Vice Provost and Dean of the Graduate School
DEDICATION

This thesis is dedicated to my mother Beatriz Fernandez, my wife Jennifer Fernandez-Vasquez, and my son Gabriel Fernandez-Vasquez. Thank you for your constant support and faith in me.
ACKNOWLEDGEMENTS

Thanks to Dr. Travis Knight for providing personal and academic guidance, directing, and reviewing this research. I want to thank my colleagues at the University of South Carolina for their effort in giving insights and suggestions to my work, namely: Kyle A. Gable, Adam Stephen, and A.S.M Fakhru1 Islam. Finally, I would like to thank the SAM Team at Argonne National Laboratory for the guidance and troubleshooting help with this simulation.
ABSTRACT

Nuclear power plants have historically been large, expensive, and prolonged construction projects. A new generation of reactors categorized as microreactors is currently being designed to address energy needs in remote locations. The University of South Carolina has proposed the MIcro Nuclear In ONe megawatt (MINION-1) design to achieve ten years of operation. The primary loop transfers 1 MW of energy from the reactor core using helium, while the secondary uses air in an open Brayton cycle to produce power and heat. This work focuses on modeling the MINION-1 design using the System Analysis Module (SAM) developed by Argonne National Labs (ANL) and conducting a complete plant analysis. A unit-cell approach was used in the state-of-the-art System Analysis Module (SAM) application, which simulates the MINION-1 steady-state and using resulting boundary condition outputs to calculate overall system efficiency. The steady-state exit temperature of the core is 664 K (391 °C) and a maximum fuel temperature of 1611 K (1338 °C). The net amount of work produced by the system is 237.71 kW resulting in an overall efficiency of 23.77%.
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LIST OF SYMBOLS

A  Cross sectional flow area.
Cp  Specific heat
Dh  Hydraulic diameter
f  Friction factor
h  Heat Transfer Coefficient
h_s  Enthalpy of a state on the secondary side
k  Heat conductance
ṁ  Mass flow rate
η  Efficiency
Nu  Nusselt number
ρ  Density
Pr  Prandtl Number
Pr_s  Relative pressure
Qth  Thermal Power
R  Radial distance
Re  Reynolds Number
T  Temperature
μ  Dynamic viscosity
v  Velocity
Ẇ  Rate of work (Power).
<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Definition</th>
</tr>
</thead>
<tbody>
<tr>
<td>AGR</td>
<td>Advance Gas Reactor</td>
</tr>
<tr>
<td>ANL</td>
<td>Argonne National Laboratory</td>
</tr>
<tr>
<td>BeO</td>
<td>Beryllium Oxide</td>
</tr>
<tr>
<td>CO₂</td>
<td>Carbon Dioxide</td>
</tr>
<tr>
<td>DoD</td>
<td>Department of Defense</td>
</tr>
<tr>
<td>DOE</td>
<td>Department of Energy</td>
</tr>
<tr>
<td>FEM</td>
<td>Finite Element Method</td>
</tr>
<tr>
<td>GFR</td>
<td>Gas-Cooled Fast Reactor</td>
</tr>
<tr>
<td>He-4</td>
<td>Helium</td>
</tr>
<tr>
<td>HTGR</td>
<td>High Temperature Gas Reactor</td>
</tr>
<tr>
<td>HX</td>
<td>Heat Exchanger</td>
</tr>
<tr>
<td>IAEA</td>
<td>International Atomic Energy Agency</td>
</tr>
<tr>
<td>INL</td>
<td>Idaho National Laboratory</td>
</tr>
<tr>
<td>LFR</td>
<td>Lead-Cooled Fast Reactor</td>
</tr>
<tr>
<td>LWR</td>
<td>Light Water Reactor</td>
</tr>
<tr>
<td>MiiHTR</td>
<td>Micro Modular High-Temperature Reactor</td>
</tr>
<tr>
<td>MINION-1</td>
<td>Micro Nuclear In ONE megawatt</td>
</tr>
<tr>
<td>MOOSE</td>
<td>Multiphysics Object Oriented Simulation Environment</td>
</tr>
<tr>
<td>NEAMS</td>
<td>Nuclear Energy Advanced Modeling and Simulation Program</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressure Water Reactor</td>
</tr>
<tr>
<td>RELAP</td>
<td>Reactor Excursion and Leak Analysis Program</td>
</tr>
</tbody>
</table>
RPV ................................................................. Reactor Pressure Vessel
SAM ................................................................. System Analysis Module
SCWR ............................................................... Supercritical water reactor
SFR ................................................................. Salt Fast Reactor
SiC ................................................................. Silicone Carbide
SNAP .............................................................. Symbolic Nuclear Analysis Package
TRACE ........................................................... TRAC/RELAP Advanced Computation Engine
UC ................................................................. Uranium Carbide
UofSC ............................................................. University of South Carolina
CHAPTER 1

INTRODUCTION

Nuclear energy has played a significant role in meeting the world’s energy demands but has received lackluster support due to the negative perceptions of nuclear energy. In recent history, support for nuclear energy in the U.S. has grown significantly. Experts believe nuclear energy will play a critical role in meeting the growing energy demands of the future and reducing net carbon emissions. The international nuclear community has developed fourth-generation nuclear reactor concepts to increase safety efficiency and decrease proliferation.

The University of South Carolina (UofSC) has proposed the core concept of MIcro Nuclear In ONe megawatt (MINION-1). The MINION-1 concept is designed to be deployed to remote locations to supply electricity and heat. To meet these criteria MINION-1 concept contains a ten-year core life without the need for refueling, no on-site spent fuel storage requirement, and a passive heat removal system for accidental conditions providing inherent safety. The primary challenge of designing the MINION-1 is configuring an optimized core that achieves these strict requirements. Despite using traditional UO₂ fuel and a well-researched prismatic core design, combining all required features will require the combination of several Generation IV nuclear reactors. To assess this design, an understanding of multiphysics and an effective set of computational tools with a sophisticated methodology is required.
The purpose of this research is to build a thermal-hydraulic model of the MINION-1 core to determine performance during steady-state conditions to allow future optimizations using state-of-the-art simulation programs built by the Argonne and Idaho National Laboratories (ANL and INL). This paper presents the concepts of the MINION-1 reactor core, fuel unit cell, primary core heat removal loop, power cycle, simulation methodology, and thermal hydraulic performance and analysis. The passive heat removal system is not modeled or analyzed in this work.

Thermal-hydraulics is a major discipline within the nuclear engineering field and directly tied to multiple physical phenomena in a nuclear reactor design including neutronics, mechanics, and radiation shielding. The thermal-hydraulics analysis of nuclear reactors has been commonly accomplished using system codes. These system codes would calculate fluid flows within intricate piping and components of a nuclear reactors forming a virtual thermal hydraulic system capable of producing data simulating performance. The codes resulting data can be analyzed and then directly applied to multiple aspects of reactor design.

A reactor’s thermal hydraulic performance has a direct correlation to neutronic profiles. For instance, as temperature in the core changes so do material microscopic cross sections therefore changing neutronics. Any resulting change in neutronics could result in a change in necessary shielding. Thermal ranges also influence material selection based on a material’s ability to withstand resulting temperatures. When these disciplines are combined the foundation to build a reliable system is developed. Finally, overall plant efficiency can be determined from an analysis of the data, which gives a sense of system feasibility and a comparable start point to optimize power plant design. This work will
build a tool to conduct thermal-hydraulic analysis for MINION-1 but most importantly creates the foundation to guide MINION-1 design.
CHAPTER 2

LITERATURE REVIEW

The allure of microreactors is simple; they can potentially yield higher efficiencies than commercial reactors due to higher theoretical operating temperatures and designed to be highly mobile using the current military and civilian transportation infrastructure. Surprisingly, the concept of mobile nuclear power plants is almost as old as nuclear science itself. In 1939, the Naval Research Laboratory proposed nuclear power for submarine propulsion, resulting in the commission of the world’s first nuclear-powered ocean-going vessel, the USS Nautilus. The pressurized water reactor (PWR), designated S2W, produced 13,400 horsepower (9.86 megawatts thermal) and “enormously increase[d] the range and military effectiveness of the submarine” [1, p. 5]. Although small reactors are conceptually old, the non-nautical implementation has not significantly impacted.

The U.S. Department of Defense (DoD) explored the utilization of ground-based small and mobile nuclear power plants during the 1960s, building eight reactors as proof of concepts that ran successfully but were ultimately defunded [2, p. 2.1]. The Atomic Energy Commission concluded that the pace of the nuclear power program could not achieve its goal at a reasonable cost resulting in defunding and ultimate discontinuation of any nuclear power research for U.S. military ground forces [3]. Despite successful results from small reactors, commercial interest focused on large-scale projects to provide power to large, concentrated populations.
In recent years the concept of microreactors has gained significant interest and funding. The origin of this worldwide interest can be traced to President Obama’s call to “harness the power of nuclear energy….to combat climate change and to advance opportunity for all people” [4]. Soon after, reducing greenhouse gas emissions became a priority for the U.S. and international partners, and nuclear energy became a favored technology in the fight against climate change. Nuclear energy did not become favored in the U.S. overnight; in fact, it took over 31 years for U.S. public perception of nuclear energy to become majority positive. After the Three Mile Island (TMI) incident in 1979, public surveys revealed that only 49% of the public accepted nuclear technology in 1983, while a similar survey revealed 74% public acceptance in 2010 [5, p. 11]. The change in public opinion and the international call to fight climate change generated the ideal conditions for nuclear power to become a renewed focus. With funding available, the U.S. DoD, in conjunction with the DOE, conducted a study in 2011 that identified a need for small nuclear power plants (producing less than 300MWe) to build assurance in the power grid, reduce dependency on fossil fuels, and reduce greenhouse gas emissions [5].

Through DOE and the DoD collaborative studies, interest in mobile reactors has renewed for military and non-military purposes using technology developed from fourth-generation reactor research. Specifically, microreactors have gained national and international attention for their ability to provide energy to previously inaccessible consumers. According to the U.S. DOE, microreactors are identified by having three characteristics: 1. factory-fabricated, 2. transportable, and 3. self-adjusting [6]. Although
microreactor designs vary, according to surveys conducted by the International Atomic Energy Agency (IAEA), they will have a capacity to produce less than 20 megawatts of electricity [7, p. 4].

Microreactor designs focus on utilizing modern fuels, coolants, and materials to increase safety, efficiency, sustainability, and reduce waste by using aspects from fourth-generation reactor concepts. Fourth-generation reactor concepts fall into six categories: Gas-cooled fast reactor (GFR), Lead-cooled fast reactor (LFR), Molten salt reactor (MSR), Supercritical water reactor (SCWR), Sodium-cooled fast reactor (SFR), Very-high-temperature reactor (VHTR) [8, p. 3]. Gas-cooled designs have been the least developed for relatively low power applications of these next-generation reactors, allowing room for new prospects.

In September 1944, Dr. Farrington Daniels conceived a helium-cooled reactor moderated by BeO, which could achieve significantly higher temperatures than light water reactors. Although the Daniels Pile project was eventually canceled, it created a foundation for future research utilizing He-4 as a coolant and BeO as a moderator [9]. Five gases have been used to cool reactor cores: air, helium, carbon dioxide, hydrogen, and nitrogen. Using air could cause fires when a core was moderated by graphite; helium is relatively expensive, hydrogen could potentially ignite, so research dwindled to CO₂ and nitrogen. In the United States, interest in mobile reactor power led to the experimentation with nitrogen coolant in the Gas-Cooled Reactor Experiment (GCRE-1), leading to the U.S. Army’s Mobile Low-Power Reactor (ML-1), which never proceeded
past the prototype stage due to budget constraints [10, p. 22]. Magnox and Advanced Gas-Cooled Reactors (AGRs) reactors were developed using CO₂ coolant. The Magnox and AGR were dual purpose-built to produce plutonium for national nuclear weapons programs and to produce electricity from recaptured heat. The design of a reactor and its thermodynamic abilities directly result from design purpose; therefore, early gas-cooled designs were limited in their power-producing potential due to their dual-purpose nature [10, p. 2]. Helium became a natural improvement to the use of CO₂ due to its unique noble gas properties, which allows a reactor to run at much higher temperatures without causing oxidation, leading to HTGR designs.

The Peach Bottom Unit 1 and Fort St. Vrain power plants are the only two HTGR style reactors constructed on U.S. soil, both serving as prototypes and providing lessons learned for the nuclear industry. The lessons from these experiments led to a simplification of the HTGR design; eliminating any need for the presence of water, minimizing the core size for its needed application, and favoring a steel reactor vessel over a prestressed concrete reactor vessel due to leaking [11, p. viii]. Although gas-cooled reactor experiments proved to have the potential to supplement the U.S.’s primarily LWR fleet, national disinterest in nuclear energy and, therefore, to defund of research limited any significant development of gas-cooled reactors in the U.S. Not until the proposal of fourth-generation nuclear reactors did gas-cooled reactors garner national and international attention again. Ventures in designing a modern helium-cooled reactors have been conceptualized in recent years; two designs of interest are the Micro Modular
High-Temperature Reactor (MiHTR) and the Gas Turbine – Modular Helium Reactor (GT-MHR).

Both the MiHTR and GT-MHR designs are considered modern modular helium-cooled reactors that meet the Generation IV design criteria, indicating that modern materials and concepts are used to maximize fuel utilization and safety. Table 2.1 outlines primary features of interest in comparison to the MINION-1 design. The GT-MHR is not designed to be mobile but modular, meaning that multiple GT-MHRs could be used synchronously to generate higher levels of power but require a significantly smaller footprint compared to a conventional reactor with equivalent power output. In comparison, the MiHTR is intended to be deployable to remote locations and requires a modest amount of maintenance during its core lifetime.

MINION-1 is akin to the MiHTR in concept but smaller, increasing its compactness and mobility. Across the three reactors, helium is utilized as the core cooling fluid, which results in some similar design features. For example, to remove heat effectively from the core using gas, turbulence through the core must be increased to increase heat transfer. The hydraulic diameter, which directly affects turbulence, of each core design is similar due to maximizing coolant mass flow and turbulence based on the size of each respective core.
Table 2.1: Comparison data between MINION-1 [12], MiHTR [13], and GT-MHR [14]

<table>
<thead>
<tr>
<th></th>
<th>GT-MHR</th>
<th>MiHTR</th>
<th>MINION-1</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Thermal Power</strong></td>
<td>600 MWth</td>
<td>10 MWth</td>
<td>1 MWth</td>
</tr>
<tr>
<td><strong>Time Without Refuel</strong></td>
<td>~1.5-2 years</td>
<td>~20 years</td>
<td>~10 years</td>
</tr>
<tr>
<td><strong>Core Diameter</strong></td>
<td>6.66 m</td>
<td>2.76 m</td>
<td>0.62 m</td>
</tr>
<tr>
<td><strong>Core Height</strong></td>
<td>7.93 m</td>
<td>7.158 m</td>
<td>0.62 m</td>
</tr>
<tr>
<td><strong>Core Hydraulic Diameter</strong></td>
<td>8.5 mm</td>
<td>8 mm</td>
<td>6.6 mm</td>
</tr>
<tr>
<td><strong>Neutron Spectrum</strong></td>
<td>Thermal</td>
<td>Thermal</td>
<td>Thermal</td>
</tr>
<tr>
<td><strong>Power Cycle Design</strong></td>
<td>Closed Brayton</td>
<td>Closed Brayton</td>
<td>Open Brayton</td>
</tr>
<tr>
<td><strong>Power Cycle Fluid</strong></td>
<td>Helium</td>
<td>Helium</td>
<td>Air</td>
</tr>
<tr>
<td><strong>Core Working Fluid</strong></td>
<td>Helium</td>
<td>Helium</td>
<td>Helium</td>
</tr>
<tr>
<td><strong>Moderator</strong></td>
<td>Graphite</td>
<td>Graphite</td>
<td>Beryllium Oxide</td>
</tr>
<tr>
<td><strong>Fuel Type</strong></td>
<td>TRISO Compact</td>
<td>TRISO Compact</td>
<td>Uranium Carbide</td>
</tr>
<tr>
<td><strong>Fuel Geometry</strong></td>
<td>Annular</td>
<td>Cylindrical</td>
<td>Cylindrical</td>
</tr>
<tr>
<td><strong>Fuel Cladding</strong></td>
<td>Graphite</td>
<td>Graphite</td>
<td>Silicon Carbide</td>
</tr>
</tbody>
</table>

The Finite Element Method (FEM) is a powerful tool used to approximate partial differential equations in physics for engineering applications. Due to the FEMs accuracy, flexibility, and ability to solve complex problems with simple total solutions, FEM became a mainstay of engineering design and analysis. The FEM consist of the following steps:

1. Dividing the domain of interest into subdomains or finite elements.
2. Applying specific function(s) at each finite element.

3. Reconstructing the finite elements to generate global function(s) to be solved.

The dividing of the domain is known as meshing, which can separate the domain into many or few finite elements. The more finite elements created, the more accurate the results would be, but more computational power is required with more finite elements. The functions applied at a finite element vary between algebraic and differential equations based on the conditions within the domain is being solved. The global functions are created by the underlying finite element functions based on each finite element’s orientation in the domain. It is essential to realize that the FEM “solution” is not a single numerical answer but a function that can be applied across the entire domain. Before the advent of modern computers, the FEM was limited to hand approximations, but as computational power increased, so did the capability of the FEM.

The Nuclear Regulatory Commission (NRC) uses three primary codes to evaluate the thermal-hydraulic behavior during normal and abnormal reactor operation: 1. TRAC/RELAP Advanced Computation Engine (TRACE), 2. The Symbolic Nuclear Analysis Package (SNAP), and 3. Reactor Excursion and Leak Analysis Program (RELAP5). The simulations conducted with these codes provide lessons learned that guide design, operation, and safety [15]. These highly sophisticated codes are primarily set to simulating water-cooled reactors, which is why the U.S. Department of Energy conceived the Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program. The NEAMS Program aims to develop a complete tool kit capable of “pellet-to-plant”
simulations that can simulate performance and safety of fourth-generation power plant designs [16]. Within the NEAMS toolkit lies the Multiphysics Object Oriented Simulated Environment (MOOSE) computational framework.

MOOSE was developed by Idaho National Laboratory (INL), which combines C++ programming language and the FEM to simultaneously generate simulations for various physical phenomena. MOOSE’s primary advantage over previous attempts of creating a multiphysics environment is the simplification of its coupling capabilities. Traditional FEM computer codes were typically proprietary, solved individual physical phenomena, and then connected post-production. This traditional method proved time-consuming and prone to error due to incompatibility and being designed by different organizations using different methods. In contrast, a MOOSE-based application can be seamlessly coupled with any other MOOSE-based application because it would utilize the same underlying functions that simulate a given physical phenomenon [17, p. 2]. MOOSE itself is a compilation of modules designed to solve parts of a simulation, including heat conduction, fluid flow, material properties, initial conditions, and boundary conditions are just a few. These modules can be accessed by multiple MOOSE-based applications and run in parallel, limited only by a user’s processing power [17, p. 3]. Since the creation of MOOSE, numerous applications have been created and tested using the MOOSE’s framework; figure 2.1 depicts applications available as of 2020.
Engineering a nuclear reactor requires linking multiphysics phenomena, including neutronics, thermal-hydraulics, material science, chemical science, to name a few. The coupling of all physics occurring in a nuclear reactor is the goal of every nuclear power plant design to safely and efficiently produces energy. This work will focus on thermal-hydraulic and system analysis to provide design guidance to future iterations of the MINION-1 concept with the most advanced toolkit available – SAM.

Under the NEAMS and MOOSE umbrella lies another unique tool; the System Analysis Module (SAM). Argonne National Laboratory (ANL) has been developing a modern MOOSE-based system analysis code to simulate and analyze next-generation non-Light Water Reactors (LWRs). The System Analysis Module’s (SAM) goal is to provide whole-plant transient simulation capability while limiting the need for significant computational power. To reduce computational power, SAM can coupling 1D, 2D, and
3D simulations, allowing users to minimize the number of finite elements required to mesh a domain. SAM’s focus on non-LWR allows the flexibility to model several single-phase coolants’ fluid dynamics and heat transfer, and its dependence on the MOOSE underlying meshing, finite-element library, linear and non-linear solvers make it an ideal candidate for this work [19, pp. 2-3].
CHAPTER 3

UOFSC MINION-1 REACTOR DESCRIPTION

The MINION-1 consists of one primary loop cycle and one open Brayton cycle connected by a counterflow heat exchanger. The primary loop is designed to remove the 1 MWth of energy from the core, consisting of a pump circulating helium, the core heating the helium, and a pressurizer maintaining system pressure. The open Brayton cycle consists of the power conversion sequence, which contains a compressor and gas turbine on a common shaft. Figure 3.1 below illustrates both cycles' main components and the cold and hot legs.

Figure 3.1: MINION-1 Power Plant Concept

The primary loop consists of the upper plenum, core, lower plenum, primary side of the heat exchanger, pump, and pressurizer. Helium is pumped into the reactor pressure
vessel’s upper plenum, then down the core, then into the heat exchanger, and circulated back into the pump.

The MINION-1 core concept consists of 55 hexagonal unit cells composed of a single cylindrical fuel rod surrounded by an annular coolant channel and hexagonal moderator. The reactor pressure vessel (RPV) is made of stainless steel-303 contains the core, control barrels, and upper and lower plenums having a volume of 0.45 m$^3$ each. Figure 3.2 illustrates the cross-sections of the MINION-1 core concept, unit cell, and internal geometry of the RPV.

![Figure 3.2: Vertical and Horizontal Core Cross-Section Concept](image)

The core unit cell consists of a single fuel pin surrounded by a coolant and neutron moderator. The fuel used is Uranium Carbide (UC) with a 1.5 cm diameter, is surrounded by a 0.02 cm thick gap filled with Helium-4 (He-4) at 0.3 MPa and a 0.1 cm thick Silicone Carbide (SiC) cladding. The fuel pin is surrounded by an annular coolant
channel that is 3.3 mm thick and filled with Helium-4 with an active core length of 62 cm. Figure 3.3 provides a dimensional illustration of the unit cell.

<table>
<thead>
<tr>
<th>Component</th>
<th>Thickness</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel-Clad Gap</td>
<td>0.02 cm</td>
</tr>
<tr>
<td>SiC Clad</td>
<td>0.1 cm</td>
</tr>
<tr>
<td>Coolant Channel</td>
<td>0.33 cm</td>
</tr>
</tbody>
</table>

Figure 3.3: Unit Cell material and dimensions

The pump dictates the mass flow rate of the primary loop and works in conjunction with the pressurizer to maintain the ~3 MPa system pressure. The pump must maintain the helium mass flow constant under normal operating conditions but has no safety-related task under accident conditions.

The primary loop consists of the compressor, secondary side of the heat exchanger, and turbine. Ambient air is compressed, forced through the primary side of the heat exchanger, and expanded in the turbine completing the open Brayton cycle.
The compressor compresses ambient air from ~0.1 MPa (1 atm) to 0.5 MPa (~4.9 atm) and maintains a designated mass flow rate. After flowing through the heat exchanger, the compressed and heated air will enter a turbine and expand, turning the turbine shaft.

The difficulty in selecting a heat exchanger (HX) is balancing heat transfer effectiveness and size. A primary goal of the MINION-1 is to maximize mobility; therefore, minimizing the HX size takes priority. A promising and proven technology that fits this criterion is the Printed Circuit Heat Exchanger, a compact and highly effective heat transfer medium. The PCHE design compacts the large heat transfer area necessary, achieving one of the industry's highest heat transfer area densities.

Various materials have been selected across the core design that balance performance and size to achieve the MINION-1 high mobility goals.
The thermal performance of the fuel pin is crucial to the overall performance and safety of the entire system. Within the fuel, pin lies one of the most critical choices, the UC fuel. UC fuel has shown stellar performance through extensive testing due to its high uranium density, irradiation stability, and high thermal conductivity. The primary disadvantage is the significant swelling UC experiences during its working life, leading to unwanted fuel-clad interactions [20]. Using pressurized He-4 has been a staple in the nuclear power industry for decades and was selected for both being inert and improving thermal conductivity across the fuel-clad gap. Silicon Carbide was selected as the cladding due to its high melting point, high strength at elevated temperatures, relative stability in radiation, and low neutron absorption cross-section. When exposed to thermal irradiation, SiC thermal conductivity degrades below traditional zircaloy cladding and is brittle. The largest risk with this fuel pin combination is the fuel swelling causing cladding failure during the ten-year life expectancy of the fuel.

MINION-1 core concept is being designed to output high core exit temperatures to increase thermal efficiency, but a unique challenge for gas-cooled reactors is the naturally low heat transfer coefficients. The cores heat transfer coefficient must be optimized to ensure fuel pins temperatures remain at acceptable levels through artificial cladding surface roughening. This practice has been extensively researched to improve the Advance Gas Reactors (AGRs), which use CO₂ coolant, allowing an avenue of future optimization [21].
Although not modeled in this research, the material selection for the moderator is essential to simulate whole core transients. Beryllium oxide’s excellent performance as a moderator/reflector, high thermal conductivity, and relatively low atomic ratio with fissile material are reasons for its selection. For beryllium to moderate sufficiently in a core, it requires at least a 2000 to 1 ratio against fissile atoms compared to graphite which requires a 4000:1 ratio [22]. Beryllium’s low atomic ratio allows minimization of the MINION-1 designed dimensions and mass.
CHAPTER 4

METHODOLOGY

In this work, the MOOSE-based SAM code was used to model and simulate the MINION-1 primary loop and provide boundary conditions for an analysis of the open Brayton power cycle. A unit-cell approach is utilized to model the normal operating conditions, which applies a single unit cell’s dimensions and geometry to approximate the behavior of the core. This methodology was adopted from ANL’s modeling of the GT-MHR as a unit cell for steady-state analysis [23]. The representation of the MINION-1 using the unit cell model is illustrated in figure 4.1. The dimensions detailed in chapter 3 were used to simulate the unit cell geometry but are summarized in table 4.1.

Figure 4.1 SAM simulation model
Table 4.1 Unit Cell geometric values

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Fuel Diameter (mm)</strong></td>
<td>15</td>
</tr>
<tr>
<td><strong>Gap and Clad Thickness (mm)</strong></td>
<td>0.2, 1</td>
</tr>
<tr>
<td><strong>Unit Cell Coolant Hydraulic Diameter (mm)</strong></td>
<td>6.6</td>
</tr>
<tr>
<td><strong>Unit Cell Coolant Flow Area (mm$^2$)</strong></td>
<td>214.6022</td>
</tr>
<tr>
<td><strong>Core Flow Area (mm$^2$)</strong></td>
<td>11803.121</td>
</tr>
<tr>
<td><strong>Total Fuel Rods</strong></td>
<td>55</td>
</tr>
<tr>
<td><strong>Active Core Height (cm)</strong></td>
<td>62</td>
</tr>
<tr>
<td><strong>Upper and Lower Plenum Volume (m$^3$)</strong></td>
<td>0.452389</td>
</tr>
<tr>
<td><strong>Upper and Lower Plenum Surface Area (m$^2$)</strong></td>
<td>1.130973</td>
</tr>
<tr>
<td><strong>Core Surface Area Density (m$^2$/m$^3$)</strong></td>
<td>254.72</td>
</tr>
<tr>
<td><strong>Pressure (MPa)</strong></td>
<td>3</td>
</tr>
<tr>
<td><strong>Power (MW)</strong></td>
<td>1</td>
</tr>
</tbody>
</table>

SAM provides the ability to input custom power distribution functions, and therefore arbitrary but more realistic than an unreflected power distribution was utilized, plotted in figure 4.2. The arbitrary power distribution is inspired by the work of Hao Sun et al. on a 2.4 MWth microreactor using BeO reflector and control barrels, this reactor’s calculated minimum, maximum, and average power distribution it plotted in figure 4.3.
Figure 4.2 Simulated axial power distribution

Figure 4.3 Axial power distribution from 2.4 MWth microreactor [24]
The pump's primary function is to ensure the inputted mass flow rate is achieved and maintained, varying the pump head as necessary until the flow is maintained. Calculating a desired mass flow rate is dependent on the specific heat capacity formula, where $q$ is core thermal power, $C_p$ is the specific heat of helium, $\Delta T$ is the change in temperature between core exit and entrance:

$$q = \dot{m} C_p \Delta T$$

$$\frac{q}{C_p \Delta T} = \dot{m}$$

It is necessary to calculate the power required to the pump at steady-state, which will be calculated using the pressure differences of the entire system to analyze system efficiency. According to Todreas and Kazimi, pumping power is the “work to raise coolant pressure per unit time expresses as a force through distance per unit time” or force times velocity [25]:

$$\frac{(\Delta p) v}{\eta_{pump}} = (\Delta p) \frac{\dot{m}}{\rho \eta_{pump}} = W_{\text{Pump}}$$

The primary function of the heat exchanger is to transfer the heat from the primary side to the secondary side. Based on the work of W. Kim et al., a PCHE with a length of rough 1.176 m, counterflow design, and a hydraulic diameter less than 2 mm will provide the optimum effectiveness of ~95% [26]. Figure 4.4 illustrates the effectiveness of different PCHE designs.
The heat transfer surface area density in a heat exchanger is the surface area per fluid volume \([\text{m}^2/\text{m}^3]\). PCHEs can be designed to have surface area densities greater than 2500 \(\text{m}^2/\text{m}^3\), easily qualifying them as a compact heat exchanger, compared to a typical shell in tube heat exchanger having a maximum of 1000 \(\text{m}^2/\text{m}^3\) surface area density [27, p. 4861]. A surface area density of 1000 \(\text{m}^2/\text{m}^3\) was chosen to minimize the size of the heat exchanger to minimize the expected HX size and maximize mobility of the entire power plant.

The individual heat transfer coefficients of the primary and secondary sides are critical to the overall system's performance. Due to the moderate-fidelity nature of the SAM code, a value from the literature had to be selected to produce realistic results. According to the work of Chen et al., supercritical CO\(_2\) are capable of achieving coefficients as high as \(\sim 675 \text{ W/m}^2\text{K}\), see figure 4.5 for the plot of the empirical data [28]. A heat transfer coefficient of 500 \(\text{W/m}^2\text{K}\) was selected to provide a non-optimized start
point for a more comprehensive Computational Fluid Dynamic Model (CFD) to be done in the future.

![Figure 4.5 Experimentally achieved overall heat transfer Coefficients with CO₂ [28]](image)

The helium entering the primary side of the heat exchanger contains 1000 kWth of energy which must be moved to the secondary side for the system to achieve a steady-state. According to the heat capacity formula, see below, a lower mass flow rate results in a higher temperature difference, indicating a lower fluid velocity allows more effective heat transfer to a specific volume of fluid. Since the primary mass flow rate is already determined based on the heat removal requirements of the core, the only parameter of the HX we can manipulate is the total flow area. To decrease the velocity, therefore increase the temperature difference, but maintain the same mass flow rate in the primary side of the HX, the flow area is double the size of the core’s total flow area.
\[
\frac{q}{C_p \Delta T} = \rho v A = \dot{m}
\]

\[
v = \frac{\dot{m}}{\rho A}
\]

Inside the HX, the 1000 kWth flowing through the primary side must be transferred to the secondary, which is the function of the mass flow rate of the secondary fluid – air.

To properly conduct a complete plant analysis, it is necessary to calculate the net amount of work from the open Brayton cycle. First the energy required to compress air must be calculated and then how much energy is generated by the turbine through the expansion of the compressed and heated air. Using the heat capacity formula and the same temperature difference from the primary side, we can solve for the required mass flow necessary to accept 1000 kWth from the primary side of the heat exchanger due to expected HX effectiveness.

An adiabatic compressor compresses air from 101325 Pa (1 atm) and 295 K (21.85 C) to a pressure of 500000 Pa (4.93 atm) at a mass flow rate determined by the conditions of the primary side. The temperature of state 3s (T3s) will be an output of the SAM simulation dependent on given conditions. The following assumptions are made to simplify the calculations: compressor isentropic efficiency of 80%, steady-state exists, the air is an ideal gas, kinetic and potential energy changes are negligible. State 1s and 2s enthalpies will remain constant regardless of mass flow rate; therefore, it can be
calculated here. Using a table of ideal-gas properties, we can interpolate and calculate the change in enthalpy, change in temperature, and compressor work with the following method [29, pp. 936-937]:

![Figure 4.6 Compressor States](image)

T = Temperature (Kelvin), h = enthalpy (kJ/kg), Pr = relative pressure (-), \( \dot{W} \) = Work (kW), \( \dot{m} \) = mass flow of air (kg/s)

\[
T_1s = 295 \text{ K} \rightarrow h_{1s} = 295.170 \text{ kJ/kg} \rightarrow Pr_{1s} = 1.3068 \quad \text{[Table Lookup]}
\]

\[
Pr_{2s} = Pr_{1s} \left( \frac{P_{2s}}{P_{1s}} \right) = 6.45
\]

\[
Pr_{2s} = 6.45 \rightarrow h_{2s}, isentropic = 466.205 \text{ kJ/kg} \quad \text{[Table Interpolation]}
\]

\[
0.80 = \eta_c = \left( \frac{h_{2s}, isentropic - h_{1s}}{h_{2s}, actual - h_{1s}} \right)
\]

\[
h_{2s, actual} = 508.96 \text{ kJ/kg} \rightarrow T_{2s} = 505.7 \text{ K} \quad \text{[Table Interpolation]}
\]

\[
\dot{W, compressor} = \dot{m} (h_{2s, actual} - h_{1s})
\]
Air enters the adiabatic turbine at a steady mass flow rate and temperature and exits at a pressure of 101325 Pa (1 atm). The following assumptions are made to simplify calculations: isentropic turbine efficiency of 96%, steady-state exists, the air is an ideal gas, kinetic and potential energy changes are negligible. As the calculation of compressor work, we can use ideal-gas air tables to calculate the change in enthalpy, change in temperature, and turbine work output with the following method [29, pp. 936-937]:

![Figure 4.7 Turbine States](image)

\[ T = \text{Temperature (Kelvin)}, \ h = \text{enthalpy (kJ/kg)}, \ \text{Pr} = \text{relative pressure (-)}, \ \dot{W} = \text{Work} \ (\text{kW}), \ \dot{m} = \text{mass flow of air (kg/s)} \]

\[ T3s \to h3s \to Pr3s \quad [\text{Table Lookup}] \]

\[ Pr4s = Pr3s \left( \frac{P4s}{P3s} \right) \]

\[ Pr4s \to h4s, \text{isentropic} \]

\[ 0.96 = \eta_T = \frac{h3s - h4s, \text{actual}}{h3s - h4s, \text{isentropic}} \]

\[ h4s, \text{actual} \to T4s \]
\[ W_{\text{Turbine}} = \dot{m} (h_3 - h_{4s, \text{actual}}) \]

The following approach was used to determine plant efficiency:

\[
W_{\text{net}} = W_{\text{Turbine}} - W_{\text{compressor}}
\]

\[
\eta = \frac{W_{\text{net}} - W_{\text{Pump}}}{1000 \text{ kW}}
\]

The Carnot efficiency formula reveals that low plant temperatures result in low efficiencies.

\[
\eta_{\text{Carnot}} = 1 - \frac{T_{\text{Cold}}}{T_{\text{Hot}}}
\]

\( T_{\text{Cold}} \) is determined by atmospheric conditions and therefore varies based on a power plant’s local climate. \( T_{\text{Hot}} \) becomes the only variable that a power plant design can affect to increase overall efficiency; therefore, the MINION-1 concept needs to reach the highest possible core exit temperature and heat exchanger primary side entrance. A maximum core exit temperature must be balanced with safe operating temperatures for the fuel, 1250°C (1523.15 K) is the target maximum fuel temperature while 750°C (1023.15 K) was selected as the target core exit temperature. Using the thermal resistance circuit analogy for a cylindrical fuel pin presented in figure 4.8, we can calculate the heat transfer coefficient required to maintain fuel temperatures at ideal conditions.
The target heat transfer coefficient, Nusselt, Prandtl, and Reynolds number can be calculated using the following relationships:

\[ D_h = \text{Core Hydraulic Diameter}, \quad h_{\text{target}} = \text{target heat transfer coefficient}, \quad C_p = \] Specific Heat of Helium, \( \mu = \) Helium dynamic viscosity, \( k_{\text{He}} = \) helium heat capacity

\[ Nu = \frac{h_{\text{target}} D_h}{k_{\text{He}}} \]

\[ Pr = \frac{C_{p,\text{He}}}{} \]

\[ Re = \left[ \frac{Nu}{0.023(Pr^{0.4})} \right]^{1.25} \]

Using the Reynolds number formula, we can calculate the target velocity in the core:
\( \rho = \) helium density

\[ v = \frac{Re \mu}{D_h \rho} \]

We can then calculate the required mass flow with the following formula:

\[ A = \text{unit cell cross-sectional flow area} \]

\[ \dot{m} = \rho v A \]

55 \( x \) \( \dot{m} \) = Core Mass Flow Rate = \( \dot{M} \)

Finally, we can then calculate a friction factor necessary to create optimum turbulent conditions to maximize heat transfer by using the following Blasius correlation:

\[ f = \frac{0.316}{Re^{0.25}} \]

Table 4.2 below summarize the values calculated above:
Table 4.2 Calculated target heat transfer values

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$h_{target}$</td>
<td>2.304 kW/m².K</td>
</tr>
<tr>
<td>$Nu$</td>
<td>48</td>
</tr>
<tr>
<td>$Re$</td>
<td>16903</td>
</tr>
<tr>
<td>$v$</td>
<td>64 m/s</td>
</tr>
<tr>
<td>$\dot{m}$</td>
<td>0.02357 kg/s</td>
</tr>
<tr>
<td>$\dot{M}$</td>
<td>1.2965 kg/s</td>
</tr>
<tr>
<td>$f$</td>
<td>0.0277</td>
</tr>
</tbody>
</table>

With the mass flow rate of helium through the primary loop determined, we can then calculate the mass flow rate of air through the open Brayton cycle. First, we use the specific heat capacity formula to solve for the change in temperature in helium:

$$\frac{Q_{th}}{\dot{m}_{He} C_{p,He}} = \Delta T$$

Now, we plug the $\Delta T$ into the specific heat capacity formula for air and calculate mass flow:

$$\frac{Q_{th}}{C_{p,Air} \Delta T} = \dot{m}_{Air}$$

Now, we solve for the velocity boundary condition using the following formula:

$$\frac{\dot{m}}{\rho A} = v_{Air}$$
Table 4.3 Calculated target air heat transfer values

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\Delta T$</td>
<td>148.60 K</td>
</tr>
<tr>
<td>$\dot{m}_{Air}$</td>
<td>1.2966 kg/s</td>
</tr>
<tr>
<td>$v_{Air}$</td>
<td>12.85 m/s</td>
</tr>
</tbody>
</table>
CHAPTER 5

RESULTS

A transient solver within SAM was utilized to run this steady-state simulation. Figure 5.2 displays the temperature and velocity profiles of the MINION-1 primary loop at steady state. Coolant flows downward in the core during normal operating conditions and coolant temperatures increase from the top to the bottom of the core. The increase in temperature decreases the coolant density which results in a higher velocity towards the exit of the core. It is also apparent in figure 5.1 that each scales illustrates that coolant temperatures and velocities are not reaching target values outlined in table 4.2 and figure 4.8. The temperature scale peaks out around 660 K while the target core exit temperature was 1023.15 K and the velocity scale peaks at 44 m/s while target core velocity was 64 m/s. Using post processors and plotting tools the following is the analysis as to why target values are not met.
Figure 5.1 Unit-Cell Model Temperature and Velocity Profiles

First, the figure 5.2 shows the temperature profile of the fuel pin at various axial locations and reveals that the fuel pin profile has a higher temperature profile at the exit of the core compared to the entrance. This indicates that the power in the fuel has shifted towards the exit of the fuel channel, this phenomenon is normal and expected due to the direction of flow of coolant from the top to the bottom and does not impact performance significantly.
Next, figure 5.3 plots the maximum temperatures in the fuel pin over time, indicating that the primary loop reaches a steady-state 20 mins after starting. We also notice that the fuel reaches a maximum temperature that is 76 degrees higher than our target while the coolant maximum temperature is 359 degrees below target. The core data has been broken down into four categories in figure 5.4: temperature, coolant density, channel velocity, and heat transfer coefficient.
As temperature in the core increases, coolant density decreases, causing velocity and turbulence to increase. The increased turbulence directly increases the local heat transfer coefficient but achieving a maximum of $967 \text{ W/m}^2\text{K}$, over $1000 \text{ W/m}^2\text{K}$ short of the target. Based on the plotted data, the target velocity and therefore target turbulence is not achieved which insulates the fuel more leading to higher fuel pin temperatures and lower coolant temperatures.

A reflexive solution would be to increase the mass flow rate which in turn increases the velocity in the core, leading to higher turbulence and heat transfer. Unfortunately, recalling the heat capacity formula, an increase in mass flow will decrease coolant temperatures resulting in a higher coolant density creating difficult conditions for coolant to achieve ideal velocity. Even more importantly the exit temperature of the core specifically will suffer significantly reducing the systems thermal efficiency. The opposite, decreasing the mass flow rate, will increase coolant temperatures and decrease
density but the increase in temperature is a result of higher fuel pin temperatures. With current boundary conditions the maximum fuel temperature reaches 76 degrees above target, decreasing the mass flow rate will force fuel temperatures to depart further into dangerously high levels.

Using the HX outputs, plotted over time in figure 5.5, and the methodology outline in chapter 4, we can calculate the net amount of work output but the open Brayton cycle. First, the work required to compress air from atmospheric to 0.5 MPa at a rate of 6.57 kg/s is 1123.96 kW. Using the secondary sides exit temperature for air, 597 K (324 C), and the same mass flow rate, we can determine a turbine work output of 1366.79 kW. The temperature of the air exiting the turbine is 395 K (122 C).
The work required by the pump in the primary loop is necessary to calculate to determine the overall system efficiency. Based on the methodology outline in chapter 4, using the pressure drops in the core and heat exchanger and assuming a pump efficiency of 97%, we calculate a pumping power of 5.12 kW. Finally, with a net amount of work of the compressor and turbine of 242.83 kW, we get an overall system efficiency of 23.77% when pumping power is accounted.
CHAPTER 6
CONCLUSION

A unit cell method was used to approximate the behavior of the MINION-1 core concept to assess steady-state conditions in the primary loop, calculate overall system efficiency and build a foundation for future optimization. The unit cell model simulated the core, lower and upper plenums, heat exchanger, and pump. The performance of the secondary side opens Brayton cycle was not modeled but calculated using the outputs of the simulation. Based on the methods outlined in chapter 4, the system operating under normal operating conditions produces an overall system efficiency of 23.77%.

The SAM code simulated a realistic steady-steady model that requires minimal computational power. The simulation provided data to analyze; revealing several areas for optimizing the MINION-1 design. Through this analysis it became apparent that the SAM code would benefit from developing a power cycle modeling to allow whole power plant analysis.

A significant number of assumptions occurred to either simplify calculations or due to the currently evolving state of MINION-1’s concept. In the primary loop, the arbitrary power distribution is not precise and can significantly affect fuel temperature performance. Next, the pressure losses are only modeled in the core and heat exchanger, which drastically reduces the pumping power required. Heat transfer is also assumed to only occur in the core and heat exchanger indicating 100% of the 1000 kW of thermal power is being transferred to the secondary side when heat is lost in several components.
in the primary loop. Finally, the open Brayton cycle assumes ideal-gas air as the working fluid with high isentropic efficiencies for the components. These assumptions superficially inflate the overall system efficiency but create a foundation to guide the MINION-1 design and increase the fidelity of the simulation.

In the primary loop, future work should precisely calculate the power distribution both axially and radially, ensuring more accurate fuel temperature gradients. Surface roughening plays a significant role in increasing heat transfer coefficients in gas-cooled reactors to keep fuel temperatures low; optimizing cladding roughness should be pursued. While steady-state results can be achieved using a unit cell simulation, it is unreasonable to expect accurate accidents transients to converge to a solution. The unit cell simulation does not consider a radial temperature or power distribution or model a passive cooling system to remove decay heat. A higher fidelity modeling can be pursued using the SAM framework.

Future work should consist of optimizing the heat exchanger to increase effectiveness by varying hydraulic diameters, flow areas, surface area densities, and friction factors in the secondary cycle. The generic compressor used can also be optimized using several smaller compressors with intercooling to reduce temperature and total work to compress air. The team at ANL has plans to develop power cycle modeling capabilities within SAM to allow full-plant analysis; assisting in this endeavor would be beneficial to this work.
REFERENCES


